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DYNAMIC CHARACTERISTICS OF REACTOR CORE FOR PIPE BREAK AND SEISMIC EXCITATIONS

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This paper investigates the dynamic responses of the reactor core for main steam line and economizer feedwater line breaks. The tributary pipe breaks in lieu of main coolant loop breaks are considered because leak-before-break evaluation has provided a technical basis for the elimination of double ended guillotine breaks. This paper also calculates the response of the reactor core due to the motions induced from safe shutdown earthquake and operating basis earthquake. The dynamic responses such as fuel assembly shear force, bending moment and displacement, and spacer grid impact loads are carefully investigated. Also, reported in this paper are the different response characteristics of each pipe break and seismic excitaion.

Key Words: Pipe Break, Safe Shutdown Earthquake(SSE), Operating Basis Earthquake(OBE), Fuel Assembly, Spacer Grid

1. INTRODUCTION

For the dynamic response analysis of the reactor internals under pipe breaks or seismic excitations, the coupled model of internals and core is used by lumping all fuel assemblies into several groups. The fuel assembly response obtained from this coupled model is not accurate enough to evaluate the motion of individual fuel assembly because lumped model properties are used. Therefore, the detailed reactor core analysis is followed to determine the fuel assembly responses.

The procedure for core analysis is described briefly in the following. As the first step, reactor vessel motion is obtained from the reactor coolant system in which a very simplified model of the internals and core is used. Subsequently, reactor vessel motion is used as input to a coupled model of 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. In the last step, 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 this paper th dynamic responses of the reactor core to the pipe break and seismic excitations are obtained and their dynamic characteristics are carefully investigated.

2. FORCING FUNCTIONS

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 (USNRC, 1984, Roos, 1989). Instead, loads of branch line pipe breaks are considered for the service limit Level D which is a faulted operating condition (ASME, 1989). Of the branch line pipe breaks postulated, LBB evaluation is performed for piping systems with a diameter of 25.4 cm or over and it is anticipated that pipe breaks with a diameter of 25.4cm or over be no more considered as design basis. But the pipe breaks of economizer feedwater line (30.5 cm SCH80) in the secondary side have been reported in several plants due to water hammer, and this break is included in the design basis. The break of main steam line, which is the largest one (61.3 cm nominal ID) in the secondary side, is analyzed in this paper for the comparison purpose. Also, the effects of seismic excitations (safe shutdown earthquake and operating basis earthquake) on the reactor core are calculated.

3. DETAILED CORE MODEL

In the detailed core model, the fuel assemblies (Fig.1) 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 the possible spacer grid impacting. The gap-spring elements are used to simulate the geometric non-linearities between the fuel assemblies as well as the clearance between the peripheral fuel assemblies and core shroud. The nominal gap sizes of 3.759 and 1.905 mm are used for core shroud peripheral and fuel assembly gap, respectively.

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 static model of the fuel bundle was modified to include dynamic effects by adjusting the bundaly 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

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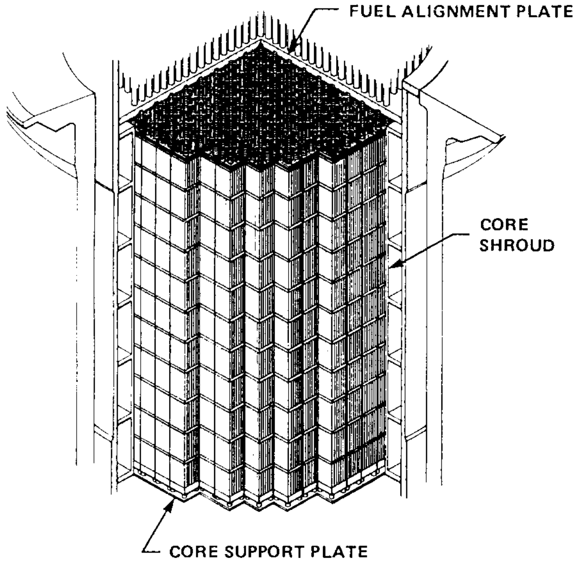


Fig. 1 Typical PWR reactor core

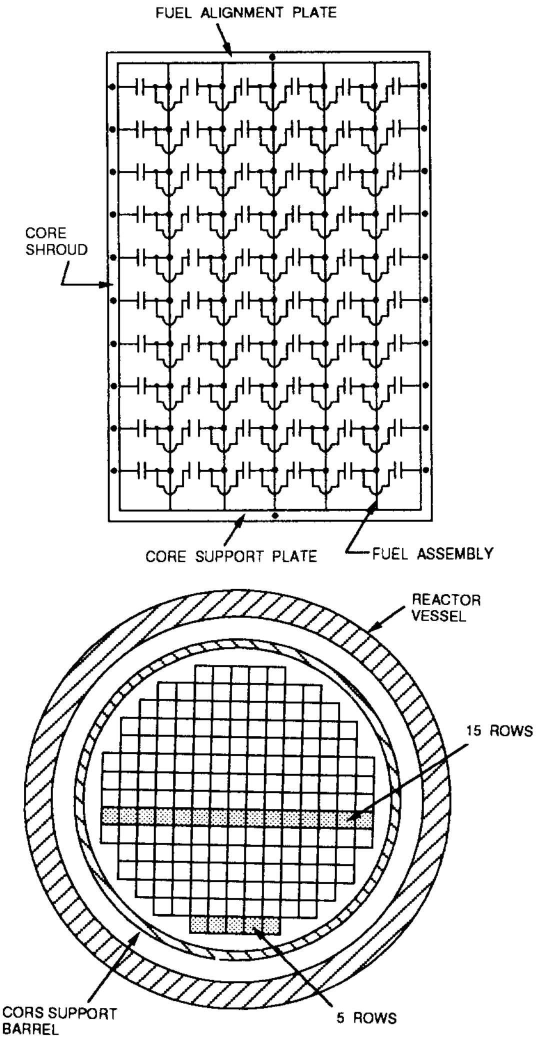


Fig. 3 Detailed core model of 5 rows and core layout

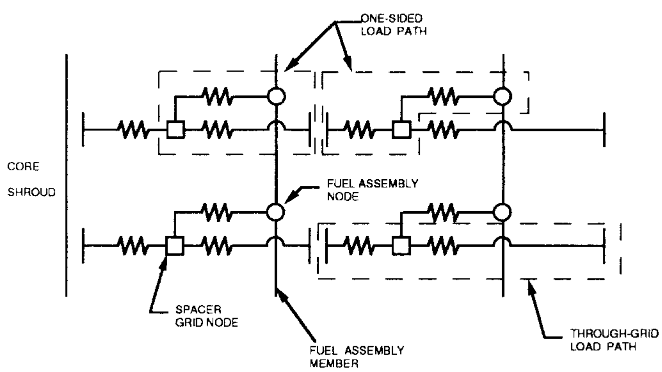


Fig. 2 Dual load path impact model of spacer grid

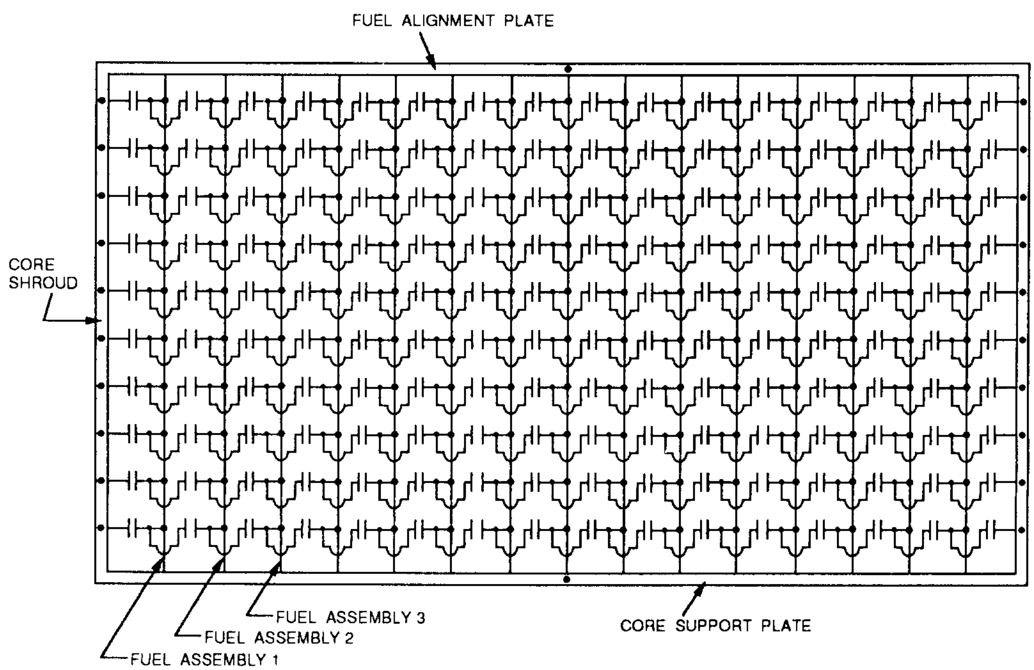
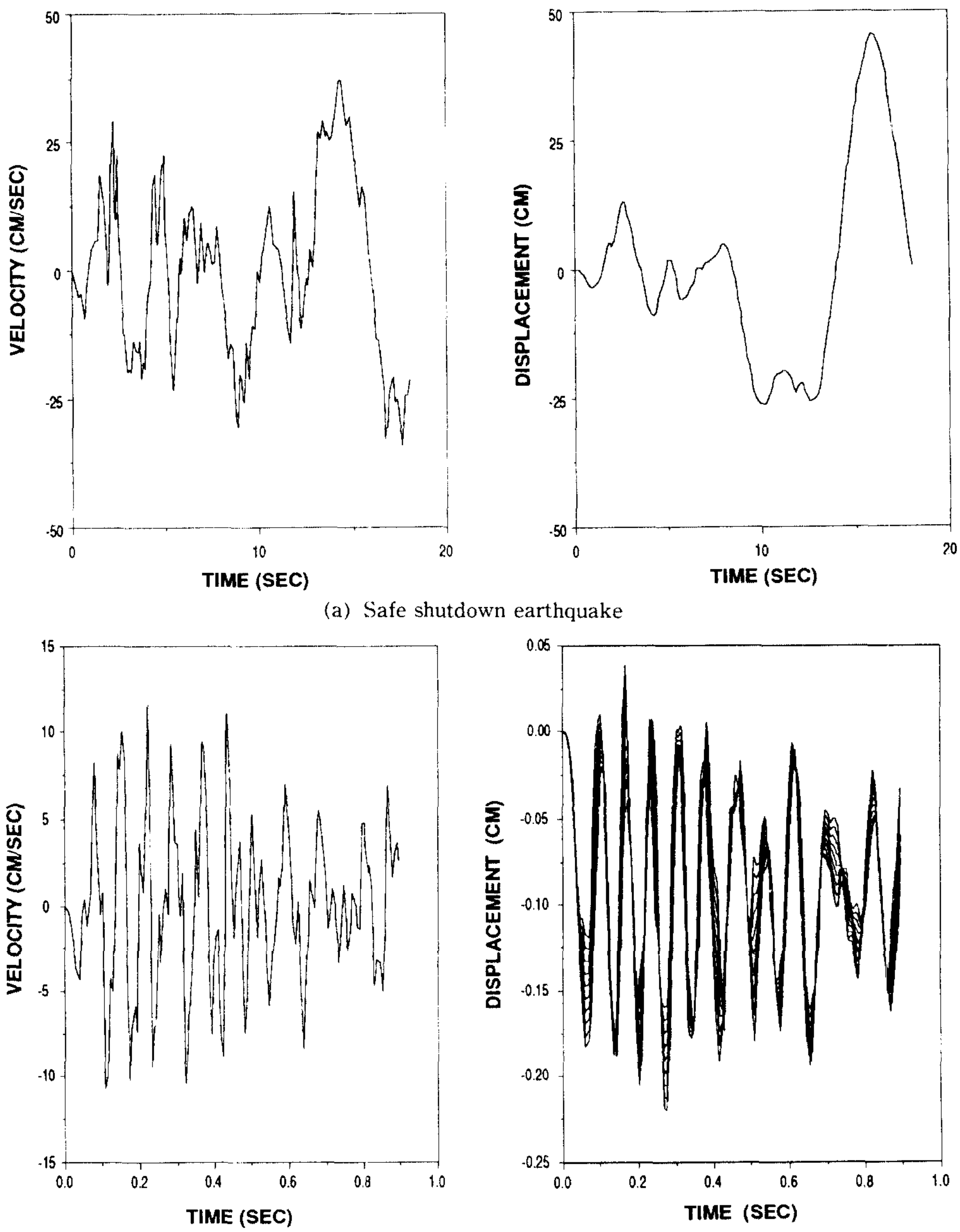


Fig. 4 Detailed core model of 15 rows

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 (Stokes, 1978), 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 differrent 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. Therefore, the spacer grid model seperates out through-grid and one-sided load paths (Fig. 2). The pluck vibration, pluck impact, spacer grid compression, and spacer grid section drop tests provide data used in determining the spacer grid impacting parameters.

Parametric studies indicated that fuel assemblies in the longest rows and in the shortest rows experience the severest response. In order to satisfy current NRC requirements for load combinations (USNRC, 1978), two separate models were developed for the shortest (5 assemblies) and longest (15 assemblies) rows across the core. Core models of five and fifteen rows and shown in Fig. 3 and Fig. 4, respectively.



(a) Safe shutdown earthquake

(b) Economizer feedwater line break

Fig. 5 Velocity time history of CSP and displacement time histories of core shroud

4. DYNAMIC RESPONSES

The input excitations to the detailed core model consist of the translational and angular time histories of the core plates and the translational time history of the core shroud. The core shroud is so stiff comparing with fuel assembly that its

Table 1 Summary of fuel assembly peak loads for pipe breaks

Load Condition	EFW BREAK		MSL BREAK	
	5 ROW	15 ROW	5 ROW	15 ROW
Relative displacement (mm)	2.474	2.474	2.497	2.497
Moment (N-m)	167.4	167.4	111.4	111.4
Shear (N)	451.0	451.0	255.8	255.8

Table 2 Summary of fuel assembly peak loads for seimic excitations

Load condition	OBE		SSE	
	5 ROW	15 ROW	5 ROW	15 ROW
Relative displacement (mm)	12.64	26.87	15.11	41.61
Moment (N-m)	396.2	409.4	531.6	856.5
Shear (N)	904.7	944.3	1134.7	1785.9

Table 3 Summary of spacer grid impact loads for seismic excitations

Load condition	OBE		SSE	
	5 ROW	15 ROW	5 ROW	15 ROW
One sided (N)	6641	8416	8749	18588
Through-grid (N)	5596	6899	7290	11089

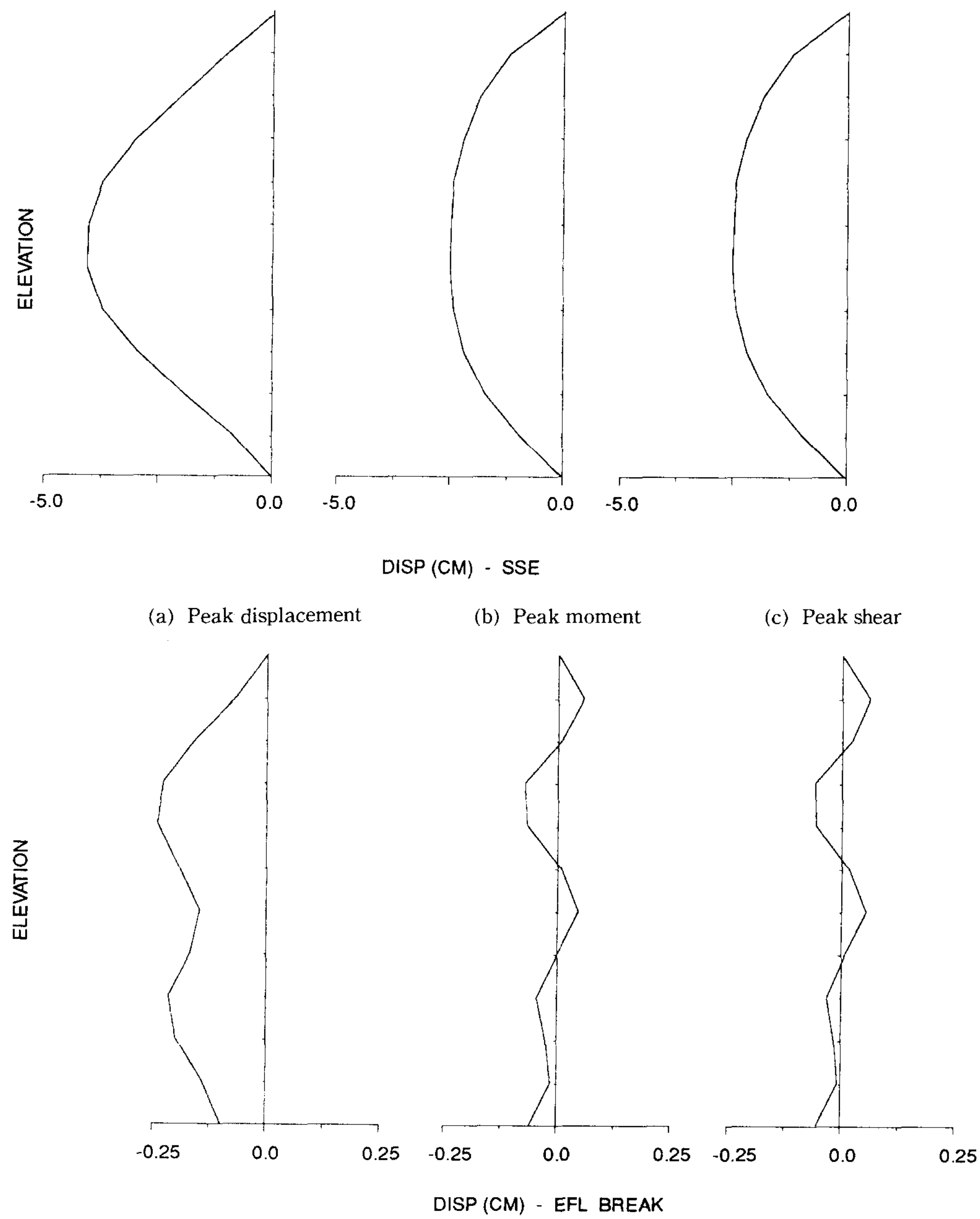


Fig. 6 Deflected shapes of fuel assembly for peak loading conditions

local effect is negligible. Therefore, only the translational component of core shroud is used. The input motions are determined from the coupled internals and core analysis (Jhung, 1991) and shown in Fig.5. The core shroud motions at 10 vertical elevations indicate that the overall components behave as if they were one lumped mass for seismic excitations. For the pipe break excitations, there is a fluctuation between components and the spacer grid is assumed to be free

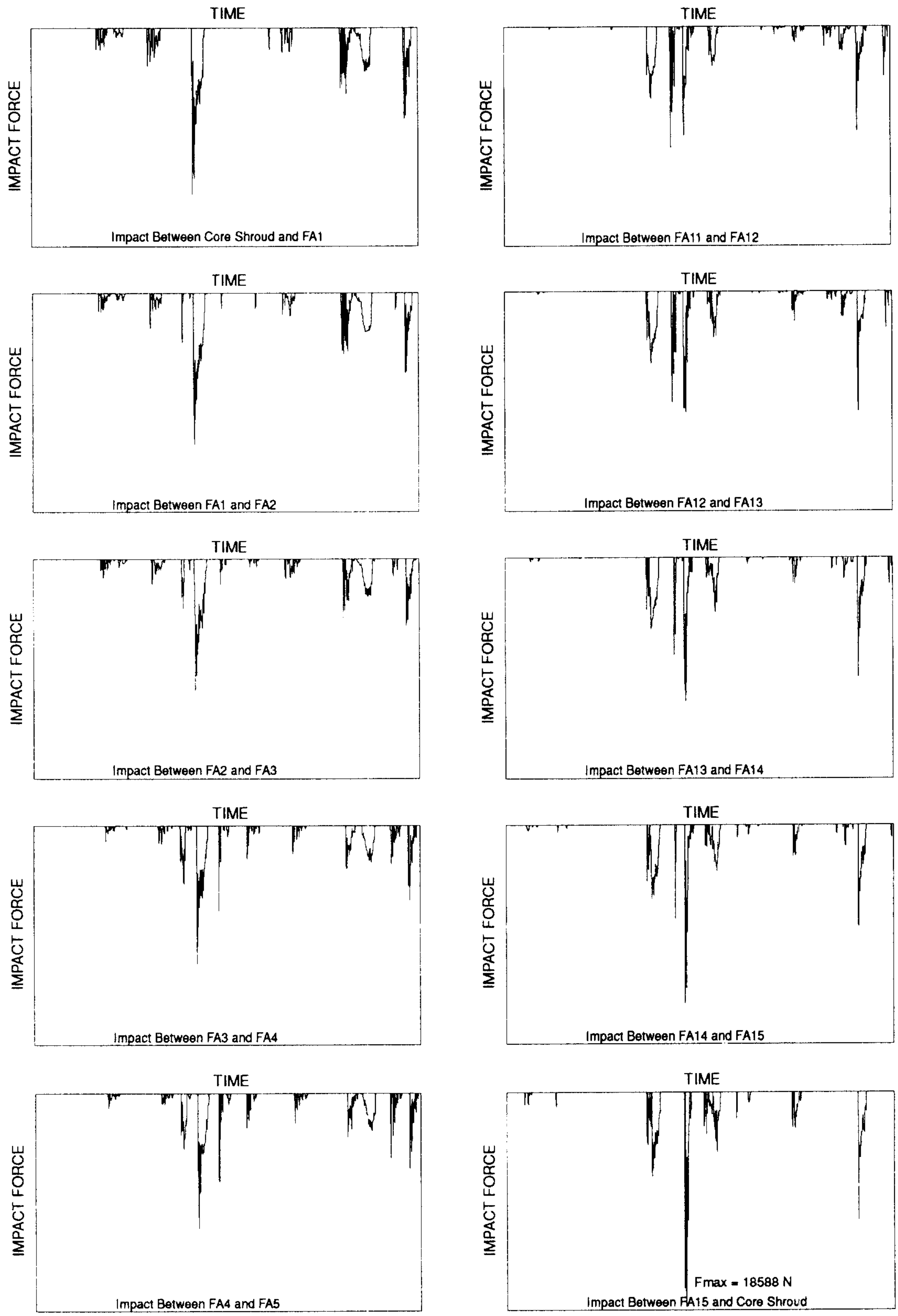


Fig. 7 Impact force on the center spacer grid of fuel assemblies

from impact load because the maximum displacement of core shroud is less than the gap size between core shroud and peripheral fuel assembly.

The equation of motion for the structural system is described by the second-order differential equation as follows

$$[M]\{A\}+[C]\{V\}+[K]\{X\}=\{F(t)\}$$

where [M], [K] and [C] are defined as the mass, spring and damping matrices and {A}, {V}, {X} and {F(t)} are

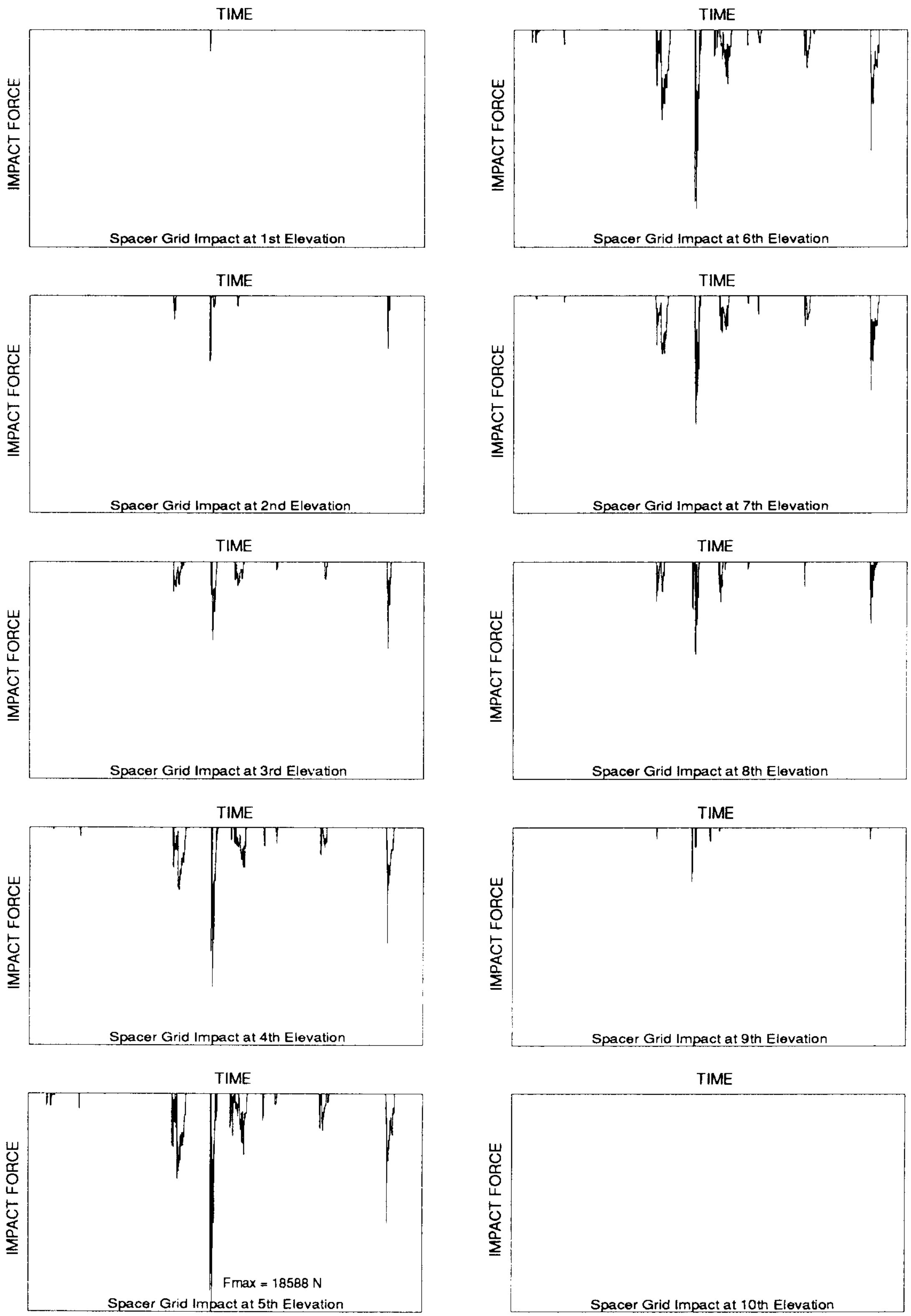


Fig. 8 Impact force on the spacer grids at each elevation

Table 4 Impact forces at center spacer grid nodes

	One-Sided IMPACT (%F_{max})														Through-Grid IMPACT (%F_{max})						
	Left Side							Right Side													
	40	50	60	70	80	90	100	40	50	60	70	80	90	100	40	50	60	70	80	90	100
FUEL ASSY 1	8	3	0	1				5	1	1					2	1					
FUEL ASSY 2	6	0	1					5							1	1	1				
FUEL ASSY 3	6	0	1					6							1	1					
FUEL ASSY 4	9	0	1					4	2						1	1					
FUEL ASSY 5	2	3	1					6	2						1	1					
FUEL ASSY 6	4	3	1					5	3	1					1	1					
FUEL ASSY 7	6	4	0	1				5	3	1					2						
FUEL ASSY 8	7	3	2	1				6	3	1	1				4						
FUEL ASSY 9	3	4	3	1				6	2	3	1				3						
FUEL ASSY10	7	2	4	1				7	4	2	1				3						
FUEL ASSY11	7	4	0	1				9	2	2					4	1					
FUEL ASSY12	7	1	2					5	4						3	2					
FUEL ASSY13	6	1	2					5	4						1	2	1				
FUEL ASSY14	4	3						4	3	2					1	3	1				
FUEL ASSY15	4	1	1	1	1			6	2	1	1	1	0	1	3	1	1				

defined respectively as the acceleration, velocity, displacement and force vectors. The responses of the fuel assemblies to the excitations were obtained using the SHOCK code (Gabrielson, 1966), which integrates the equations of motion by the Runge-Kutta-Gill method or a Newmark method for first-order differntial equations and provides the time-history response of the fuel assemblies.

The integration timestep was determined based on the impact pulse which is typically estimated to be 10 milliseconds for seismic exciation. The number of steps per pulse will be $10/(2\times10^{-4})=50$ for the constant timestep of 2×10^{-4} second, which is large enough for this kind of analysis. In this case, the maximum frequency range encompassed is $[(20)(2\times10^{-4})]^{-1}=39.8$ Hz because timestep is almost equal to (1/20)x(minimum period τ). The 39.8 Hz is wide enough to cover the fuel frequencies becuase fuel assembly responds to the seismic exciation by moving back and forth approximately at the first mode frequency of 1 Hz. The integration timestep for the pipe break excitation was also determined in the same way as in the seismic case.

5. RESULTS AND DISCUSSION

The result of the detailed core analysis consists of peak spacer grid impact loads, fuel assembly moments, shear forces and deflected shapes. The impact loads are used to evaluate the structural integrity of spacer grids. The deflected shapes which corrspond 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. For the pipe break excitation considered here, no spacer grid impact load exists. Only the impact loads due to seismic excitation will be used to evaluate spacer gird integrity. The peak loads and deflected shapes are shown in Tables 1 through 3 and Fig. 6. The deflected shapes for seismic excitations indicate that the fuel assemblies respond to the seismic excitation by moving back and forth across the core approximately a their first mode natural frequencies. Whereas, it is remarkable that high modes (i.e., over 3rd mode) contribute to the peak loads for the pipe break excitations. Fig. 7 shows the time histories of the impact loads acting on the center spacer grides for a short time period. It is clearly seen that the magnitede of the spacer grid impact force decreases as the fuel assembly gets away from the core shroud. The spacer grid impact loads at 10 elevations for two adjacent fuel assemblies are shown in Fig. 8. The peak impact is found at the spacer grid located in the middle of the fuel assembly.

Table 4 gives the impact histories for SSE; each cell contains the number of occurrence of the impact force between the center spacer grids of the various assemblies in the row. For example, 6 one-sided impacts for the spacer grid (right side) of fuel assembly 15 have a maximum force less than $0.4F_{max}$, 2 impacts have a maximum force between $0.4F_{max}$ and $0.5F_{max}$, etc. For the one-sided forces, the impacts on the left and right side of the gird are independent. The damage (if any) to a grid from an one-sided impact occurs to the first few cells, i.e., a left-sided impact does not affect the right side of the grid.

6. CONCLUSIONS

The dynamic responses of the reactor core are investigated for the pipe break and seismic excitations. The following conclusions are obtained:

For spacer grid loads:

(1) Spacer grid does not contact each other from the main steam line and economizer feedwater line break excitations, and therefore no spacer grid impact loads occur.

(2) For seismic excitations, maximum spacer grid impact loads occur on the longest row model.

For fuel assembly responses (shear force, moment and deflection):

(3) The maximum fuel assembly responses occur at the core periphery for both pipe break and seismic excitaitons.

(4) The maximum fuel assembly deflection occurs in the middle of the fuel assembly height for both pipe break and seismic excitations.

(5) The deflected shape of fuel assembly at peak loading conditions corresponds to the first mode for seismic excitations and to the third or fifth mode for pipe break excitations.

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