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〈論　文〉

Structural Integrity of a Fuel Assembly for the Secondary Side Pipe Breaks

2차측 배관파단에 대한 핵연료 집합체의 구조 건전성

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Key Words : Reactor Core(원자로노심), Fuel Assembly(핵연료집합체), Spacer Grid(지지격자체), Pipe Break(배관파단)

ABSTRACT

The effect of pipe breaks in the secondary side is investigated as a part of the fuel assembly qualification program. Using the detailed dynamic analysis of a reactor core, peak responses for the motions induced from pipe breaks are obtained for a detailed core model. The secondary side pipe breaks such as main steam line and economizer feedwater line braks are considered because leak-before-break methodology has provided a technical basis for the elimination of double ended guillotine breaks of all high energy piping systems with a diameter of 10 inches or over in the primary side from the design basis. The dynamic responses such as fuel assembly shear force, bending moment, axial force and displacement, and spacer grid impact loads are carefully investigated. Also, the stress analysis is performed and the effect of the secondary side pipe breaks on the fuel assembly structural integrity under the faulted condition is addressed.

요　약

본 연구에서는 핵연료집합체의 검증계획의 일환으로 2차측 배관파단의 영향을 조사하였다. 원자로노심의 상세모델을 이용한 동적해석으로 배관파단에 의한 응답을 구하였다. 파단전 누설개념의 적용으로 10인치이상의 고에너지 배관에 대하여 양단파단이 설계에서 배제됨에 따라 본 연구에서는 주증기관과 급수관의 파단을 가정하였다. 핵연료집합체의 전단력, 굽힘모우멘트, 변위 및 지지격자체의 충격하중에 대하여 자세히 고찰하였고 이들 동적해석 결과를 이용하여 핵연료집합체의 구조적 건전성을 평가하였으며 사고조건에서 2차측 배관파단이 핵연료집합체의 구조적 건전성에 미치는 영향을 검토하였다.

1. Introduction

The reactor core of a pressurized water reactor is composed of several hundreds of assemblies 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 the dynamic interactions through fluid coupling forces.

Safety qualification of the reactor core is one of

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the crucial issues in the design of a pressurized water reactor. The structural integrity of the fuel assemblies and the control rod insertion capabilities should be secured against the design loads.

For the purpose of assessing safety of the reactor core, many hypothetical accriduents are postulated and the dynamic response analyses to these accidents are performed. The strucural integrity of a fuel assembly should be assured for the worst loading conditions which is called a faulted condition. Under this condition, pipe break and safe shutdown earthquake(SSE) responses are combined. The pipe break loads are from either the main steam/feedwater pipe break, or loss of coolant accident (LOCA) loads whichever are larger. LOCA is defined as the loss of reacotr coolant at a rate in excess of the reactor coolant normal makeup rate, from breaks in the reactor coolant pressure boundary inside primary containment up to, and including, a break equivalent in size to the largest remaining primary branch line not eliminated by leak-before-break criteria.

The pipe break load considered in this paper is one of the faulted condition loadings. During and after pipe break, it is imperative that the structural integrity should be assured, i.e., stress of a fuel assembly must be kept within allowables defined in the regulatory code.

In the present study, peak responses are obtained for the motions induced from pipe break excitations using method for the detailed dynamic anlysis of a reactor core[1]. Also, the stress analysis is performed and the effect of the secondary side pipe breaks on the fuel assembly structural integrity under the faulted condition is investigated.

2. Pipe Breaks

Nuclear power plants have been designed for the postulated pipe breaks since the earliest plants. The first reason for postulating a break was to calculate the design basis containment pressure. The next reason was to consider the effect of the loss of coolant in the design basis for the emergency core cooling systems. These considerations are relatively independent of the mechanical details at a given location of postulated pipe break, and governed primarily by thermal-hydraulic system parameters.

More recent considerations of the mechanical and structural consequences of postulated pipe ruptures, including thrust forces on the piping, jet impingement on the surrounding compartment, internal hydraulic loads and sub-compartment pressurization, require more precise definition of the mechanical details at a postulated pipe break location.

It is emphasized that catastrophic pipe breaks which result in double ended guillotine break (DEGB) are highly improbable, but are postulated to establish a highly conservative design basis. The stringent quality assurance provisions imposed on the design, the quality control provisions in the manufacturing process[2] and the inservice inspection procedures employed on site[3] provide a very high level of assurance that catastrophic pipe breaks will not occur in nuclear class 1 piping[4]. To balance these considerations against the consequences of a postulated accident, even though remote, the United States Nuclear Regulatory Commission (USNRC) and the nuclear industry standards groups have established a reasonable yet conservative pipe break philosophy. The basic philosophy predicts that the location of a pipe break should be postulated at the point of the highest stress range or cumulative usage factor.

Prior to 1983, General Design Criteria 4(GDC-4) of 10CFR 50 Appendix A[5] required plant designers to consider the dynamic effects of postulated main coolant loop (MCL) breaks as well as tributary pipe breaks in mechanical design. Pipe break requirements for mechanical design have evolved to a more reasonable technical basis than the full double ended guillotine break originally required by GDC-4. Probabilistic and deterministic studies performed in the 1980's under USNRC sponsorship demonstrated that the probability of leakage, especially a DEGB is very low and that flaws in pipes can be detected before the flaws can grow to a critical length from which a DEGB could occur. The deterministic studies utilized a fracture mechanics technology now termed leak-before-break(LBB). The development of LBB methodology culminated in NUREG-1061 Volume 3[4], in which the USNRC established guidelines for application of LBB.

In parallel, regulatory requirements evolved to the 1986 "limited scope" rule of GDC-4, allowing the application of LBB techniques to demonstrate that consideration of MCL breaks in pressurized water reactors could be eliminated, and to the 1987 "broad scope" rule of GDC-4, allowing the LBB approach to extend, when justified, to all high energy piping systems in nuclear power plants. The USNRC Standard Review Plan (SRP) Section 3.6.3 implements the "broad scope" rule of GDC-4 and endorses the LBB methodology of NUREG-1061 Volume 3.

In 1983, Combustion Engineering performed LBB evaluation for the MCL and submitted a Safety Analysis Report(CESSAR), which was later accepted by the USNRC. MCL pipe breaks are not design basis for the nuclear power plant which is being constructed in Korea because those piping system is virtually the same as CESSAR plants'. Instead two inlet(14 inch safety injection nozzle and 3 inch pressurized spay line nozzle) and two outlet (16 inch shutdown cooling nozzle and 12 inch surge line nozzle) breaks in the primary side were postulated for the branch line pipe breaks. Of these four breaks, LBB evaluation was performed for the piping system with a diameter of 10 inches or over. Once elimination of those piping systems from the design consideration is accepted based upon current LBB evaluation, only the 3 inch 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[6]. 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. Analysis

3.1 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 analysis of reactor coolant system in which a very simplified model of the reactor vessel internals and core is used. Subsequently, the reactor vessel motion is used as 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[9]. 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 is modeled individually. In the last step, the deflected shapes of the fuel assembly and spacer grid impact loads from the detailed core anlysis are used to assure the struc-

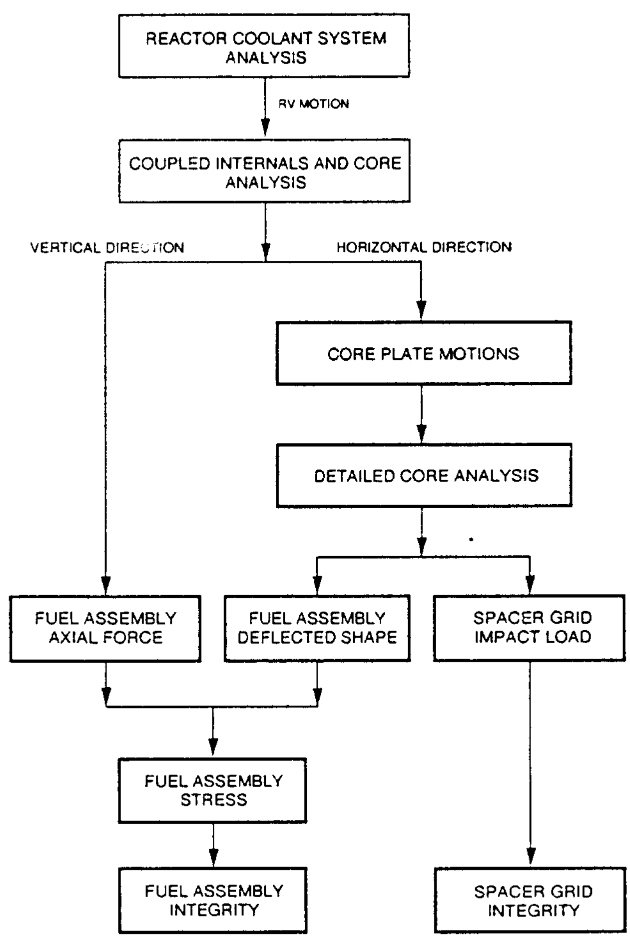

Fig. 1 Analysis procedure for fuel assembly design

tural integrity. Fig. 1 illustrates the overall fuel assembly design analysis flow.

The detailed horizontal core model is developed for the time history analysis for the pipe break 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 required[10].

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 a pipe break analysis of a coupled internals and core model which has a much less detailed representation of the core.

The reactor vessel motions for one of nuclear power plants in Korea as shown in Figs. 2 and 3 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. 4 and 5.

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[1].

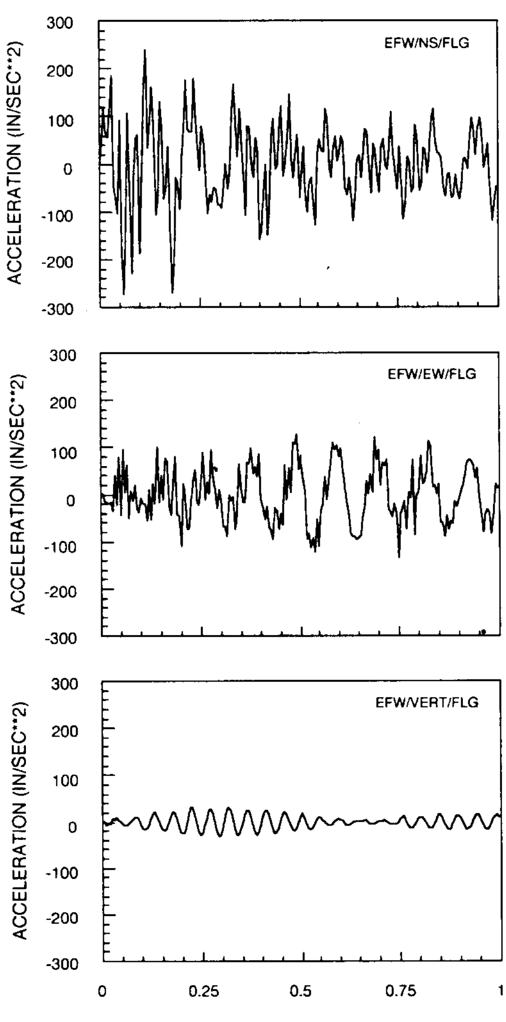

Fig. 2 Acceleration time history of RV flange for economizer feelwater line break

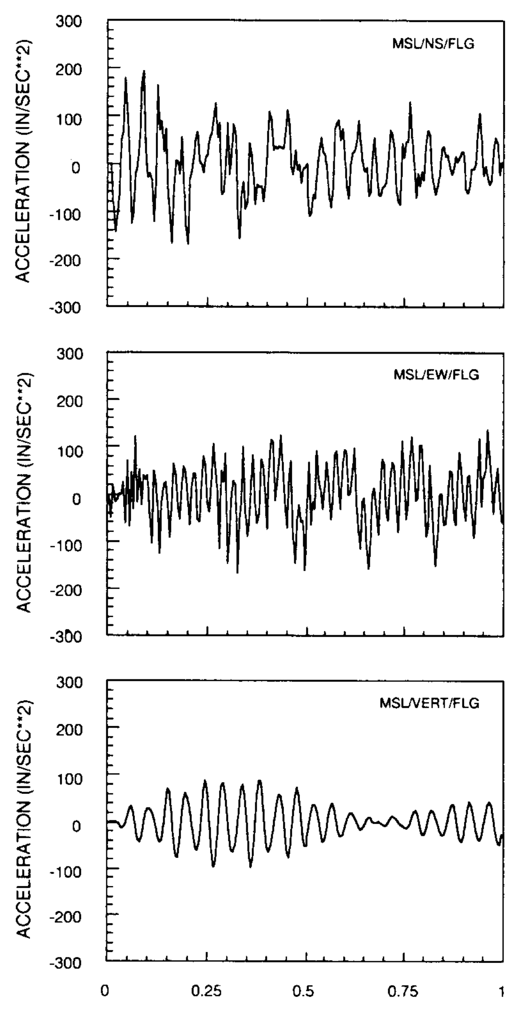

Fig. 3 Acceleration time history of RV flange for main steam line break

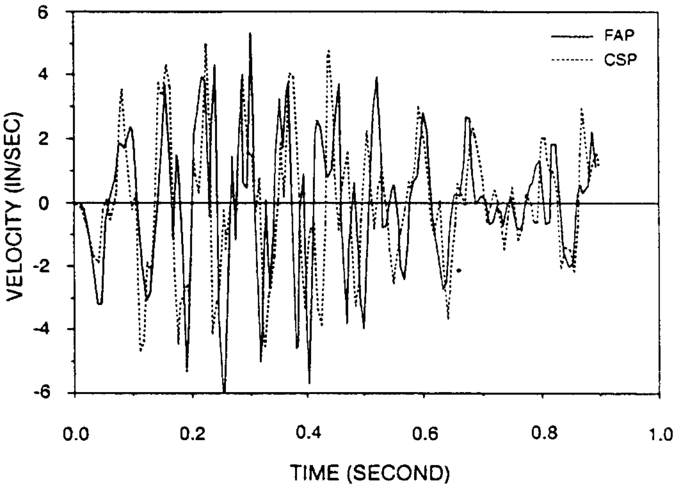

Fig. 4 Velocity time history of FAP and CSP for economizer feedwater line break

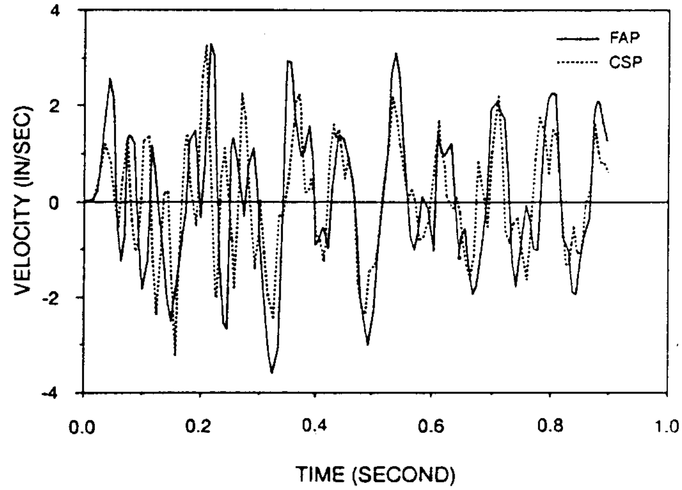

Fig. 5 Velocity time history of FAP and CSP for main steam line break

3.2 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 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[7]:

$$\frac{d^3v}{dx^3} = \frac{-p}{EI} \tag{1}$$

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

$$v(0) = v_1 \tag{2a}$$

$$\frac{dv}{dx}(0) = \tan\theta_1 \tag{2b}$$

$$\frac{dv}{dx}(L) = \tan\theta_2 \tag{2c}$$

where θ_1 and θ_2 are shown in Fig. 6.

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

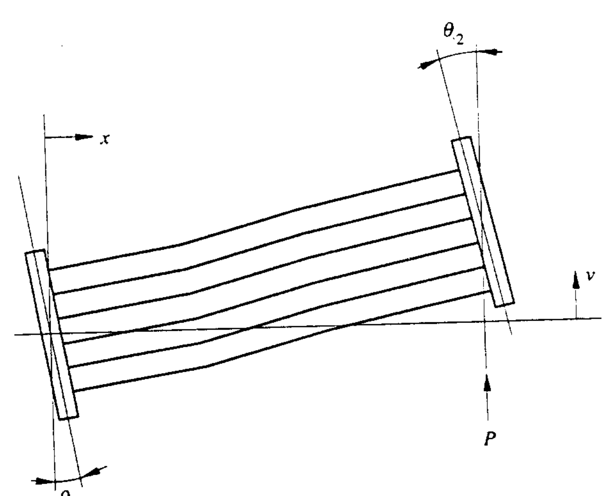

Fig. 6 Typical element under load

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

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

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

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

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

From this equation the net load applied to one guide tube or fuel oad 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

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

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. 7) 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[1].

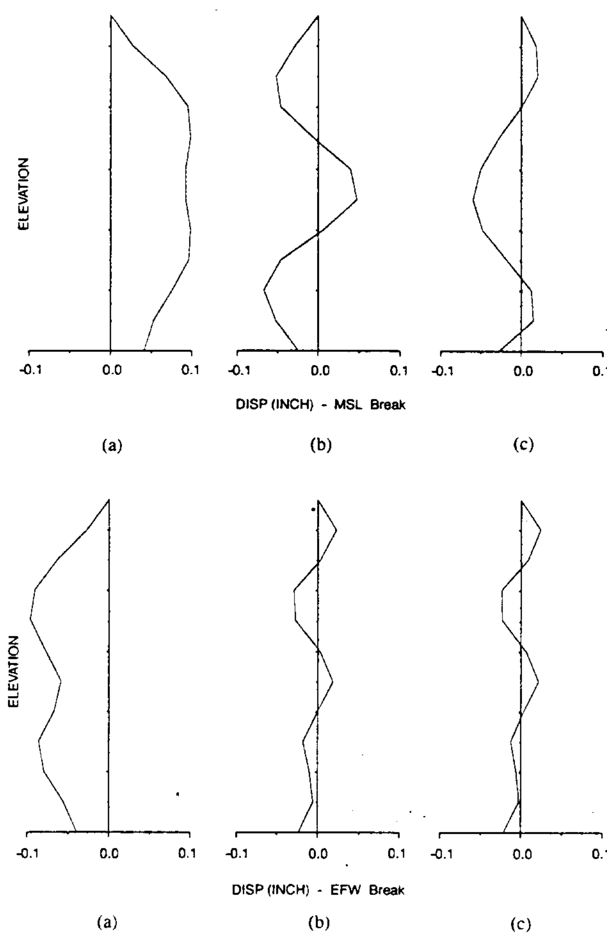

Fig. 7 Deflected shapes of fuel assembly for peak loading conditions : (a) peak displacement, (b) peak moment, (c) peak shear

Table 1 Responses of fuel assembly for pipe break excitations

Component	MSL	EFW	SSE[8]
Spacer grid			
One-sided impact (lbs)	0	0	2801
Through-grid impact (lbs)	0	0	2091
Fuel assembly			
Deflection (inch)	.097	.098	1.596
Shear (lbs)	1481	986	394
Moment (lb-inch)	101	58	7560
Axial force (lbs)[10]			
@ Fuel rod	106	84	506
@ Guide tube	9	7	45

Table 2 Stress intensities of fuel assembly for pipe break excitation

Component	Pipe Break	SSE[8]	Allowable*
Fuel rod(ksi)**			
Primary membrane + bending	21.6	25.8	31.8
Primary membrane only	21.6	21.6	21.7
Guide tube(ksi)			
Primary membrane + bending	3.2	35.1	61.9
Primary membrane only	0.7	7.6	44.7

* Allowables are for the faulted condition where pipe break and SSE responses are combined.

** Includes stress components due to fuel rod differential pressure.

The spacer grid impact loads and the fuel assembly responses are shown in Table 1. No spacer grid impact loads exist for the pipe 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[1]. Therefore only the impact loads due to earthquake excitation will be used to evaluate spacer grid integrity. For the axial response of a fuel assembly, the axial force of fuel rods is 106 1bs and 84 1bs for MSL and EFW breaks, respectively.

The required data for stress calculation of the fuel assembly is a deflected shape which represents the most severe 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 excitation is almost negligible. The stress intensities from other loadings are considered to be compared with code allowables to verify the structural integrity.

The dynamic responses such as fuel assembly shear force, bending moment, axial force and displacement, and spacer grid impact loads are obtained for the motions induced from pipe break excitations in the secondary side 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 are compared with those of earthquake and are found to be so small. The stress comparions between pipe break and earthquake show that pipe break contributions are negligible for the faulted condition when leak-before-beak concept is applied to the primary side piping systems with a diameter of 10 inches 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.

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