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Lift-off characteristics of PWR fuel assemblies for tributary pipe break and earthquake

Jhung, M.J., Kim, T.H.
Korea Atomic Energy Research Institute, Power Reactor Mechanical Engineering, Taejon, Korea

ABSTRACT : A general approach to the dynamic time-history analysis of the coupled internals and core in the vertical direction is presented as a part of the fuel assembly qualification program. A fuel assembly of a pressurized water reactor is not restrained mechanically in the vertical direction and it is necessary to check an uplift for a large dynamic motion. Dynamic analyses are performed for the excitations induced from earthquake and pipe break. The peak responses such as fuel assembly axial forces and lift-off characteristics are investigated.

1 INTRODUCTION

The reactor core of a pressurized water reactor (PWR) 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. The guide tubes, the structural frame of the assembly, are individually fitted into circular tubes and pins held by the FAP and CSP, respectively. They are positioned by preload only and are not restrained mechanically in the vertical direction. As a result, the fuel assemblies may suffer uplift when they are excited by a large vertical dynamic motion. Thus, in an earthquake or pipe break event, their vibratory motions may have a complicated nature including non-linearity due to the effects of frictions between fuel rods and grid cage.

Safety qualification of the reactor core is one of the crucial issues in the faulted condition design of a PWR, and it should be secured that the structural integrity of the fuel assemblies and the vertical height of the guide pins be not exceeded against the design loads. The procedure of analysis is described briefly as follows. As the first step, reactor vessel motion is obtained from the reactor coolant system analysis in which a very simplified model of the internals and core is used. Subsequently, the reactor vessel (RV) motion is used as an input to a coupled model of internals and core. The analysis of internals and core generates design loads of reactor internals and fuel assembly. For the pipe break analysis, the hydraulic loads as well as reactor vessel motions are used as forcing terms. But there exists no hydraulic load for the earthquake excitation and also it is negligible for the secondary side pipe breaks.

For the purpose of assessing safety of the reactor core, various efforts for the dynamic response of the fuel assemblies in the horizontal direction have been made to understand dynamic characteristics [1, 2, 3]. By contrast with the horizontal direction, the axial responses are not investigated much because they are not important relatively in the structural integrity point of view [4, 5]. But it should be pointed out that fuel assemblies should not lift off the core plates during the transient.

In the present study, a method for dynamic analysis of the reactor internals and core is developed. The SHOCK computer code [6] is used to solve the dynamic response of the lumped mass systems. Two earthquakes - operating basis earthquake (OBE) and safe shutdown earthquake (SSE) - and six - 4 primary side and 2 secondary side - tributary pipe breaks are analyzed. Peak responses such as fuel assembly axial forces and lift-off characteristics are investigated.

2 DESCRIPTION OF FUEL ASSEMBLY

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. The fuel assembly, which provides for 236 fuel rod positions (16 x 16 array), includes 5 guide tubes welded to 11 spacer grids and is closed at the top and bottom by end fittings. The guide tubes each displace four fuel rod positions and provide channels which 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 in-core 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*

The mathematical model of the internals and core 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 control element assembly guide tube and upper end fitting of the fuel assembly. 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, stiffness is calculated using the well known formula as $K = AE/L$, where K, A, E and L are axial stiffness (lb/in), cross-sectional area (in^2), Young's modulus (psi) and length of segment (in), respectively. If a beam has one or more changes in sections, the equivalent stiffness is the series sum of the parts. Stiffnesses for the complex structures such as flanges and tube bank assembly are determined by finite element analyses.

The core is modelled by grouping all fuel assemblies into a single grouping which includes the weight of the entire core. The vertical stiffness properties of the grouping are obtained by combining the individual stiffnesses of all the fuel assemblies in the core as springs in parallel. The fuel assembly grouping is subdivided into fuel rods and guide tubes with slip stick friction elements representing the connectivity between the two single stick models. This connectivity represents the friction force between the fuel rods and spacer grid arches and tabs. Both static and dynamic friction values are used. The fuel rods do not slip until the value of static friction is exceeded. When slipping of the fuel rods occurs, the resisting force is equal to the dynamic friction force. Beginning-of-life friction values are determined from tests by measuring the forces required to withdraw fuel rods from the bundle in an unirradiated condition. To determine end-of-life friction values, beginning-of-life experimental friction values are adjusted by factors based on the estimated relaxation of the grid material.

3.2 *Input excitations*

The input excitations to the model consist of the hydraulic loads of internals and reactor vessel motions. The reactor vessel motions are determined from the reactor coolant system analysis and are acceleration time histories at the RV ledge. The internals hydraulic loads are calculated using a control volume formulation, where the internal structures and contained water are sectioned into solid plus fluid control volumes. Across each volume, the fluid momentum equation is solved as a function of time to compute the hydraulic loads. This method accounts for fluid pressure and momentum effects which act on all of the structures within each control volume.

The vertical loads of the core model are calculated using the pressure differentials across the ends plus the drag and fluid momentum terms. Separate loads are calculated for individual nodes presenting the fuel rods, guide tubes, upper-end-fitting (UEF) and lower-end-fitting (LEF).

Drag loads represented by fluid shear term are composed of two components - friction drag and form drag. These loadings are dependent on the channel equivalent diameter, channel cross-sectional area, fluid flow rate and fluid density. Frictional drag is apportioned to the guide tubes and to the fuel rods on the basis of fraction of total wetted perimeter adjacent to a given flow channel or subchannel. The only form losses present are due to the spacer grids. Crud effects are accounted for by multiplying the drag loads by an empirically determined factor. Time history plots of the total vertical hydraulic loads are shown in Fig. 1. Also, acceleration time histories of the RV ledge are shown in Fig. 2. For the earthquake excitations, no hydraulic loads exist, therefore the only forcing terms are the RV motions as shown in Fig. 3 for a safe shutdown earthquake.

3.3 *Dynamic responses*

Equilibrium conditions, prior to the application of the dynamic transient conditions, are established by determining the static displacements associated with the weight of the internals and core in water, preloads and the core drag steady state forces. These calculated static displacements are used as the initial conditions. Without these, some of the masses would be subjected to large accelerations because of the resulting force unbalance.

The responses of the fuel assemblies to the excitations were obtained using the SHOCK code, which integrates the equations of motion by the Runge-Kutta-Gill method for first-order differential equations and provides the time-history response of each component [6].

4 RESULTS AND DISCUSSION

The result of the internals and core analysis consists of initial, total maximum and minimum loads on the reactor internals and core. The design loads are determined by subtracting initial loads from total loads and are used to assure the structural integrity of the components (Table 1). In addition, maximum axial direction fuel assembly end fitting displacements relative to the core support plate and fuel alignment plate are examined to insure that vertical height of the guide pins is not exceeded. Figs. 4 and 5 show a relative displacement time histories of the fuel assembly lower end fitting with respect to the core support plate for pressurizer (PZR) breaks and SSE, respectively. The non-zero values in the displacement time history are associated with lifting fuel assemblies from the core plates. The amount of lift-off depends on the magnitude of the particular site specific excitation. The magnitude of this relative displacement is examined to ensure that vertical assemblies from the core plates. The amount of lift-off depends on the magnitude of the particular site specific excitation. The magnitude of this relative displacement is examined to ensure that vertical height of the guide pins is not exceeded. Guide pins have a height (3.125 inch) which is significantly larger than the relative displacement as indicated in Table 2.

The response spectra plot for RV ledge and UEF of fuel assembly are shown in Fig. 6 for PZR break and SSE. The peak of the fuel assembly spectra is amplified by a factor of 5.3 from the excitation point in the 4" PZR break transient. But almost 2 times is excited for earthquake case. This indicates that the hydraulic loads considered in the pipe break analysis play an important part in the dynamic response. That's why the primary side pipe break produce higher responses than the secondary side pipe break, which results from the fact that the former uses RV motion and internals hydraulic loads as forcing functions but RV motion only is applied for the forcing term in the latter. The opposite was true for the horizontal response of the reactor core [2].

The vertical analyses of various breaks indicated that the response obtained from a PZR spray line intermediate break would result in a maximum impact force, in this case an impact load, of 394 lbs at the bottom of the fuel assembly. Based on the maximum responses obtained from the faulted conditions, it should be shown that the fuel assembly is structurally capable of resisting the hypothetical accident and also adequately designed to remain functional, i.e., the core coolable geometry will be maintained throughout the transient.

5 CONCLUSIONS

The method to calculate axial response of the reactor internals and core due to the dynamic excitations is developed. Dynamic analyses were performed for the two earthquake and six tributary pipe break excitations. The axial forces are calculated and the fuel assembly lift-off is examined to guarantee the proper operation of the nuclear power plant. As indicated in the results, the lift-off of fuel assembly does not exceed the guide pins even for the SSE condition. Also, it is shown that the responses due to the tributary pipe break conditions are lower than those of SSE in the faulted condition design.

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Table 1. Axial forces of fuel assembly (unit = lbs)

Excitation	Fuel rods	Guide tubes
3" PZR spray line nozzle break	225	20
4" PZR spray line intermediate break	394	35
6" PZR safety valve inlet nozzle break	73	6
3" SDC long-term SI line intermediate break	198	18
SG feedwater economizer nozzle break	84	7
SG steam line nozzle break	106	9
Operating basis earthquake	279	25
Safe shutdown earthquake	506	45

Table 2. Relative deflections of fuel assembly end fittings to core plates (unit = inch)

Excitation	LEF - CSP	UEF - FAP
3" PZR spray line nozzle break	-.6035E-4	-.2057E0
4" PZR spray line intermediate break	-.6927E-4	-.2066E0
6" PZR safety valve inlet nozzle break	-.5229E-4	-.2047E0
3" SDC long-term SI line intermediate break	-.5681E-4	-.2047E0
SG feedwater economizer nozzle break	-.7060E-4	-.1369E0
SG steam line nozzle break	-.7179E-4	-.1372E0
Operating basis earthquake	-.5191E-4	-.1339E0
Safe shutdown earthquake	-.4059E-4	-.1321E0

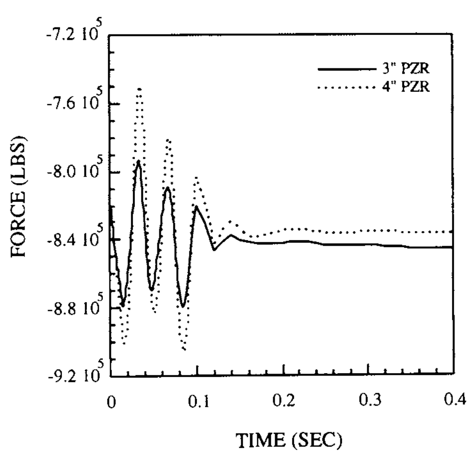

Fig. 1 Force time histories for PZR spray line break

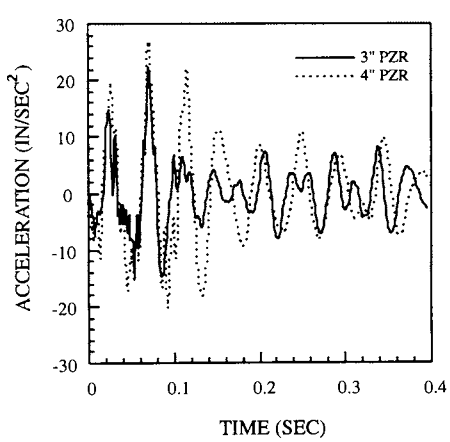

Fig. 2 RV motions for PZR spray line break

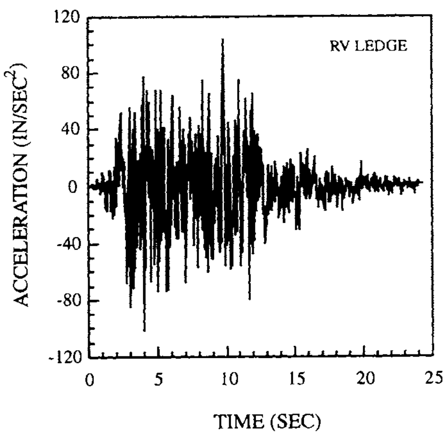

Fig. 3 RV motion for safe shutdown earthquake

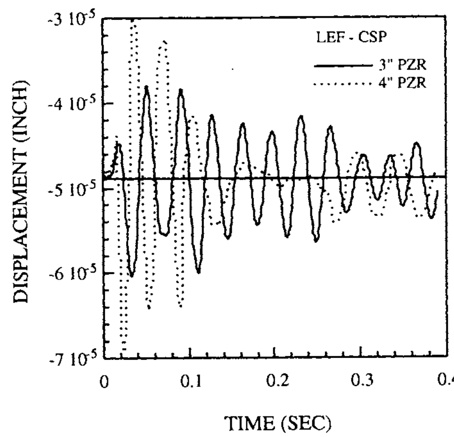

Fig. 4 Relative displacement time histories for PZR break

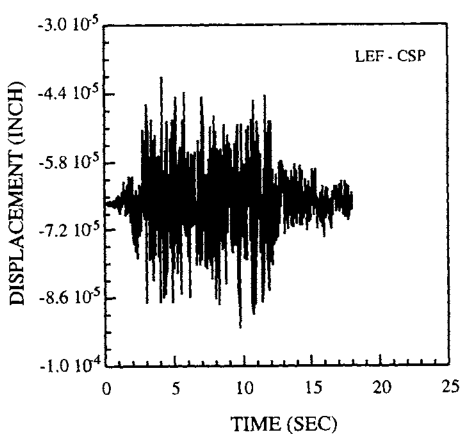

Fig. 5 Relative displacement time histories for SSE

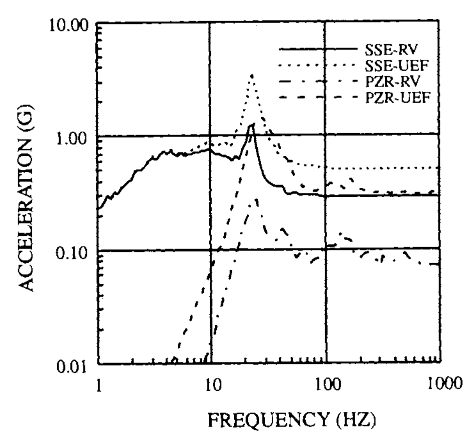

Fig. 6 Response spectra for 4" PZR break and SSE