안전정기지진하의 원자로내부구조물 거동분석

Dynamic Behavior of Reactor Internals under Safe Shutdown Earthquake

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요 약

원자력발전소 부품중 안전과 관련된 구조물은 지진하중하에서 그 건전성을 유지하도록 설계되어야 한다. 그중 원자로내부구조물부품은 1차 내진분류에 속하는 것으로써 지진하중하에서의 건전성이 발전소 안전과 경제적인 관점에서 매우 중요하다. 지금까지 이러한 원자로내부구조물의 모델링에 대해서는 여러 사람들에 의해 연구되고 발표되었으나, 본 논문에서는 국내 발전소 중에서 Turn-key base로 건설되어 이미 가동 중에 있는 영광 1&2호기의 원자로내부구조물에 대한 안전정지지진하의 거동을 Global Beam Model이라는 단순화된 모델을 이용하여 분석하였다. 이 모델의 설정을 위해서 주요부품들을 double pendulum의 보요소로 표현하였고, 이들 주요부품들의 특성해석을 범용유한요소해석코드인 ANSYS에 의해 구하여 이를 상부 및 하부에서 간격을 갖는 비선형스프링으로 모델링하였다. 또한 이 비선형스프링뿐아니라 원자로용기와 원자로내부구조물부품들 사이의 유체동적현상을 묘사한 유체동력학적 coupling에 의해 pendulum의 보요소를 서로 연결시켜 모델링을 하였다. 가진자료인 안전정지하중은 영광 1&2호기의 원자로용기 지지부에 가해지는 응답스펙트럼을 시간이력함수로 바꾸었으며, 이 모델과 가진 하중을 가지고 비선형해석 code인 KWUSTOSS의 explicit Runge-Kutta-Gills algorithm을 이용하여 적분을 수행하므로써 안전정지지진하의 원자로 내부구조물에 대한 거동을 구하여 이 구조물의 주요부품에 대한 내진검증 및 구조물 내부에 있는 핵연료집합체의 내진해석을 위한 입력자료를 확보할 수 있었다. 그리고 본 연구에서 사용된 Global Beam Model의 간편성 및 효율성과 explicit Runge-Kutta-Gills algorithm에 대한 경제성을 확인할 수 있었다.

Abstract

The safety related components in the nuclear power plant should be designed to withstand the seismic load. Among these components the integrity of reactor internals under earthquake load is important in stand points of safety and economics, because these are classified to Seismic Class I components. So far the modelling methods of reactor internals have been investigated by many authors. In this paper, the dynamic behaviour of reactor internals of Yong Gwang 1&2 nuclear power plants under SSE(Safe Shutdown Earthquake) load is analyzed by using of the simplied Global Beam Model. For this, as a first step, the characteristic analysis of reactor internal components are performed by using of the finite element code ANSYS. And the Global Beam Model for reactor internals which includes beam elements, nonlinear impact springs which have gaps in upper and lower positions, and hydrodynamical couplings which simulate the fluid-filled cylinders of reactor vessel and

이 논문에 대한 토론을 1995년 3월 31일까지 본 학회에 보내주시면 1995년 9월호에 그 결과를 게재하겠습니다.

core barrel structures is established. And for the exciting external force the response spectrum which is applied to reactor support is converted to the time history input. With this excitation and the model the dynamic behaviour of reactor internals is obtained. As the results, the structural integrity of reactor internal components under seismic excitation is verified and the input for the detailed fuel assembly series model could be obtained. And the simplicity and effectiveness of Global Beam Model and the economics of the explicit Runge-Kutta-Gills algorithm in impact problem of high frequency interface components are confirmed.

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. Thus the knowledge of the dynamic behaviour under accident loads is important for the design of the reactor internals components.

The modelling method for the dynamic behaviour analysis of reactor internals in PWR with impact occuring between various components have been studied : A.N.Nahanvandi and G.J.Bohm introduced the simplified single vertical beam model considering the components to be a continuous network and striking against massless spring[1]. And they developed the model to two vertical beam model subjected to a forced vibration at the upper ends and striking against each other at the lower end. But they did not consider the hydroelastic effect in the model[2]. The study on coupled fluid-structure dynamics of a thin cantilevered shell which simulate the fluid filled cylinders of reactor vessel and core barrel was conducted by B.E.Schineider and J.A.Stevens. They compared the finite difference coupled technique in KWU and a finite element-added mass technique in Combustion Engineering, and they showed that both of two techniques were shown to given good agreement[3]. But in those both cases they did not simulate the differently moving fluids of two cylinders separated by fluid filled gap. In domestic case, in the past the analysis technique of reactor internals behaviour under excitation force was not established owing to the lack of design information and analysis technique. But recently the lumped mass model which connsists of the beam elements, the gap elements in many positions and hydrodynamicic effect has been used since Yong Gwang 3&4 nuclear power plants[4].

The objectctive of this study is to present a Global Beam Model(GBM) shwon in Fig.1, impact computational technique, and the analysis results for the Yong Gwang 1&2 900Mwe reactor internals consisting of reactor vessel, core barrel, vessel support and fuel assemblies. Principally the reactor internals system has a complex vibration system capable of carrying out pendular and torsional motions of the pressure vessel and core barrel, bending motions of the core plates etc. It was shown by aid of the vibration measurement test that GBM plane model of double pendulum is sufficiently exact for the computation of dynamic behaviour of

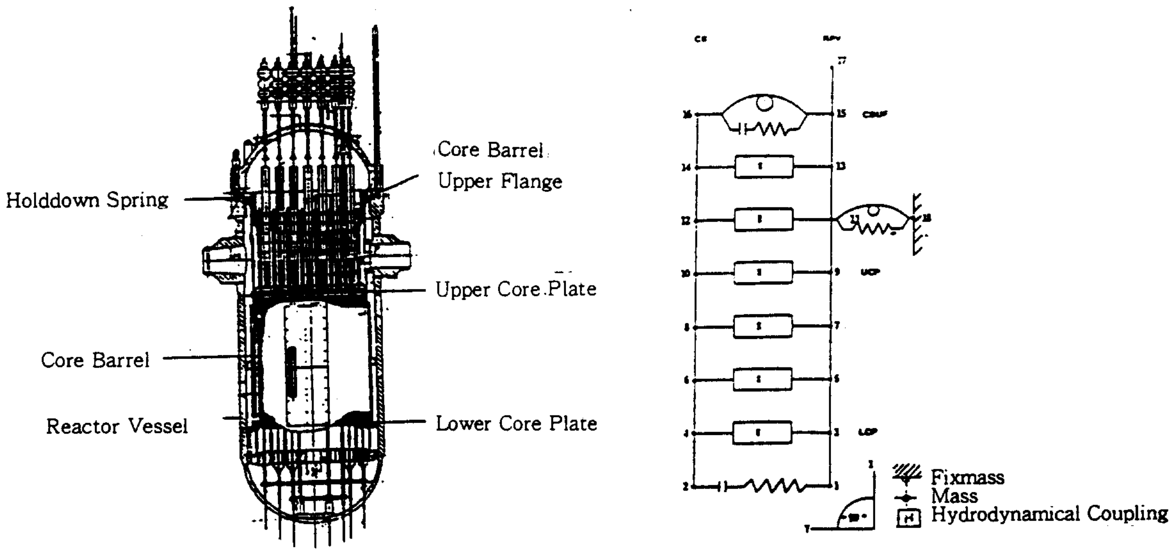


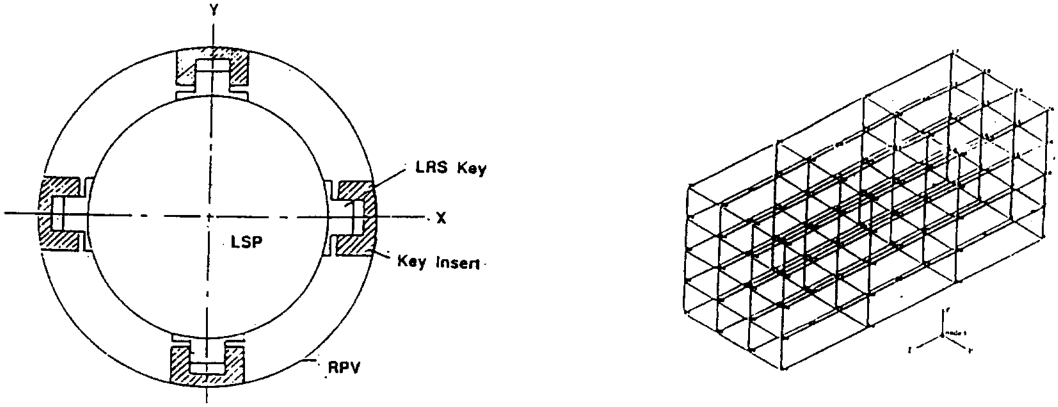
Fig. 1 Reactor Internals Model(Global Beam Model)

reactor internals[5]. In GBM modelling, the reactor internal components such as upper guide structure, upper core support structure, lower core support structure etc. are represented as a beam element. And fuel assemblies are also represented as a single beam element without considering the gap between fuel assemblies. And the gaps between reactor vessel and core barrel structures in upper and lower positions are represented as the impact spring. Taking into account the hydrodynamic effect of the mobil fluid-filled cylinders simulating the reactor vesel and core barrel structures the hydrodynamic coupling is connected between two vertical pendulum beams. After finishing the modelling in computational precedure with the external forcing function the explicit modified Runge-Kutta-Gills method is used to integrate the equation of motion in order to reduce the computing time to 1/10 of the other standard algorithms.

For the dynamic behaviour analysis of reactor internals of Yong Gwang 1&2 nuclear power plants, in the present study, the reactor internals are modeled for seismic analysis by using the GBM modelling method, of which the characteristics are obtained by using of the finite element code 'ANSYS'[6].

After the model is established, the modal analysis is performed to examine whether the inputs are properly prepared and the calculate the structural damping values. The displacement time history converted from acceleration response spectrum of SSE of Yong Gwang 1&2 nuclear 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 it is shown that the modified explicit method is good to the integration of nonlinear impact equation of motion of the high frequency system oscillation components. Through this



(a) Lower Radial Support Key Arrangement

(b) Finite Element Mesh

Fig. 2 The Arrangement of Lower Radial Support Key(Top View) and Finite Element Mesh

analysis the responsive motions of the upper core plate and lower core plate as the outputs are obtained, which can be used as an input for the analysis of the core model.

2. ANALYTICAL METHOD

2.1 Model Development

The reactor internals model(Global Beam Model) consists of beam elements, nonlinear impact springs, rotational spring and hydrodynamic couplings : The nonlinear impact springs which simulate the gaps between the radial restraint and lower radial clevis insert / between the core barrel upper flange and reactor vessel have nonlinearties because of the gap. The finite element code 'ANSYS' are used to obtain the spring characteristics of these components. Fig. 2&3 show the arrangement of lower radial support key, as an example, and its spring characteristic obtained by using ANSYS code.

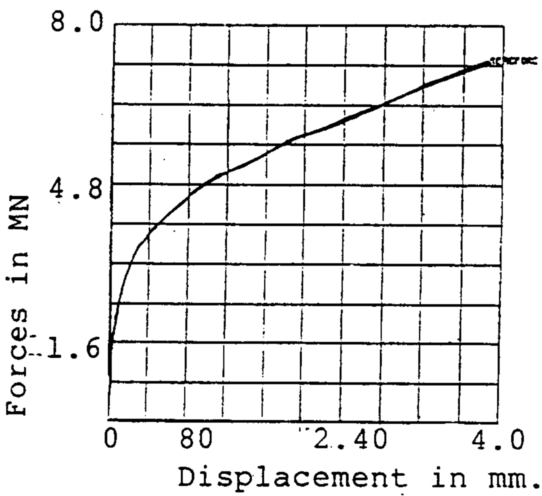


Fig. 3 The Spring Characteristic of Lower Radial Support Key

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(refer Fig. 4). For the calculation of the rotational spring characteristic, the spring characteristic of each component of core barrel upper part is obtained by using of ANSYS code as shown in Fig. 5 and combined the each characteristic. The rotational spring characteristic of core barrel upper part is shown in Fig. 6.

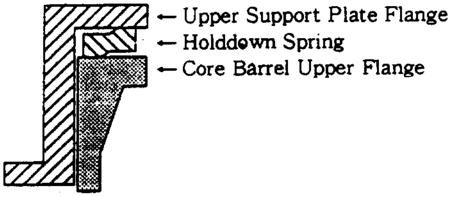
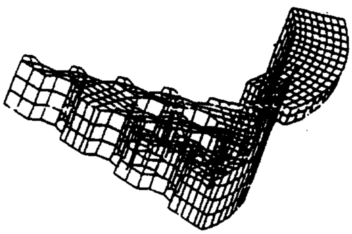
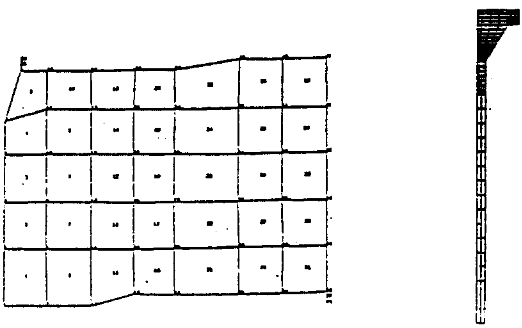


Fig. 4 The Combination of Upper Support Part



(a) Upper Support Plate(1/8 section)



(b) Holddown Spring　(c) Core Barrel Upper Flange

Fig. 5 Finite Element Mesh of Upper Support Part

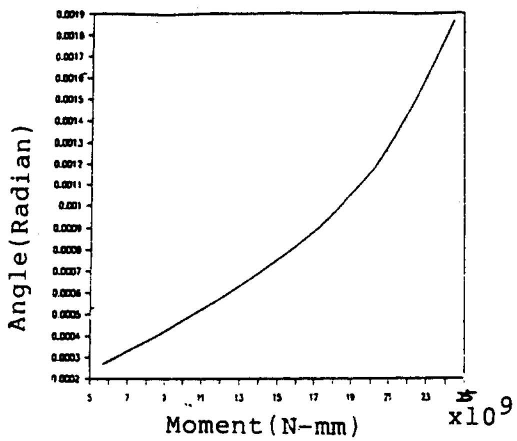


Fig. 6 The Rotational Spring Characteristics of Upper Support Part

And the hydrodynamic couplings simulate the fluid-filled gaps between the core barrel and reactor vessel according to the position of cylinder segments. Since the dynamic behaviour of structures which consist of two concentric cylinders separated by a fluid-filled gap is strongly affected by interaction with the fluid, in this study, the fluid-structure interaction is modelled by inclusion of an additional mass matrix, hydromatrix, in the equation of motion. In the other model cases the hydrodynamic effects in the case of two infinitly long cylinders which move uniformly over their entire length is stated[4][7]. But in this paper, the hydromatrix for the case of finite length two cylinders which are capable of differing motion their entire is used[8]. If the motions of the inner and outer cylinders are represented by K_1 and K_2 points distributed over the length, the corresponding hydromatrix is symmetric$(\bar{K}_1+\bar{K}_2)x(\bar{K}_1+\bar{K}_2)$ matrix which is generally fully occupied, as follows.

$$M_H = \begin{bmatrix} M_{11} & M_{12} \\ M_{21} & M_{22} \end{bmatrix}$$

with

$$(M_{11})_{IJ} = -\frac{2\pi\rho R_1}{L} \sum_{N=1,3}^{\infty} \frac{S_{I,1}^{(N)} S_{J,1}^{(N)}}{a_N \tau_N}$$

$$[I_1(a_N R_1) K_1(a_N R_2) - K_1(a_N R_1) I_1(a_N R_2)]$$

$$(M_{12})_{IJ} = (M_{21})_{IJ} = -\frac{2\pi\rho}{L} \sum^{\infty} \frac{S_{I,1}^{(N)} S_{J,2}^{(N)}}{a_N^2 \tau_N}$$

$$(M_{22})_{I,J} = -\frac{2\pi\rho R_2}{L} \sum_{N=1,3}^{\infty} \frac{S_{I,2}^{(N)} S_{J,2}^{(N)}}{a_N \tau_N}$$

$$[-K_1(a_N R_1) I_1(a_N R_2) - I_1(a_N R_1) K_1(a_N R_2)]$$

where

$$S_{I,P}^{(N)} = \int_{SegI,P} \cos(a_N\, y)\, dy$$

for $I=1, \cdots \overline{K}_P$, where $P=1,2$

Seg I, P : Cylindrical Segments
(P=1 : inner cylinder,
P=2 : outer cylinder)

$\overline{K}_P$: Number of points

$\overline{K}_1$: Number of points on inner cylinder

$\overline{K}_2$: Number of points on outer cylinder

I_1, K_1 : Modified Bessel function of 1st and 2nd Kind

a_N : $n\pi/2L$

R_1 : Radius of inner cylinder

R_2 : Radius of outer cylinder

ρ : Fluid density

Indices(I,J) : Nodes of the fluid coupling

The nodes which represent the mass and position of reactor components are linked by beam element. i.e According to the position of reactor internals components-upper core support structure, lower core support structure, fuel assemblies and core barrel etc.,. the simplified model comprised of elements with variable mass distribution, cross section and moment of inertia etc. The springs between reactor vessel support and 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 of Yong Gwang 1&2 nuclear power plants[9] by using of STARDYNE code[10]. The response spectrum and displacement time history are shown in Fig. 7 & 8, 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 Fig. 1 is performed by using of ANSYS code under assumption that the gaps are closed. As the modal analysis results, the fundamental natural frequency is 7.1Hz, and the structural α- and β- damping values which are calculated from the obtained natural frequencies by using following equations :

$$\alpha = \frac{2(\omega_j D_j - \omega_i D_i)}{\omega_j^2 - \omega_i^2}$$

$$\beta = \frac{\omega_i \omega_j (\omega_j D_i - \omega_i D_j)}{\omega_j^2 - \omega_i^2}$$

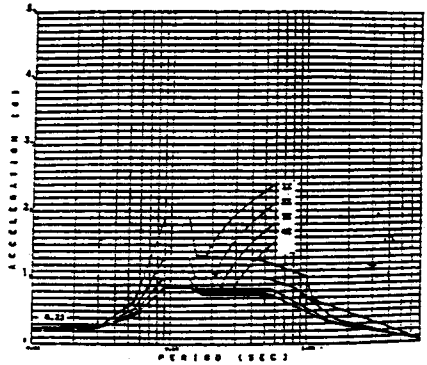


Fig. 7 Acceleration Response Spectrum

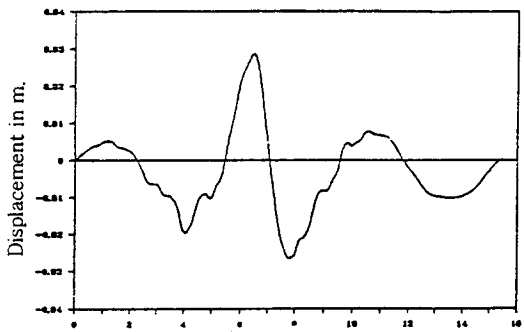


Fig. 8 Artificial Seismic Displacement Time History

where

D_i, D_j : Critical damping values of mode i and j, respectively

ω_i, ω_j : Natural frequencies of mode i and j, respectively.

The calculated α- and β- damping values are 1.863E-4 and 1.601, respectively. The governing mode shapes of the model are shown in Fig. 9.

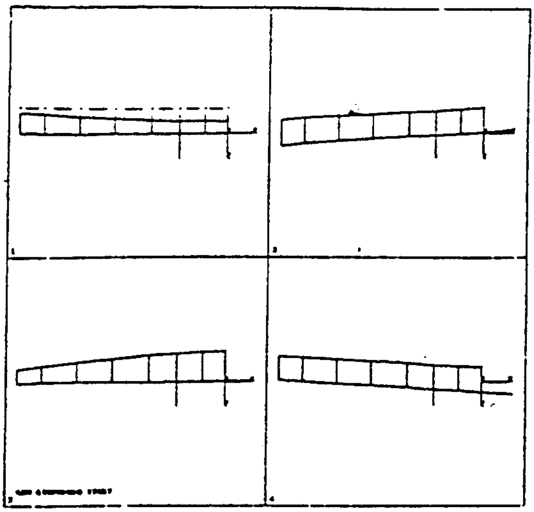


Fig. 9 Mode Shape of Reactor Internals Model

2.2 Analytical Procedure

For dynamic behavior analysis of the established reactor internals model under the given seismic load, the governing differential equations of motion are integrated numerically by way of KWUSTOSS computer code which can be used the dynamic analysis of structures of nonlinear beam-spring system in particular [11], and it is performed by the explicit Runge-Kutta-Gills algorithm in time.

The basis for this algorithm is the differential equation which is known up to time $n\Delta t$:

$$y = g(t,y)$$

Then the solution y_{n+1} is calculated for time $(n+1)\Delta t$ assuming that :

$$\Delta y_n = y_{n+1} - y_n$$

$$= 1/6[z_o + (2-\sqrt{2})z_1 + z_2 + z_3$$

, where

$$z_0 = g(n\Delta t,\ y_n)\Delta t$$

$$z_1 = g(n\Delta t + \frac{\Delta t}{2},\ y_n + \frac{Z_0}{2})\Delta t$$

$$z_2 = g(n\Delta t,\ \frac{\Delta t}{2},$$

$$y_n + (-\frac{1}{2} + \frac{1}{\sqrt{2}})z_0 + (1 - \frac{1}{\sqrt{2}})z_1\Delta t$$

$$z_3 = g(n\Delta t + \Delta t,\ y_n - \frac{1}{\sqrt{2}}z_1 + (1 + \frac{1}{\sqrt{2}})z_2)\Delta t$$

This algorithm is especially suitable for the treatment of the impact problems which could be occured because of the gaps between high frequency interference components such as reactor components. Comparing with the other standard algorithms-implicit and Stoer-Bulirsch algorthms, in the case of implicit algorithm, it is possible to calculate with a large time step than the explicit algorithm. And for the Stoer-Bulirsch algorithm, this lead to accurate results while computation time is shortened beacuse the time step size is optimized, but in case of error, the algorithm will continue trying to solve without an end. Thus the computing time by the explicit Runge-Kutta-Gills method can be achived the reasonable computing times with modal integration[12].

3. ANALYSIS RESULTS

The maximum force could be obtained at the position of core barrel upper part(between node 14 and 16) at 10.18 sec. The calculated stress in this part is 2.97 N/mm^2. The maximum displacement could be obtained at the top level of the model(node 17).

Because of the position of the excited node (node 18), the reactor internals model behaves like a pendulum and its dynamic behaviours under seismic excitation are independent on 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 occur 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 obtained at the position of the gap between core barrel upper flange and reactor vessel(the impact spring connected between nodes 15 and 16) at 9.87 sec. and is shwon in Fig. 10.

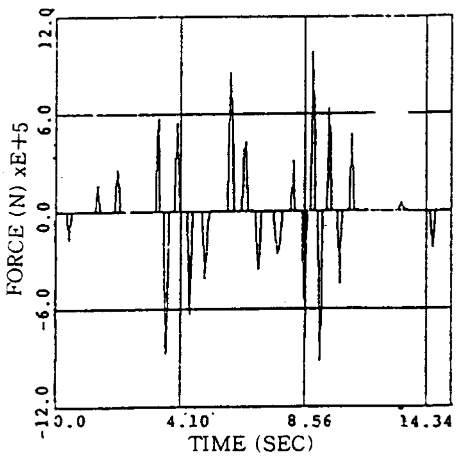


Fig. 10 Impact Forces at Coupling 15－16

The absolute displacements at upper core plate and lower core plate are obtained as shown in Fig. 11, which are to be used as an input for the analysis of the detailed fuel assembly series model. Considering the structural integrity 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 on their integrity.

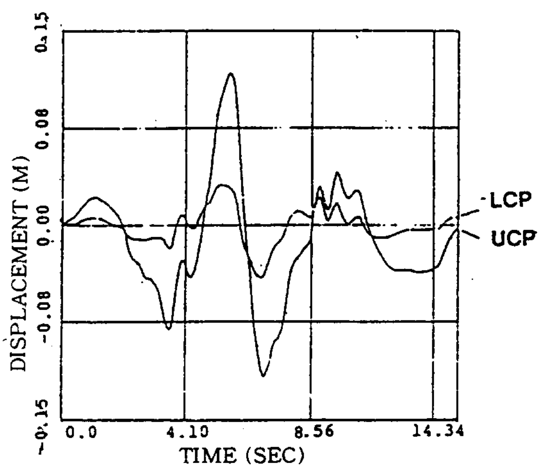


Fig. 11 Displacements of Nodes 4 & 10

4. CONCLUSION

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

－The Global Beam Model is simple for modelling but effective to analysis the reacor internals motion.

－The explicit Runge-Kutta-Gills algorithm is good for integration of the nonlinear impact equation of motion of high frequency interference components.

－The behaviours of reactor internals of Yong Gwang 1&2 900Mwe nuclear power plants during seismic event were subjected to the second and fourth modes in linear mode.

－The maximum impact on the reactor internals occurred at the core barrel upper flange position.

－Even if the absolute values of displacements of reactor internals under SSE are 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 structureal integrity.

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