SMiRT–12 / K. Kussmaul (Editor)

EFFECT OF FUEL ASSEMBLY MECHANICAL DESIGN CHANGES ON DYNAMIC RESPONSE OF REACTOR PRESSURE VESSEL SYSTEM UNDER EXTREME LOADINGS

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ABSTRACT

This paper presents the results of a study to assess the effect of fuel assembly mechanical design changes on the dynamic response of a pressurized water reactor vessel and reactor internals under Loss-Of-Coolant Accident (LOCA) conditions. The results of this study show that the dynamic response of the reactor vessel internals and the core under extreme loadings, such as LOCA, is very sensitive to fuel assembly mechanical design changes.

1.0 INTRODUCTION

In the design of nuclear power plants it is not only imperative that the reactor's internal components maintain their structural integrity under extreme loads, but is also important that the core-coolable geometry of the reactor is maintained. These extreme load conditions are, in general, the Loss-Of-Coolant Accident (LOCA) and the Safe Shutdown Earthquake (SSE). For a typical pressurized water reactor the reactor vessel, the reactor internals, and the reactor core are shown in Figure 1. The reactor core consists of many individual, but structurally identical, fuel assemblies. Each fuel assembly consists of a square array of closely spaced fuel rods retained in place by spacer grids at various elevations. The fuel assembly, supported by top and bottom nozzles, is held in place by the reactor internals' upper and lower core plates. A baffle plate assembly surrounds the core and guides the coolant flow through the fuel assemblies. In the reactor core, gaps exist between adjacent fuel assemblies as well as in between the peripheral fuel assemblies and baffle plates.

In order to ensure that the reactor core components maintain their structural integrity under extreme loads (LOCA and SSE), detailed nonlinear reactor pressure vessel (RPV) system dynamic analyses are usually performed. Mechanical non-linearities in the reactor pressure vessel system are created by gaps between adjacent components, frictional sliding between various interfaces and potential lift-off the components from their supports. Since the reactor internals sub-assemblies are not structurally fastened to adjacent members or to the supporting ledge of the vessel flange, they may slide and impact when subjected to external forces. Thus, the transient dynamic analysis of the RPV system includes impacting due to gap closures, energy losses due to impacting bodies and Coulomb friction between the sliding surfaces.

Most solution procedures for transient dynamic analysis can be divided into two categories: a) direct integration method; and b) modal superposition method. Application of direct integration method to complex nonlinear dynamic problems typical of nuclear power plants under seismic and LOCA excitations become extremely expensive because of excessive computational resources, or cost. Therefore, to reduce the computational cost of such nonlinear problems, the application of modal superposition method [1] is considered. Reference [2] describes the application of modal superposition technique to study the effect of Coulomb friction on the dynamic response of the reactor pressure vessel system under extreme loadings.

This paper presents nonlinear transient dynamic time history analyses of a pressurized water reactor vessel, reactor internals, and its core subjected to LOCA loads. The results of these non-linear analyses show that changes in mechanical characteristics of the core (fuel assemblies) alter the dynamic response of the reactor pressure vessel system, and therefore, may have significant influence on the dynamic response of the reactor internals and core.

2.0 MATHEMATICAL PRELIMINARIES

The generalized equation of motion of a structure is given by:

$$[M]\ \{\ddot{X}\} + [C]\ \{\dot{X}\} + [K]\ \{X\} = \{F\} \qquad (2\text{-}1)$$

where [M], [C] and [K] denote mass, damping and stiffness matrices, respectively. The displacement, velocity, acceleration, and the applied force vectors are represented by $\{X\}$, $\{\dot{X}\}$, $\{\ddot{X}\}$ and $\{F\}$, respectively. The dependence of $\{F\}$ on the spatial coordinates and time is understood.

The damping and stiffness matrices in eqn. (2-1) are nonlinear in terms of displacement vector $\{X\}$ and are decomposed into linear and nonlinear parts:

$$[C] = [C_l] + [C_{nl}] \qquad (2\text{-}2a)$$

$$[K] = [K_l] + [K_{nl}] \qquad (2\text{-}2b)$$

Here $[C_l]$ and $[K_l]$ are damping and stiffness matrices representing the reference state of the structure; $[C_{nl}]$ and $[K_{nl}]$ are the non-linear parts dependent on velocity and displacement. Substitution of eqns. (2-2a) and (2-2b) into eqn. (2-1) yields:

$$[M]\ \{\ddot{X}\} + [C_l]\ \{\dot{X}\} + [K_l]\ \{X\} = \{F\} - \{F_{nl}\} \qquad (2\text{-}3)$$

where the pseudo force vector is defined by:

$$\{F_{nl}\} = [C_{nl}]\ \{\dot{X}\} + [K_{nl}]\ \{X\} \qquad (2\text{-}4)$$

Equation (2-3) is solved by the use of modal decomposition whose modes are solution of eqn. (2-3) with vanishing righthand side. The homogeneous, undamped equation of motion representing the reference state of the structure is:

$$[M]\ \{\ddot{X}\} + [K]\ \{X\} = \{0\} \qquad (2\text{-}5)$$

Let $[\Omega]$ and $[\Psi]$ denote the natural frequency and normalized mode shape matrices associated with eqn. (2-5). Then with the aid of transformation:

$$\{X\} = [\Psi]\ \{q\} \qquad (2\text{-}6)$$

and the orthogonality relations

$$\begin{aligned} [\Psi]^T\ [M]\ [\Psi] &= [I] \\ [\Psi]^T\ [K_l]\ [\Psi] &= [\Omega_j^2] \qquad (2\text{-}7) \\ [\Psi]^T\ [C_l]\ [\Psi] &= [2\Psi_j\ \xi_j] \end{aligned}$$

The resulting modal equations become:

$$\{\ddot{q}\} + [2\Omega_j\ \xi_j]\ \{\dot{q}\} + [\Omega_j^2]\ \{q\} = \{Q\} - \{Q_{nl}\} \qquad (2\text{-}8)$$

where,

ξ_j = percent of critical damping for jth mode

$\{Q\} = [\Psi]^T \{F\}$, generalized applied force vector

$\{Q_{nl}\} = [\Psi]^T \{F_{nl}\}$, generalized pseudo force vector

Arrays $\{q\}$, $\{\dot{q}\}$, and $\{\ddot{q}\}$ are the modal displacements, velocity and accelerations, respectively. The generalized pseudo force vector is a function of displacement and velocity. Equation (2-8) represents a set of uncoupled equations. These equations are integrated analytically to eliminate numerical damping or frequency distortions during integration; the integration scheme is described in [1].

3.0 FINITE ELEMENT REPRESENTATION OF REACTOR VESSEL SYSTEM

As discussed earlier, a typical pressurized water reactor vessel, reactor internals and its core are shown in Figure 1. The reactor internal components can be characterized by two major assemblies consisting of upper and lower internals assemblies. The main purpose of these assemblies is to channel flow and provide support and restraint to the core (fuel assemblies). The upper internals assembly consists of upper support plate, guide tubes, upper support columns, and upper core plate. The upper support plate transfers the dynamic loading of the upper internals assembly in the vessel. The lower internals assembly consisting of the lower support plate, core barrel, lower support columns, and the lower core plate provides support to the core and transfer loads to the vessel through the core barrel flange.

The mathematical representation of the reactor pressure vessel system shown in Figure 2 is a three-dimensional nonlinear finite element model. This model, which is developed using WECAN [3] Computer Code, is best described as three concentric submodels consisting of a combination of beam and spring elements in addition to nonlinear impact (gap) and friction elements. The outermost submodel represents the reactor vessel and its support structure. The second submodel represents the reactor core barrel assembly and is located inside the reactor vessel model. The third, or innermost submodel represents the upper and lower support plates, the upper and lower core plates, and the fuel assemblies. The horizontal impact between the submodels is simulated using concentric nonlinear gap elements. The core (fuel assemblies) representation, which is supported between the upper and lower core plates, is also shown in Figure 2.

4.0 DYNAMIC RESPONSE OF REACTOR PRESSURE VESSEL SYSTEM

The results from numerous dynamic analyses indicate that the response of the reactor pressure vessel system under extreme loadings of seismic and LOCA is very sensitive to seemingly insignificant mechanical design changes to the fuel assemblies. These fuel assembly mechanical design changes may include: addition of extra grids; changes in grid elevations in addition to changes in fuel assembly masses and stiffnesses; and the design of fuel grids. Design modifications of this nature typically manifest themselves by changes in vibrational characteristics of the fuel assemblies which in turn influence the response of the reactor internals. For example, the results reported herein are indicative of large variations in the response of the core plate motions, which in turn are considered important for the determination of fuel grid impact loads and the loads on certain reactor internal components. In some cases, the fuel assembly mechanical design changes diminish the design margins of both the reactor internals and fuel assembly. The reduction in design margins is, in general, limited by the structural integrity of the reactor core as compared to the reactor vessel and its internals.

The fuel assembly response resulting from the LOCA and seismic events have been analyzed using time history numerical techniques. The reactor core mathematical model which simulates the fuel assembly interaction during lateral excitation consists of fuel assemblies arranged in a plannar array with inter-assembly gaps. The input to this mathematical model for determining the magnitude of fuel grid impact loads is the time history motions for the lower core plate, upper core plate, and the core barrel. Shown in the following figures are the dynamic responses of the lower core plate, core barrel, and the upper core plate with two different kinds of fuel design when subjected to LOCA transients.

Figure 3 shows the normalized time history displacements of the lower core plate with Type A and Type B fuel assembly designs. It is clearly seen from Figure 3 that with the two different fuel assembly designs, although they are mechanically compatible with the reactor internals, the maximum peak to peak displacement of the lower core plate varies by approximately 40 percent. Similarly, Figure 4 also shows the time history motions of the core barrel at the elevation of the upper core plate.

The time history displacements of the upper core plate are transformed into spectral accelerations as a function of frequency. This is done in order to show the spectral accelerations also change when the fuel design is altered. Thus, shown in Figures 5 and 6 are the ratio of spectral accelerations as a function of frequency for the upper and lower core plates, respectively, with two different fuel assembly designs under consideration. It is therefore apparent, depending on the resonant frequencies and other dynamic characteristics of the components involved, that variations in loadings of 50% or more can be realized from minor fuel assembly mechanical design changes.

5.0 CONCLUSIONS

From the above results it is evident that fuel assembly mechanical design changes which appear to small can significantly effect the dynamic response of the reactor pressure system. Consequently, it is important that whenever the fuel assembly mechanical design is changed, a system dynamic analysis of the reactor pressure vessel system must be performed to ensure that the fuel assembly design is dynamically compatible with the reactor pressure vessel system. Also if the fuel assembly mechanical design for subsequent reloads is different from the design basis core, re-analysis may be warranted to verify that sufficient design margins for the reactor internals and reactor vessel are maintained for the faulted conditions. Furthermore, the structural integrity of the core must be evaluated to ensure that the reactor core-coolable geometry is maintained in compliance with the USNRC requirements [4].

6.0 REFERENCES

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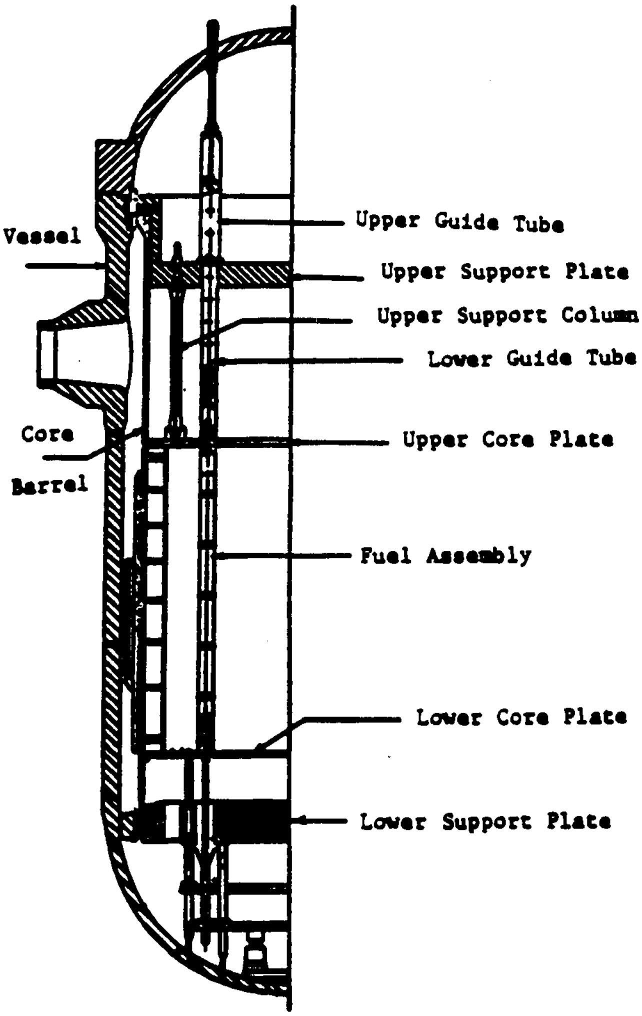

Figure 1

Reactor Vessel and Internals Components

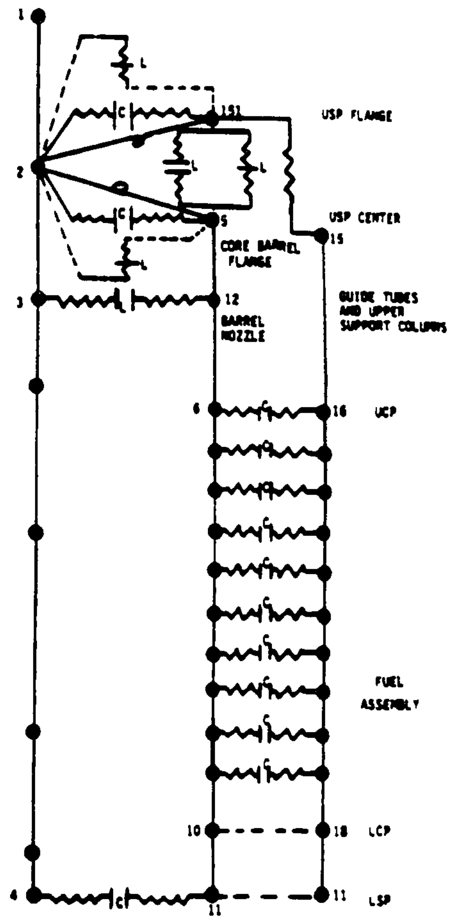

Figure 2

Finite Element Representation of Reactor Pressure Vessel System

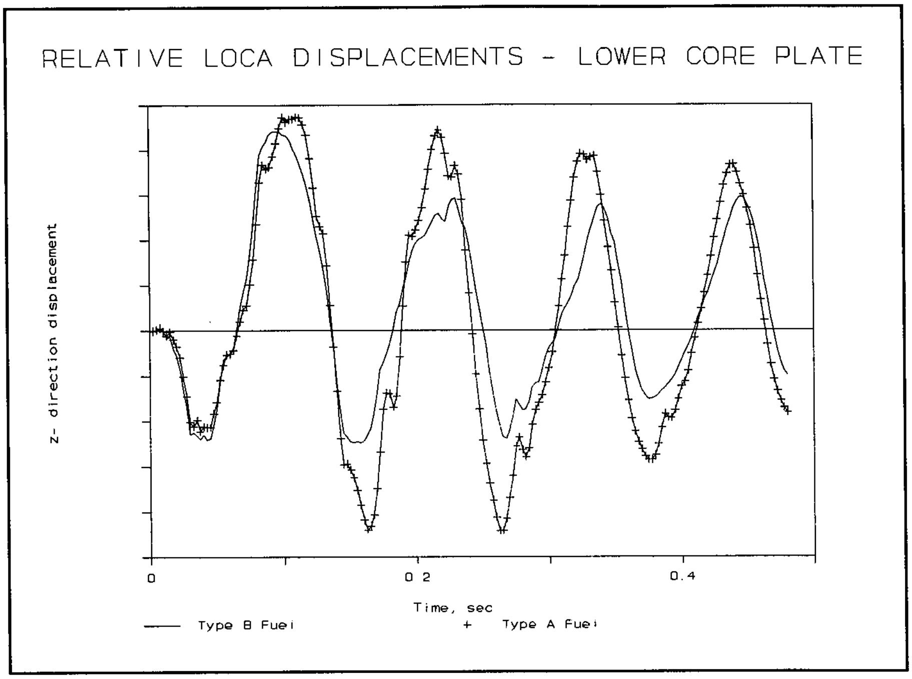

Figure 3. Comparison of typical displacements of the Lower Core Plate for Type A vs. Type B fuel during LOCA event

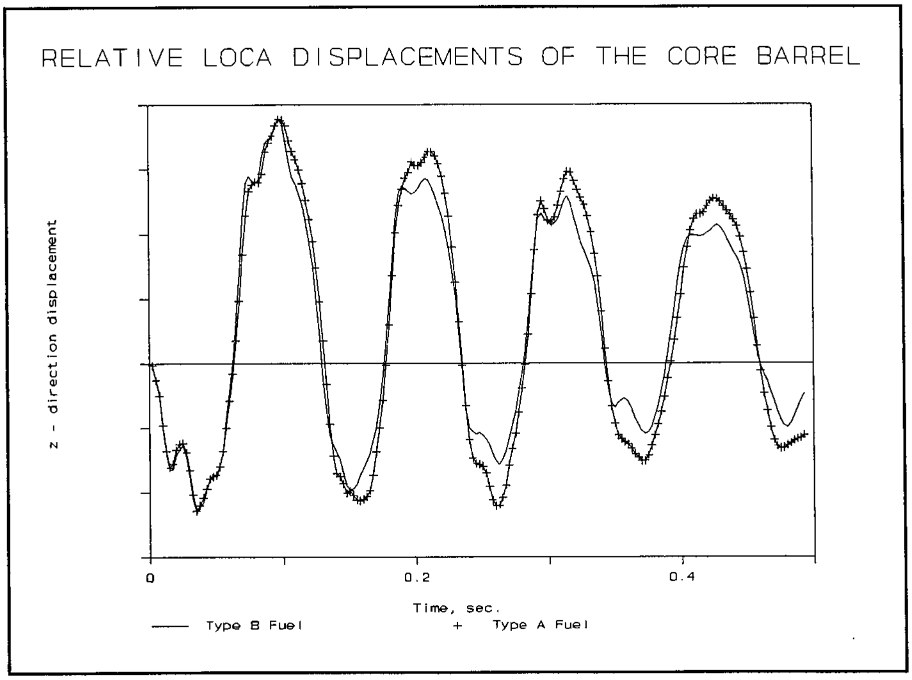

Figure 4. Comparison of typical displacements of the Core Barrel for Type A vs. Type B fuel during LOCA event

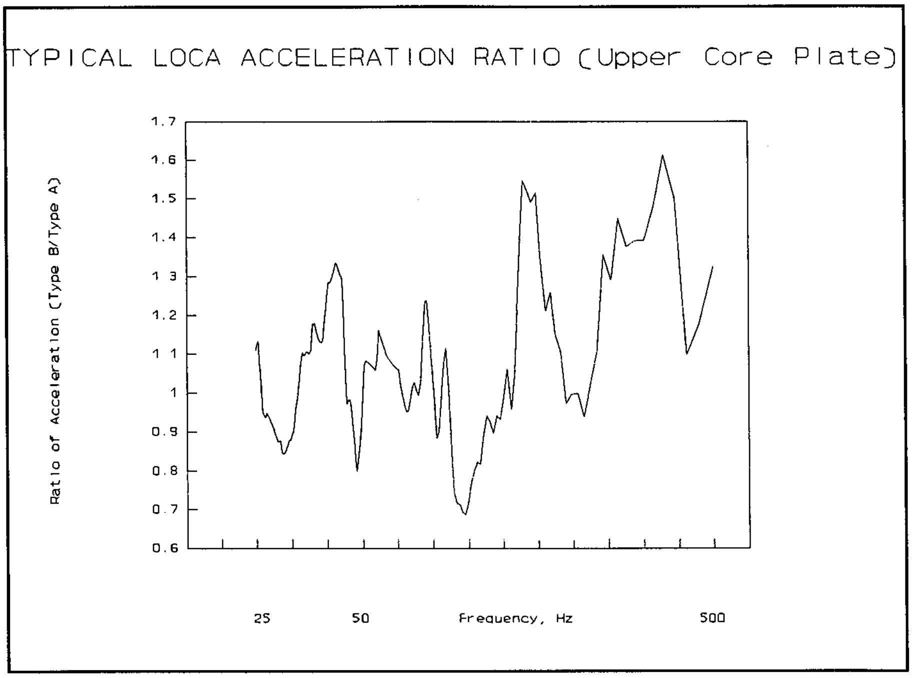

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Figure 5. Ratio of spectral accelerations (Type B/Type A) as a function of frequency for the Upper Core Plate

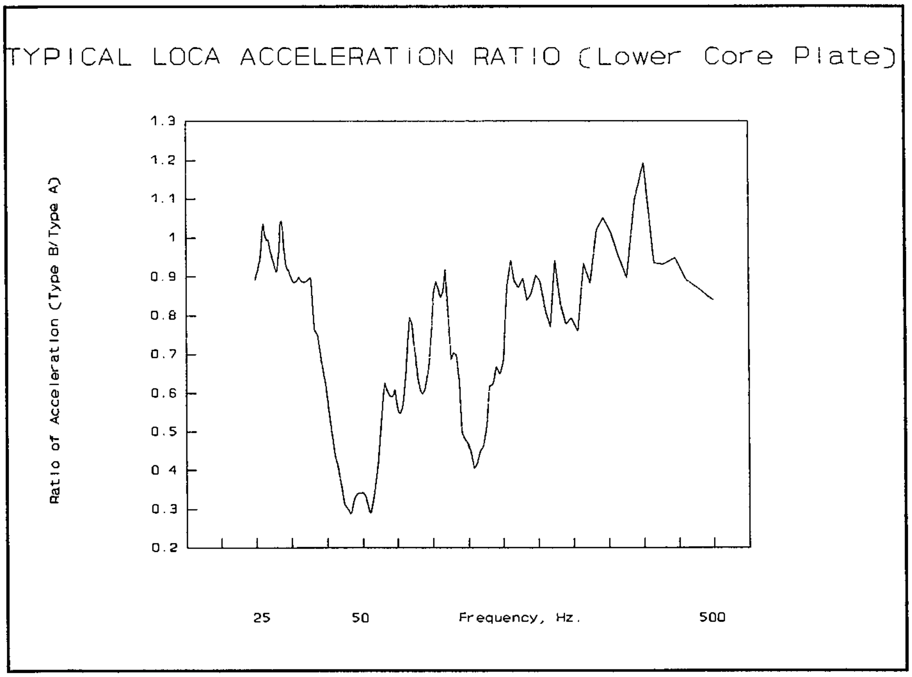

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Figure 6. Ratio of spectral accelerations (Type B/Type A) as a function of frequency for the Lower Core Plate