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Study on the Seismic Analysis of the Reactor Vessel Internals

Myung Jo Jhung and Keun Bae Park
Korea Atomic Energy Research Institute

Won Gul Hwang
Chonnam National University

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원자로내부구조물의 지진해석에 관한 연구

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황원걸
전남대학교

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Abstract

Much effort is being done to standardize the PWR-type nuclear power plant in Korea. This paper presents the development of seismic design criteria for the reactor internals as a part of the standardization program for nuclear power plant. The seismic design loads of the reactor internals are calculated using the reference input motions of reactor vessel taken from Yong-gwang Nuclear Power Plant Units 3 and 4. An overview of analysis related to the basic parameters and methodologies is presented. Also, the response of internal components for the reactor vessel motions is carefully investigated.

요 약

최근 국내에서 가압경수로형 원자력발전소를 표준화하기 위한 작업이 이루어지고 있다. 본 논문에서는 설계표준화 작업의 일환으로서 원자력발전소 원자로내부구조물에 대한 내진설계기준을 제시하였다. 영광 3,4호기 최종설계단계에서의 운전기준지진에 대한 원자로용기 플랜지와 스너버의 거동을 입력하중으로 사용하여 지진설계하중을 계산하였고 이로부터 원자로내부구조물의 설계에 허용가능한 원자로용기의 거동을 규정하였다. 해석방법등 해석의 전반적인 개요에 대하여 설명하였고 원자로용기의 거동에 따른 원자로내부구조물 각각의 응답에 대하여 자세히 고찰하였다.

1. Introduction

The Korean Standard Safety Analysis Report has been prepared in support of the Korea Electric Power Corporation's effort to standardize nuclear power plant designs (Ref.1). It demonstrates the compliance of the Korean Standard Design with all current regulations for existing plants as well as the guidelines for future plants outlined in the US NRC's Severe Accident Policy Statement.

The starting point for the standardization program of the nuclear power plant was the Yonggwang Nuclear Power Plant Units 3 and 4 (YGN 3 and 4) design, which has two closed loops connected in parallel to the reactor vessel and the reactor core consisting of 177 fuel assemblies with its thermal power level of 2825 MWt. In developing YGN 3 and 4 design referencing in System 80 design of Combustion Engineering (C–E), modifications from earlier C–E designs were made to respond to utility needs and provide increased conservatism. While the standard design contains most of the features of YGN 3 and 4, a variety of engineering and operational improvements are included. Specifically, the Electric Power Research Institute's Advanced Light Water Reactor Requirement Document has been used as a guide for utility requirements regarding plant design.

Since the standard design is based on the assumed site–related parameters which were selected to envelope most potential nuclear power plant sites in Korea, the reactor vessel (RV) motions used for YGN 3 and 4 design are used as a reference input to calculate the seismic design loads of the reactor internals. As a result, the seismic response spectra criteria at the reactor vessel location are generated. An overview of the analysis related to the basic seismic parameters and methodologies is also presented. The response of each internal component for the vessel motions is carefully investigated.

2. Description of the Reactor Internals

The reactor internals consist of the core support barrel (CSB) assembly, the lower support structure (LSS) & incore instrumentation nozzle assembly, the core shroud, and the upper guide structure (UGS) assembly (Fig.l). The core support barrel, a right circular cylinder supported by a ring flange from a ledge on the reactor vessel, carries the entire weight of the core. The lower support structure transmits the weight of the core to the core support barrel by means of a beam structure. The core shroud surrounds the core and minimizes the amount of bypass flow. The upper guide structure provides a flow shroud for the CEAs, and limits upward motion of the fuel assemblies during pressure transients. Lateral snubbers are provided at the lower end of the core support barrel assembly.

The principal design bases for the reactor internals are to provide the core with vertical supports and horizontal restraints during all nommal operating, upset, and faulted conditions.

The core should be supported and restrained during normal operation and postulated accidents to ensure that core will be held in place and adequate coolant path is preserved for heat removal.

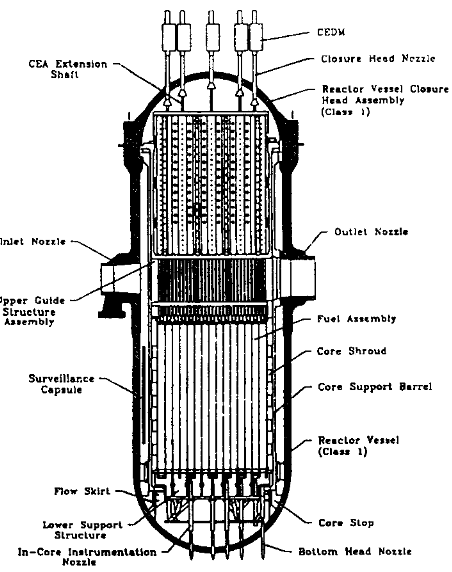


Fig. 1. Schematic Diagram of the Reactor Internals

3. Structural Model

The mathematical model of the reactor internals consists of lumped masses and elastic beam elements to represent the beam-like behavior of the internals, and nonlinear elements to simulate the effects of gaps between components (Fig.2). Typical component gaps represented by nonlinear elements are the core support barrel/reactor vessel snubber gap and core shroud guide lug gap. The gap between the core shroud and core support barrel or between the core support plate and core support barrel is sufficiently large and therefore no interference occurs.

At appropriate locations within the internals and core, nodes are chosen to lump the weights of the structure. The criterion for choosing the number and location of mass points is to provide for accurate representation of the modes of vibration for each of the internal components. For the beam element connecting two nodes, properties are calculated for moments of inertia, cross-sectional areas, effective shear areas, stiffnesses and length. Stiffnesses for the complex internal structures such as UGS and CSB flanges, CSB snubber, hold-down ring and CEA guide tubes are determined by finite element analyses.

The effects of fluid/structure interaction between internal components due to their immersion

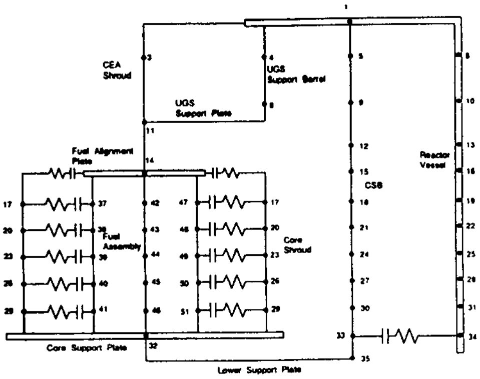


Fig. 2. Lumped Mass Model of Reactor Internals

in a confining fluid are considered (Ref.2). The hydrodynamic mass matrix is applied to the analytical model representing the reactor internal structures in the horizontal direction. Fluid/structure interaction is characterized by the full hydrodynamic mass matrix including the off-diagonal hydrodynamic coupling terms which will affect significantly the dynamic characteristic of the solid structure with narrow gap or annulus.

Fluid/structure interaction for the remaining components is characterized by hydrodynamic added masses. The diagonal hydrodynamic added mass matrix components consist of two terms, i.e., contained water mass and displaced water mass. The contained water mass representing the effect of internal liquid on the structure is the mass of the internal liquid carried by the structure as it responds horizontally in the beam bending mode. The displaced water mass represents the effect of surrounding liquid on the structure. It is due to the force required to move the structure through the surrounding liquid. The displaced water mass is equal to the mass of liquid displaced by the structure assuming it to be solid.

The fluid/structure interaction of the core region is determined experimentally. The complex configuration of fuel assemblies, closed spaced fuel rod arrays, and fuel assembly internal hysteresis and friction preclude a formal evaluation of the hydrodynamic mass matrix. Test data is used to obtain the net effect of fluid/structure interaction on the fuel assemblies' natural frequencies. The fuel assembly hydrodynamic added mass is then evaluated to yield natural frequencies consistent with test data.

4. Seismic Analysis

The design basis earthquake has maximum free field horizontal ground accelerations at the foundation level of 0.20g for the safe shutdown earthquake (SSE) and 0.10g for operating basis earthquake (OBE). The maximum vertical ground accelerations at the foundation level are 0.13g for the SSE and 0.067g for the OBE. The site design response spectra for these maximum acceleration values are defined by Ref.3.

The forcing function to the model consists of acceleration time histories at the RV flange and snubber elevations determined from the reactor coolant system analysis. RV is so stiff comparing with internal components that its local effect is negligible. Therefore, only translational accelerations on the RV between the flange and snubbers are computed by linear interpolation and are input into the model. These translational accelerations along the vessel are required for the calculation of hydrodynamic forces between CSB and RV annulus. The acceleration time histories of RV flange and snubber which were generated from the reactor coolant system analysis and the corresponding response spectra are shown in Fig.3.

The response of the internals is computed by the SHOCK code (Ref.4), which solves for the response of the structures represented by lumped mass and spring systems under a variety of loadings. This is done by numerically solving the differential equations of motion for an N degree of

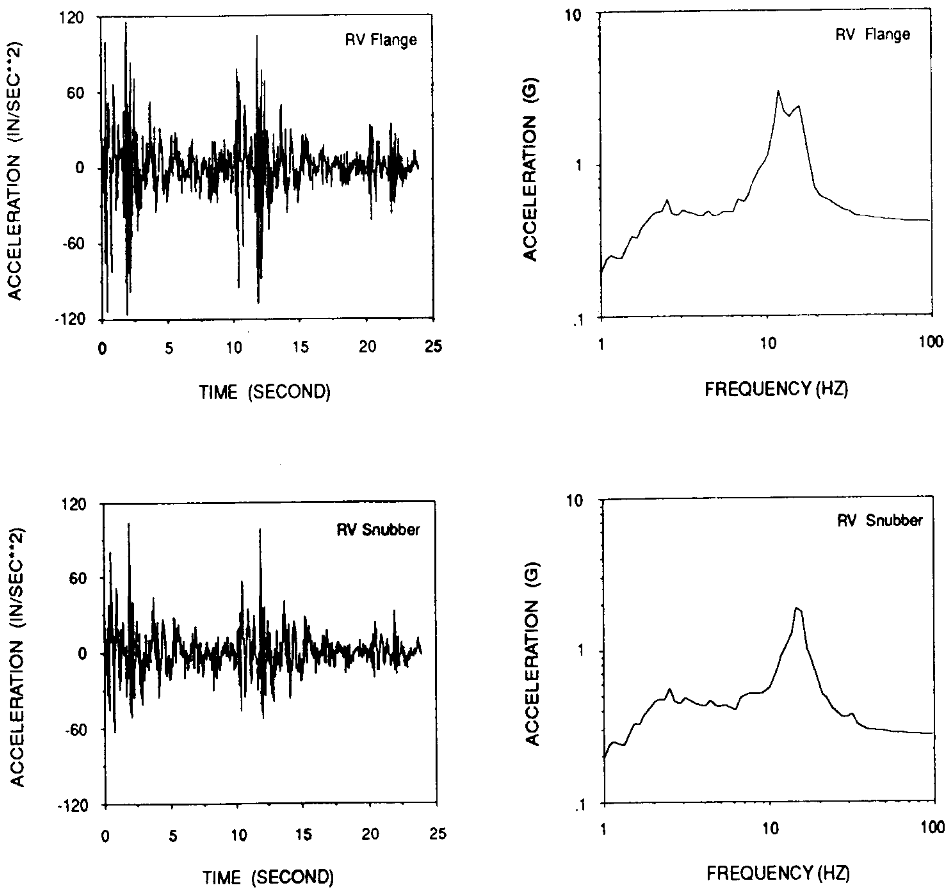


Fig. 3. Acceleration Time Histories of RV Flange and Snubber and the Corresponding Response Spectra (2% Damping) for OBE

freedom system using the Runge–Kutta–Gill technique. The equation of motion can represent an axially responding system or a horizontally responding system i.e., an axial motion or a coupled horizontal and rotational motion. The code is designed to handle a large number of options for describing load environments and includes such transient conditions as time–dependent forces and moments, initial displacements and rotations, and initial velocities. Options are also available for describing steady–state loads, preloads, accelerations, gaps, nonlinear elements, hydrodynamic mass, viscous damping, friction, and hysteresis.

The results of analysis consist of minimum and maximum values of shears and moments of each component which will be used for design loads, and motions for fuel alignment plate and core support plate which will be used for the detailed core analysis.

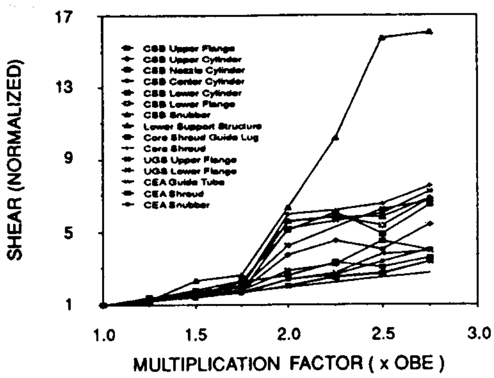
5. Results and Discussion

The design loads of each component are generated for the designer to determine the initial sizing of the internals (Table 1). Fig.4 shows the response ratio based on the OBE excitation, where a big jump is noticed at the multiplication factor of 1.75. This means beyond that factor the CSB snubber contacts reactor vessel frequently and its non–linear characteristics give out high response loads. The non–linear characteristics of responses are severer in the CSB lower region and also in LSS. But the responses of CEA shroud, UGS assembly and CEA guide tube show less non–linearity. This is caused by the fact that the load path is from RV through CSB snubber and CSB lower to CSB upper flange and to LSS and core shroud, and that the load decreases in intensity as it moves to another point.

The relative displacement time histories of CSB snubber to RV are shown in Fig.5. The CSB snubber gap, which is normally 0.02 inches, is closed

Table 1. Maximum Loads of Internals for OBE

COMPONENT	SHEAR (1b×E5)	MOMENT (in-1bs×E6)
CSB Upper Flange	3.90	32.3
CSB Upper Cylinder	3.90	25.0
CSB Nozzle Cylinder	1.30	19.3
CSB Center Cylinder	1.30	8.2
CSB Lower Cylinder	1.20	8.2
CSB Lower Flange	1.10	9.1
CSB Snubber	0.68	–
LSS	0.98	9.1
CS Guide Lug	0.51	–
Core Shroud	0.84	8.2
UGS Upper Flange	3.90	26.5
UGS Lower Flange	0.58	1.9
CEA Guide Tube	0.25	1.6
CEA Shroud Assembly	0.36	1.5
CEA Snubber	0.24	–



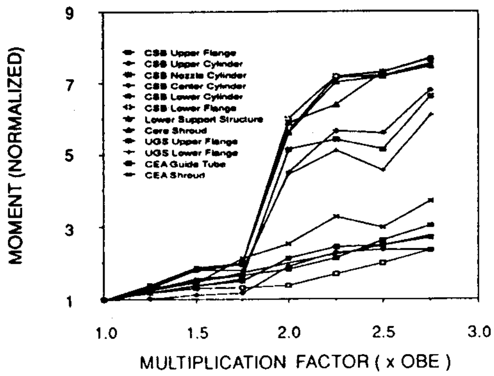


Fig. 4. Response Ratio of Shear and Moment for OBE Loads

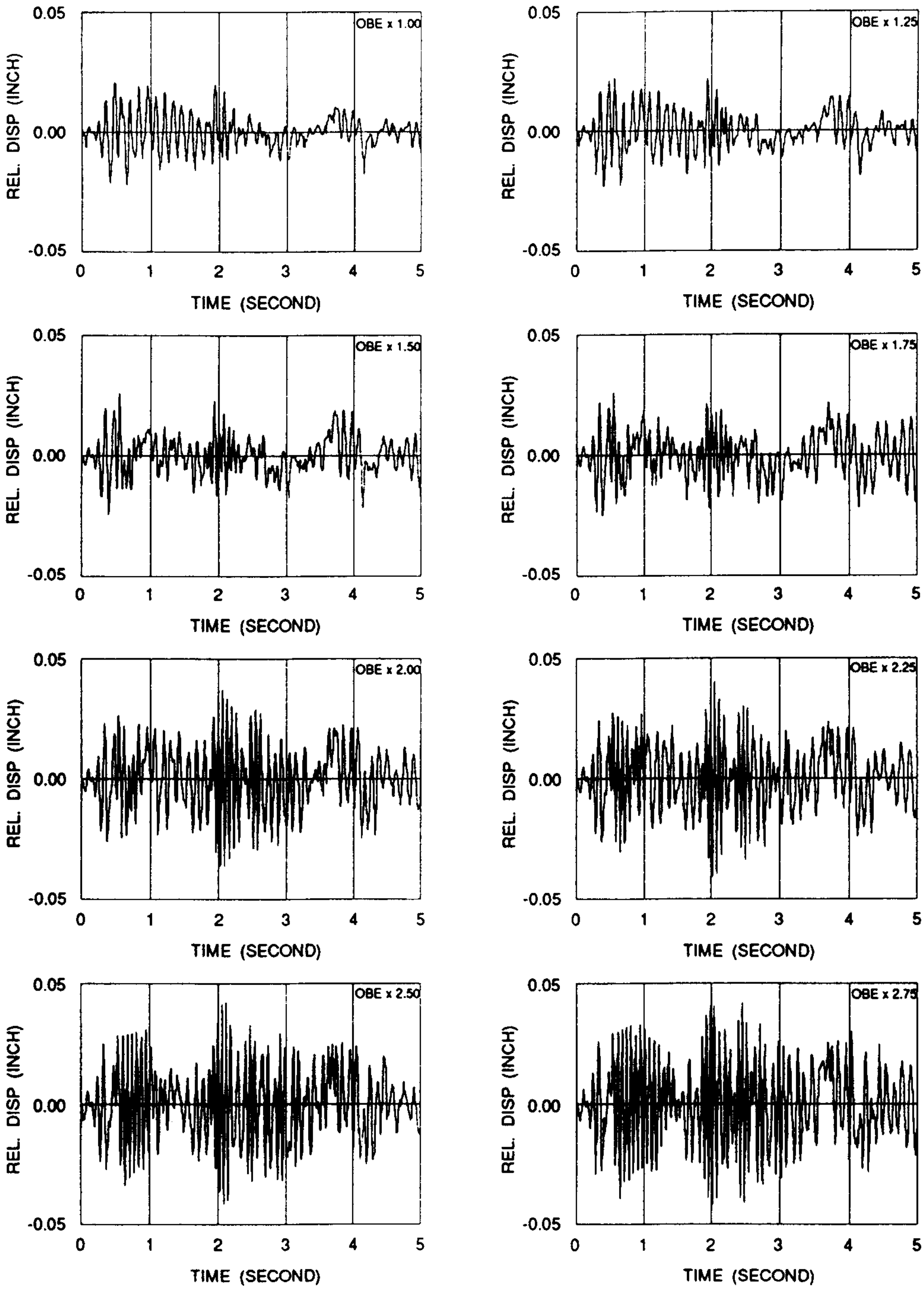


Fig. 5. Relative Displacement Time Histories of CSB Snubber to RV

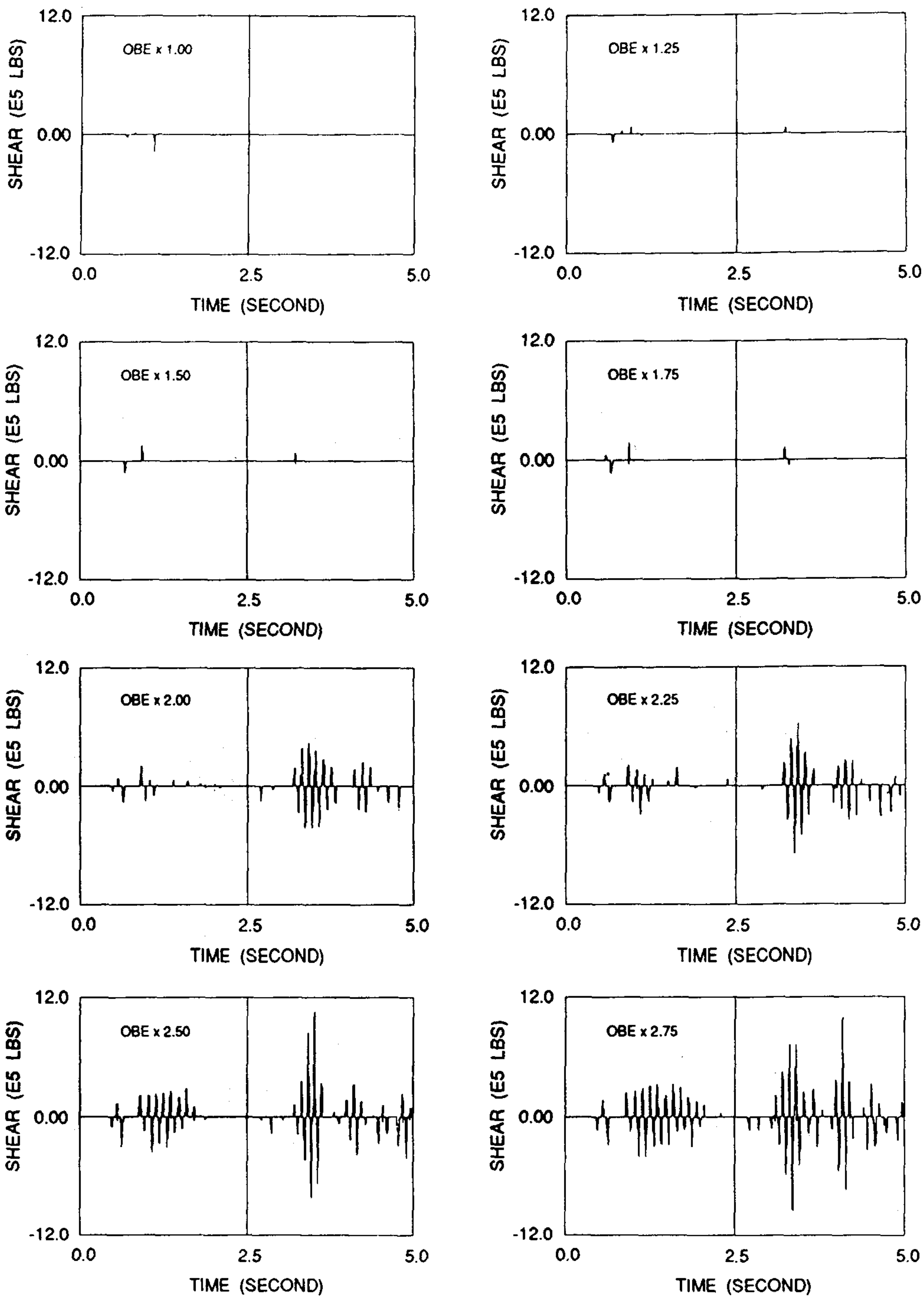


Fig. 6. Shear Force Time Histories of CSB Snubber

even for the OBE excitation, and therefore the impact occurs. The gap closed more frequently as the multiplication factor becomes bigger. This can be explained by Fig.5, where the frequency is about 8 and 12 cycles per second for OBE×1.00 and OBE×2.75, respectively. Fig.6, the shear force time histories of CSB snubber, also indicates that impact pulses for the first 3 seconds are 3~6 and 36~58 for OBE×1.0~OBE×1.75 and OBE ×2.00~OBE×2.75, respectively.

The loads which do not exceed the allowable limit were quantitatively determined in the design of YGN 3 and 4. These values were used to determine the acceptable multiplication factor in Fig.4 and the response spectra at that point are chosen as the upper bound of the acceptable limit. The acceptable criteria of response spectra for reactor internals design were generated as shown in Fig.7, which were broadened by±15% and smoothened according to Reg. Guide 1.122 (Ref.5).

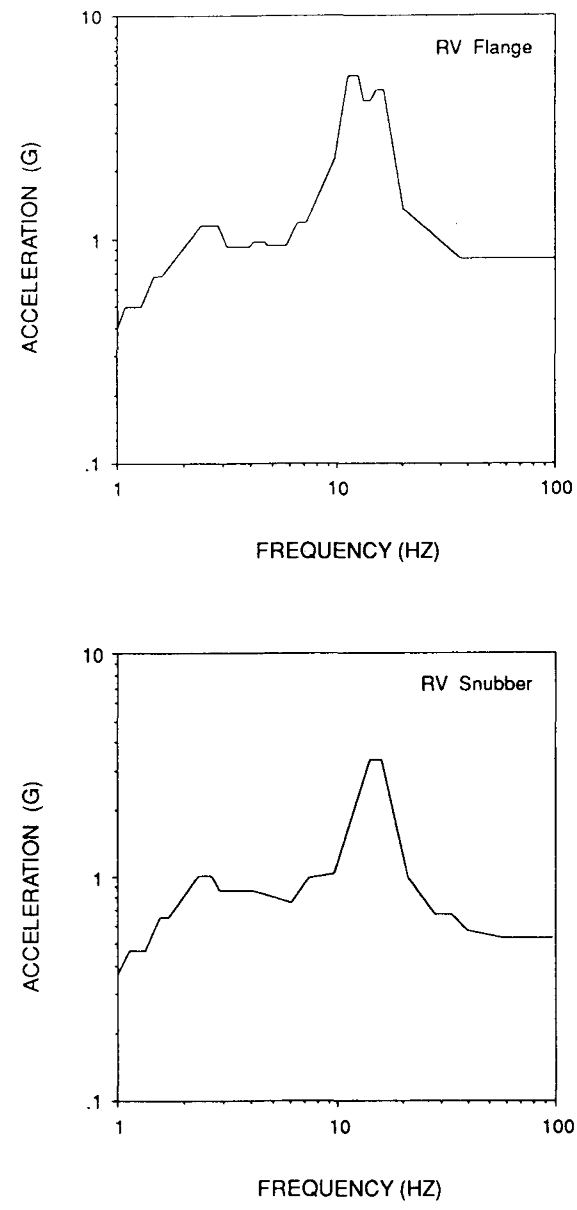


Fig. 7. Response Spectra Criteria of RV Flange and Snubber for 2% Damping

6. Conclusions

The seismic design criteria of the reactor internals were suggested for the future nuclear power plant. The acceptance criteria aim at predicting the seismic responses of reactor internals and minimizing additional analysis tasks to generate design loads. The acceptance criteria were provided as a form of response spectra at the reactor vessel location. Also, the resulting responses showed that the CSB lower part and LSS are more sensitive to the RV motions than CEA shroud and UGS assembly.

References

1. "K-SSAR : Korean Standard Nuclear Power Plant Safety Analysis Report," Korea Electric Power Corporation (1991)
2. M.J. Jhung, "The Effect of Fluid/Structure Interaction on the Reactor Internals Responses for Seismic and Pipe Break Excitations," SDM92-RM-002, KAERI (May 1992)
3. USNRC Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear

Power Plants" (1973)

4. V.K. Gabrielson, "SHOCK, A Computer Code for Solving Lumped-Mass Dynamic Systems," SCL-DR-65-34 (January 1966)
5. USNRC Regulatory Guide 1.122, "Development of Floor Response Spectra for Seismic Design of Floor Supported Equipment or Components," Revision 1 (February 1978)