지진 및 배관파단에 대한 핵연료집합체의 동적 검증

Dynamic Qualification of Fuel Assembly for Earthquake and Pipe Break

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국문요약

핵연료집합체 검증 프로그램의 일환으로 본 연구에서는 지진과 배관파단이 핵연료집합체의 건전성에 미치는 영향을 검토하였다. 원자로 노심의 상세 동적해석을 이용하여 지진 및 배관파단시 핵연료집합체에 발생하는 전단력, 굽힘 모우멘트 및 변위를 계산하였고 또한 집합체를 지지하고 있는 지지격자체의 충격력을 검토하였다. 이들 하중에 대한 핵연료집합체의 응력해석을 수행하여 사고조건하에서의 구조적 건전성에 대하여 언급하였고 추후 설계시 고려할 사항을 제시하였다.

주요어 : 원자로 노심, 핵연료집합체, 지지격자체, 배관파단, 지진

ABSTRACT

As a part of the fuel assembly qualification program, the effect of earthquake and pipe break excitations on the fuel assembly integrity is investigated. Using the detailed dynamic analysis of a reactor core, peak responses for the motions induced from earthquake and pipe break excitations are obtained. The dynamic responses such as fuel assembly shear force, bending moment and displacement, and spacer grid impact loads are carefully investigated. Also, the stress analysis is performed and the fuel assembly structural integrity under the faulted condition is addressed.

Key words : *reactor core, fuel assembly, spacer grid, pipe break, earthquake*

1. Introduction

The reactor core of a pressurized water reactor is composed of several hundreds of assemblies of different kinds such as ordinary fuel assemblies and control element assemblies. They are rectangular beams supported by a fuel alignment plate (FAP) and a core support plate (CSP) at the top and bottom ends, respectively, immersed in coolant with very narrow spacings between adjacent assemblies. Thus, in an earthquake and/or a pipe break event, their vibratory motions as a whole cluster may have a complicated nature including non-linearity due to the effect of collisions between assemblies and dynamic interactions through fluid coupling forces.

Safety qualification of the reactor core is one of the crucial issues in the design of a pressurized water reactor, and it should be secured that the structural integrity of the fuel assemblies and the control rod insertion capabilities be maintained against the design loads.

For the purpose of assessing safety of the reactor core, many hypothetical accidents are postulated and the dynamic response analyses to these accidents are performed. The structural integrity of a fuel assembly should be assured

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for the worst loading condition which is called a faulted condition. Under this condition, pipe break and safe shutdown earthquake responses are combined.[1]

In the present study, peak responses are obtained for the motions induced from pipe break and earthquake excitations using the method for the detailed dynamic analysis of a reactor core.[2] Also, the stress analysis is performed and the effect of the pipe break and earthquake excitations on the fuel assembly structural integrity under the faulted condition is addressed.

2. Reactor Internals and Fuel Assembly

The reactor internals are designed to support the reactor core, maintain the core in a coolable array, and guide the control element assemblies (CEAs) into the top of the core, and to constrain and protect the CEAs from coolant flow.

The components of the reactor internals are divided into two major parts consisting of the core support barrel (CSB) assembly and the upper guide structure (UGS) assembly. The flow skirt, although functioning as an integral part of the coolant flow path, is separate from the internals and is affixed to the bottom head of the reactor vessel. The arrangement of these components is shown in Fig. 1.

The core support barrel assembly includes the core support barrel, the lower support structure and incore instrumentation nozzle assembly, and the core shroud. The CSB is a right circular cylinder supported by a ring flange from a ledge on the reactor vessel. It 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 grid beam structure. The core shroud surrounds the core and minimizes the amount of bypass flow. The upper guide structure assembly includes the UGS barrel assembly, the CEA shroud assembly, the top hat, the holddown ring, and the heated junction thermocouple shroud assembly. The UGS assembly provides the CEAs a protection from coolant flow, and limits upward motion of the fuel assemblies.

The typical PWR core of 2825 MWt is composed of 177 fuel assemblies and 73 or more control element assemblies. The fuel assemblies are arranged to approximate a right circular cylinder with an equivalent diameter of 123 inches and an active length of 150 inches. (Fig. 2) The fuel assembly, which provides for 236 fuel rod positions (16×16 array), includes 5 guide tubes welded to 11 spacer grids and is closed at the top and bottom by end fittings. (Fig. 3) The guide tubes each displace four fuel rod positions and provide channels which

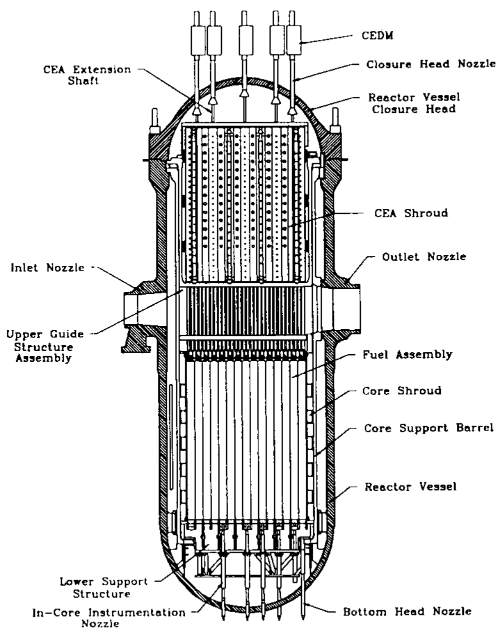


Fig. 1 Arrangement of reactor internals

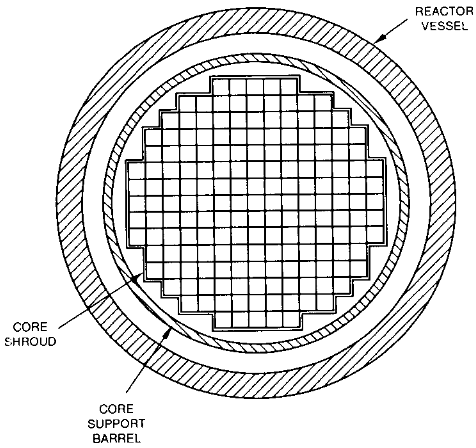


Fig. 2 Reactor core arrangement

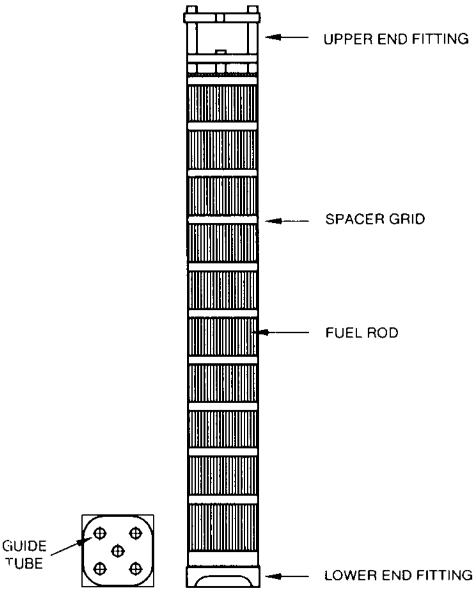


Fig. 3 Schematic diagram of fuel assembly

guide the control element assemblies over their entire length of travel. In-core instrumentation is installed in the central guide tube of selected fuel assemblies. The incore instrumentation is routed into the bottom of the fuel assemblies through the bottom head of the reactor vessel. The outer guide tubes, spacer grids and end fittings form the structural frame of the assembly.

The fuel spacer grids maintain the fuel rod array by providing positive lateral restraint to the fuel rod but only frictional restraint to axial fuel rod motion. The grids are fabricated from preformed Zircaloy or Inconel strips (the bottom spacer grid material is Inconel) interlocked in an egg crate fashion and welded together. Each cell of the spacer grid contains two leaf springs and four arches. The leaf springs press the rod against the arches to restrict relative motion between the grids and the fuel rods.

3. Analysis

3.1 Model Development

3.1.1 Coupled Reactor Internals and Core Model

The mathematical model of the 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. Typical component gaps represented by nonlinear elements are the core support barrel, pressure vessel snubber gap and core shroud guide lug gap. The gaps between the core shroud and core support barrel or the core support plate and core support barrel are sufficiently large that no contacting occurs. However, for every analysis performed, this assumption is verified by confirming that the relative deflections of component are in fact smaller than existing gaps.

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 dynamically significant 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. Unit deflections and rotations are applied and the resulting reaction forces are calculated. These results are then used to derive the equivalent member properties for the structures. The CSB upper region is modeled to account for the possible interactions between the CSB upper flange, UGS upper flange, hold-down ring and the RV ledge using the nonlinear, hysteresis and friction elements. But if justified by analysis, it can be modeled as one mass point. A dynamically equivalent representation of the CEA shroud is included in the model. This representation is based on a frequency analysis of the detailed finite element model.[(3),(4)]

A typical coupled internals and core model is shown in Fig. 4. The actual arrangement and detail in the model may vary with the function of plant design, and the magnitude and nature of the excitation.

3.1.2 Detailed Core Model

In the core model, the fuel assemblies are modeled as uniform beams. Lumped masses are included at spacer grid locations to represent the significant modes of vibration of the fuel and to account for possible spacer grid impacting. Nonlinear spring couplings are used to simulate the gaps in the core. Each spacer grid is characterized by the dual load path model which represents the load paths

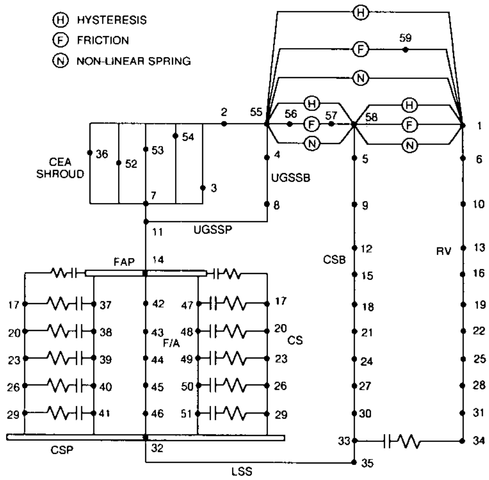


Fig. 4 Lumped mass model of reactor internals and core

associated with both one-sided and through-grid impacts. One-sided loads are the loads experienced by one side of a grid when it impacts on another grid or the core shroud. Through-grid loads are the loads developed through grid loadings on a spacer grid.

The fuel analytical model was constructed by calculating nodal properties for corresponding locations based on the weight distribution data. The dynamic characteristics of the fuel bundle including natural frequency and damping were also determined from the test data. The analytical model of the fuel bundle was modified to include dynamic effects by adjusting the bundle stiffness to obtain the proper natural frequency and prescribing the damping as a percentage of critical damping.

Hydrodynamic (diagonal coupling coefficients) mass was added to the structural mass to obtain the proper natural frequency in water. The off-diagonal coupling terms are not considered in the core model, that is, hydraulic coupling between the fuel assemblies is neglected. This was justified by water loop tests,[(5)] which indicate that the natural frequency drop can be accounted for by added masses corresponding to the displaced liquid, meaning that a fuel assembly in a channel does not behave in a significantly different manner as a fuel assembly in an infinite fluid. Physically this means that without a wrapper tube, the fluid can flow from one side of the assembly to the other, across the fuel assembly rather than around it.

The spacer grid model was developed considering impacting of adjacent fuel assemblies or peripheral assemblies and the core shroud. If two fuel assemblies hit another or if one assembly strikes the core shroud, then the spacer grids are loaded on only one force. This type of impact has been called a one-sided impact. The second impact type is called a through-grid impact because the impact force is applied simultaneously to opposite faces of the spacer grid. For example, a through-grid impact occurs when one fuel assembly is lying against the core shroud and a second assembly hits it.[6] Therefore, the spacer grid model separates out through-grid and one-sided load paths.(Fig. 5) The pluck vibration, pluck impact, spacer grid compression, and spacer grid section drop tests provide data used in determining the spacer grid impacting parameters. The eigenvalue analysis was performed to calculate the frequencies and mode shapes of the fuel assembly, which are compared to the test results to verify the model. Core model of fifteen row fuel assemblies is shown in Fig. 6.

3.2 Excitations

Structures and equipment in a nuclear power plant are required to be designed or qualified to resist the combined effects of a large number of loads including static loads (e.g., dead weight, pressure loads and temperature loads) and multiple dynamic loads. The dynamic loads are either transmitted directly to the entire primary structure of a nuclear power plant in the form of vibratory loads or they may be generated within the primary structures due to plant conditions. The dynamic loads which are considered in the design/evaluation include those from natural phenomena like earthquakes, and from plant conditions which are either postulated to occur, for example, the pipe break loads, or those that trigger automatically to prevent accidents within the plant.

The seismic design loads may be computed based on either an actual earthquake record calibrated for a given plant site, or an artificial earthquake computable with USNRC regulatory guide 1.60[7] spectra. A site-specific or natural earthquake is generally narrow frequency band earthquake as compared to one generated

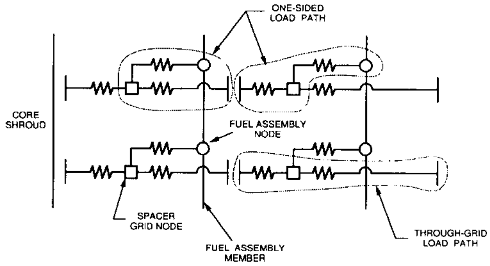


Fig. 5 Dual load path impact model of spacer grid

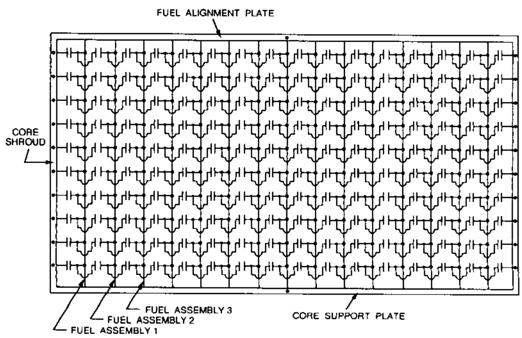


Fig. 6 Core model of 15 row fuel assemblies

from a regulatory guide 1.60 spectra. The frequency band of the earthquake depends on the plant site. For the Ulchin nuclear power plant units 3 and 4 (UCN 3 and 4) in Korea, 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 the 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. Based on this information, the architect engineer generated base motions and one of building analysis results is support motions of the reactor coolant system. Reactor vessel motions are obtained from the reactor coolant system analysis in which a very simplified model of the internals and core is used. They are used as input motions to the coupled model of the reactor vessel internals and core.

The pipe break loads which are significant to the design of nuclear power plant structures and equipments are produced by a postulated design basis break. In the recent design of nuclear power plants, main coolant loop double ended guillotine breaks are eliminated from the design basis because of leak-before-break (LBB) concept.[8] Instead, branch line pipe breaks are considered as one of the Level D service loadings. Of the branch line pipe breaks postulated, LBB evaluation is performed for piping systems with a diameter of 0.254m (10inches) or over and it is anticipated that pipe breaks with a diameter of 0.254m or over be no more considered as design basis. Once elimination of those piping systems from the design consideration is accepted based upon current LBB evaluation, only the 0.072m (3inch) pressurizer spray line nozzle break in the primary side remains in the design basis, but its responses are expected to be lower than those of the secondary side pipe breaks which are requested for the faulted condition.[9] In this paper, therefore, the responses are calculated for the secondary side pipe breaks such as main steam line (MSL) and economizer feedwater line (EFW).

3.3 Dynamic Response Calculation

The procedure for detailed horizontal core analysis is described briefly in the following. As the first step, reactor vessel motion is obtained from the reactor coolant system analysis in which a very simplified model of the reactor vessel internals and core is used. Subsequently, the reactor vessel motion is used as an input to a coupled model of the reactor vessel internals and core. In this model, only a lumped model of the core is used with a primary purpose to include interaction effects between the response of the fuel assemblies, core plates and core shroud .[10] Core plates and core shroud motions from the coupled internals and core analysis are input to the detailed core model in which each fuel assembly is modeled individually. In the last step, the deflected shapes of the fuel assembly and spacer grid impact loads from the detailed core analysis are used to assure the structural integrity. Fig. 7 illustrates the overall fuel assembly design analysis flow.

The detailed horizontal core model is developed for the time history analysis for the pipe break and earthquake excitations, and dynamic response is determined using the core plate motions from the coupled internals and core analysis. The vertical response is obtained in the coupled internals and core model and therefore separate analysis is not

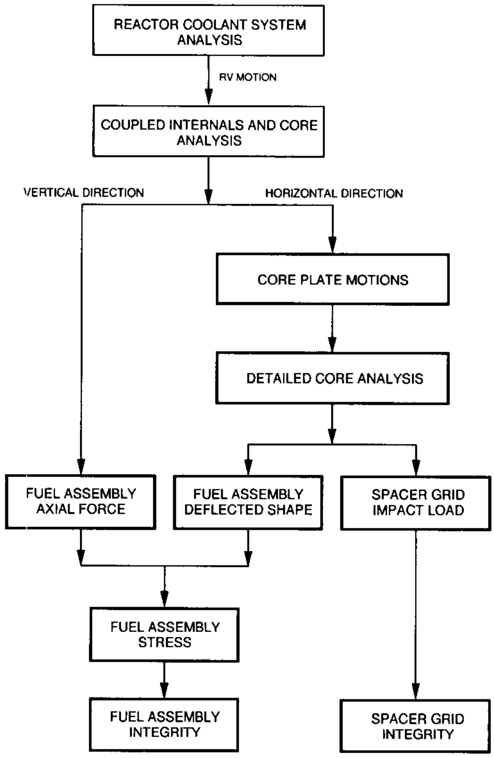


Fig. 7 Analysis procedure for fuel assembly design

required.[11]

The input excitations to the detailed core model consist of the translational and angular motions of the core plates and the translational motion of the core shroud. The core shroud is so stiff comparing with fuel assembly that its local effect is negligible. Therefore, only the translational component of the core shroud is used. The input motions are obtained from the analysis of a coupled internals and core model which has a much less detailed representation of the core.

The reactor vessel motions for UCN 3 and 4 in Korea as shown in Fig. 8 are used to excite the coupled internals and core model. The analysis of the coupled internals and core model generates the core plate motions which are shown in Fig. 9. The responses of the fuel assemblies to the excitations were obtained using the integration of equations of motion by the Runge-Kutta-Gill method for first-order differential equations.[2]

3.4 Stress Calculation

The fuel assembly axial and lateral internal support consists of five guide tubes (primary structural members) supported by spacer grids welded to the guide tubes. Because this configuration is complex it requires consideration of the distribution of shear forces and bending moments within the bundle in order to calculate guide tube and fuel rod stresses for a given loaded condition. The stress analysis is performed based on the premise that for lateral fuel assembly deflections and the resultant material strains, there is a direct correspondence between the deflected shape of the assembly and the strains in the assembly structure.

Consider the general configuration which an element assumes under an arbitrary state of load (P), linear displacement (v) is allowed between the ends as well as independent angular rotation of each end (θ_1, θ_2). Guide tube or fuel rod may be treated as simple circular cross section beams sharing a common deflected shape within a given element. The differential equation governing flexure of one guide tube or fuel rod is[12]

$$\frac{d^3 v}{dx^3} = \frac{-p}{EI} \qquad (1)$$

where v is the displacement of any point on the guide tube, x the distance along the guide tube, E the elastic modulus, I the section moment of inertia and p the load per guide tube or fuel rod. The appropriate boundary conditions are

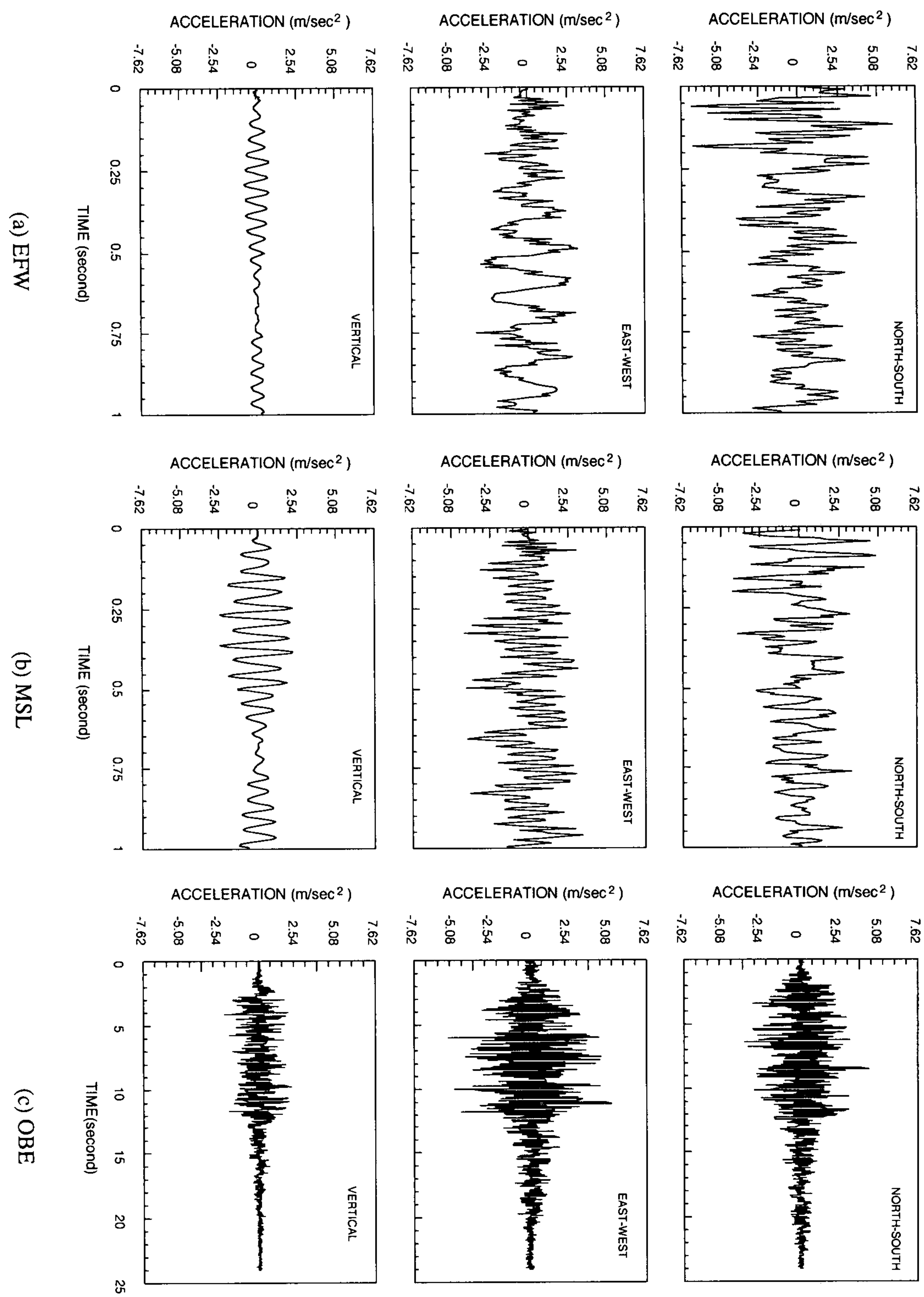


Fig. 8 Acceleration time histories of RV flange

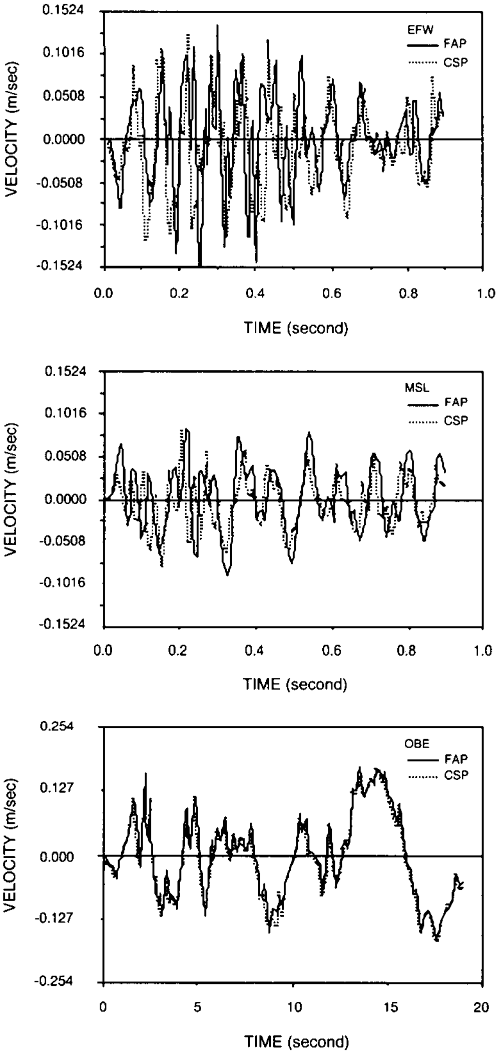


Fig. 9 Velocity time histories of FAP and CSP

$$v(0)=v_1$$
$$\frac{dv}{dx}(0)=\tan\theta_1 \tag{2}$$
$$\frac{dv}{dx}(L)=\tan\theta_1$$

where θ_1 and θ_2 are shown in Fig. 10.

Solving equation (1) for boundary conditions (2) gives the following solution

$$v=\frac{-p}{6EI}x^3+\left(\frac{\tan\theta_2-\tan\theta_1}{2L}+\frac{pL}{4EI}\right)x^2 + \tan\theta_1 x+v_1 \tag{3}$$

$$\frac{dv}{dx}=\frac{-p}{2EI}x^2+\left(\frac{\tan\theta_2-\tan\theta_1}{L}+\frac{pL}{2EI}\right)x + \tan\theta_1 \tag{4}$$

$$\frac{d^2v}{dx^2}=\frac{M}{EI}=\frac{-p}{EI}x + \left(\frac{\tan\theta_2-\tan\theta_1}{L}+\frac{pL}{2EI}\right) \tag{5}$$

For the case with given θ_1, θ_2 and $v(L)=v_2$, equation (3) becomes for small θ

$$v_2-v_1=\frac{pL^3}{12EL}+\left(\frac{\theta_2+\theta_1}{2}\right)L \tag{6}$$

From this equation the net load applied to one guide tube or fuel rod is determined as

$$p=\frac{12EI}{L^3}\left[v_2-v_1-\frac{(\theta_2+\theta_1)L}{2}\right] \tag{7}$$

and equation (5) becomes for small θ

$$M=\frac{p}{2}(L-2x)+\frac{\theta_2-\theta_1}{L}EI \tag{8}$$

Therefore, the moment at $x=L$ is calculated as

$$M=\frac{2EI}{L}\left[(2\theta_2+\theta_1)-\frac{3}{L}(v_2-v_1)\right] \tag{9}$$

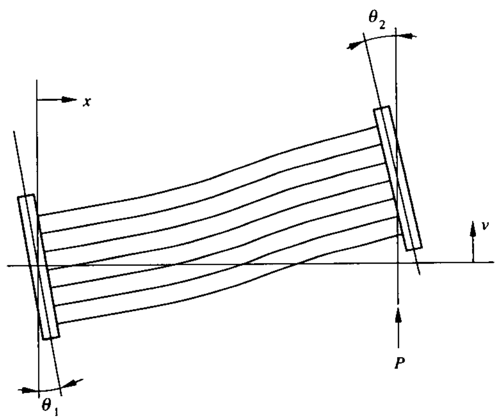


Fig. 10 Typical element under load

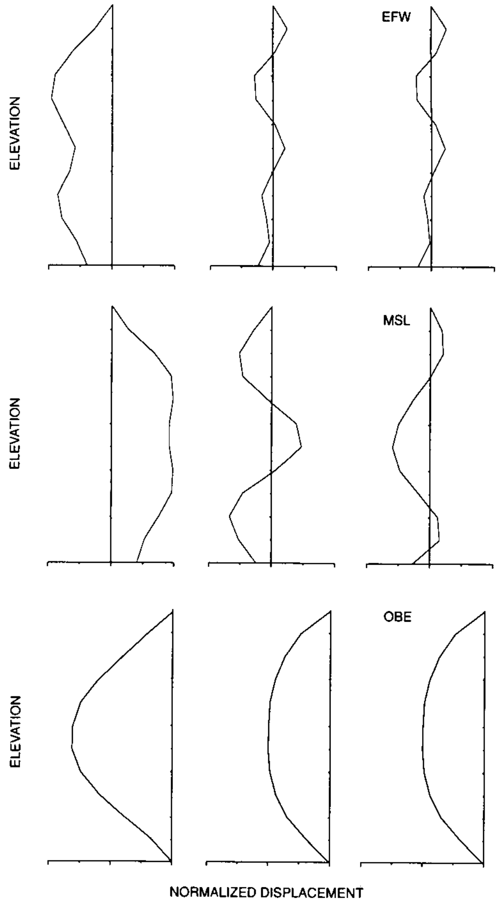
From this moment, the bending stress can be computed.

The total stress intensity due to dynamic loads will have stress contributions from the bending loads caused by overall deformation and axial loads induced by frictional restraint at the spacer grids.

4. Results and Discussion

The result of the detailed core analysis consists of peak spacer grid impact loads, fuel assembly moments, shears and deflected shapes. The impact loads are used to evaluate the structural integrity of spacer grids. The deflected shapes which correspond to peak loading conditions - peak displacement, peak shear and peak moment - are used to calculate stresses using a detailed static model of the fuel assembly. The deflected shapes for pipe break excitation (Fig. 11) indicate that high modes (i.e., over 3rd mode) contribute to the peak loads comparing with the first mode natural frequencies for the earthquake case.

The spacer grid impact loads and the fuel assembly responses are shown in Table 1. No spacer grid impact loads exist for the pipe



(a) peak displacement (b) peak moment (c) peak shear

Fig. 11 Deflected shapes of fuel assembly for peak loading conditions

Table 1 Responses of fuel assembly

Components	MSL	EFW	OBE	SSE	Allowable
Spacer grid					
One-sided impact (N)	0	0	8607	17019	19630
Through-grid impact (N)	0	0	5796	12188	15106
Fuel assembly					
Deflection (mm)	2.464	2.489	30.734	40.538	
Shear (N)	6588	4386	885	1753	
Moment (N-m)	11	7	443	854	
Axial force (N)[11]					
@ Fuel rod	472	374	1239	2252	
@ Guide tube	40	31	111	200	

break excitations which means that the collisions between fuel assemblies, and fuel assembly and core shroud do not occur. This is explained by the fact that the longest (15) and the shortest (5) row models give the same results. In an earthquake case, the longest row model generates the highest response. Therefore only the impact loads due to earthquake excitation will be used to evaluate spacer grid integrity. The square root of the sum of the squares (SRSS) of one-sided impact are 8607N and 17019N for OBE and SSE, respectively. The OBE impact is almost half of SSE impact. For the through-grid impacts, the SRSS values are 5796N and 12188N for OBE and SSE, respectively. The ratio of OBE to SSE is 48%.

For the axial response of fuel assembly, the axial force of fuel rods is 1239N and 2252N for OBE and SSE, respectively. The response ratio of OBE/SSE ranges in 48% to 55% for the fuel rods, end fittings and guide tubes. For the pipe break excitations, the axial force of fuel rods is 472N and 374N for MSL and EFW breaks, respectively.

The required data for stress calculation of the fuel assembly is a deflected shape which represents the severest conditions. Stresses are calculated using the equation (9) and the guide tube and the fuel rod stress intensities are summarized in Table 2. For the fuel rod stress calculation the differential pressure which always exists during a plant operation is included. Most of the stress intensity for fuel rod came from the differential pressure load and the contribution of the pipe break and earthquake excitations is almost negligible. The resulting stress intensity in each direction is combined by the SRSS method to give the total stress intensity and also the stress intensities from other loading conditions are considered to be compared with code allowables to verify the structural integrity.

5. Conclusions

The dynamic responses such as fuel assembly shear force, bending moment and displacement, and spacer grid impact loads are obtained for the motions induced from pipe break and earthquake excitations using the detailed dynamic analysis of a reactor core. The stress analysis is performed using the deflected shape of a fuel assembly and the stress intensities for pipe break excitations are compared with those of earthquake. The stress comparisons between pipe break and earthquake show that pipe break contributions are

Table 2 Stress intensities of fuel assembly

Components	Pipe Break	OBE	SSE	Allowable[1]
Fuel rod (Mpa)[2]				
Primary membrane + bending	148.9	148.9	177.9	219.3
Primary membrane only	148.9	148.9	148.9	149.6
Guide tube (Mpa)				
Primary membrane + bending	2.489	164.1	242.0	426.8
Primary membrane only	4.8	37.9	52.4	308.2

1) Allowables are for the faulted condition where pipe break and SSE responses are combined
2) Inculdes stress components due to fuel rod differential pressure

negligible for the faulted condition when leak-before-break concept is applied to the primary side piping systems with a diameter of 0.254m (10inches) or over. It is, therefore, concluded that the secondary side pipe breaks may be no longer considered in the mechanical design of a fuel assembly for the future nuclear power plant. The analysis results showed that the present design of the fuel assembly satisfies the allowables for the faulted conditions and therefore is dynamically qualified.

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