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# SEISMIC STUDY OF IMPACT FORCES IN A REACTOR CORE DUE TO NATURAL EARTHQUAKES

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## 1 INTRODUCTION

The effect of a safe-shutdown earthquake on the structural adequacy of a 15x15 fuel bundle in a reactor core has been studied analytically. The study shows that a fuel bundle would safely withstand the effects of the seismic loadings. A suite of five natural earthquakes in three specific soil conditions was selected for the system model since a highly non-linear core response was expected. Seismic input was defined by time-history-displacement records at the upper and lower core supports. All frequency regions were excited by at least one of the records. Effects of the different inputs were studied to determine if an analysis time duration less than 28 seconds could be justified and if the number of cases could be reduced. A truncation study model consisting of a single fuel assembly non-linear impact spring model including a representation of the core shroud (baffle) was used to justify a 6.5 second interval for the final evaluation. Allowance was made for simulated impact with another assembly. Fuel assembly response in the core region is dependent on several critical parameters. Those parameters included the gap sizes between the core shroud (baffle) and the peripheral fuel assemblies, gaps between the various fuel assemblies, impact element stiffness values and the energy of the input transient.

Structural adequacy is demonstrated if the core region maintains a coolable geometry and the control blades can be inserted following the earthquake. Acceptance criteria of the various fuel assembly components were developed based, in general, on the rules set forth in the ASME Boiler and Pressure Vessel Code, Section III, Appendix F. Geometric evaluations were performed, in addition to the stress evaluations, demonstrating that the core region would maintain its integrity for the identified seismic events.

A flow diagram for the subject study is shown in Figure 1.

## 2 PHYSICAL DESCRIPTION OF SUBJECT SYSTEM

The system analyzed is a part of the Combustion Engineering, Inc. Nuclear Steam Supply System (NSSS) shown in the report (Wesley and Salmon, 1989). A coupled model of the building and the NSSS was used to generate system-time-history responses.

The NSSS consists of a reactor pressure vessel (RPV), two steam generators, coolant pumps, and a pressurizer with piping and supports. The vessel is supported by two inlet and one outlet nozzles and contains the (RPV) internals including the core, core support barrel, upper guide structures, and core shroud and support assembly. The internals are supported both vertically and horizontally at the top by the core support barrel, and at the vessel/core support barrel flange near the nozzles. Lateral support is also provided by snubbers located between the bottom of the barrel and the reactor vessel. Seismic input is transferred from the free field to the building which excites the NSSS. Vessel motion is transmitted to the core support barrel and to the core region which is bounded on its top by the upper core alignment plate, on the bottom by the lower core support plate, and on the periphery by the core shroud. The core equivalent diameter is 3.5 m, and the height of the fuel assembly region is 3.8 m.

There are 204 fuel assemblies in the core, arranged symmetrically and aligned laterally in rows, the shortest of which is six assemblies, and the longest, 16 assemblies. Gaps separate the fuel assemblies. Gaps between the peripheral assemblies and the shroud are 3.6 mm, and gaps between assemblies range from 2.8 mm to 9.4 mm to allow for cruciform control blade insertion. A major non-linear aspect of the core is that its geometry permits contact between adjacent assemblies and/or between peripheral assemblies and the core shroud during a seismic event. A fuel assembly approximates a 210 mm square cross section beam 3.8 m long. It consists of a cage, fuel rods, lower tie plate, and upper tie plate. The cage consists of eight guide bars which hold ten spacer grids, welded at guide bar locations on each spacer grid. The fuel rods are frictionally supported in a 15x15 array with a length of 3.6 m. Eight of the peripheral locations are filled by guide bars. The assembly is held together by attachments to the upper and lower tie plates. Alignment holes in the fuel assembly upper and lower tie plates match alignment pins in the upper and lower core support plates. These connections limit lateral movement of the fuel assembly at the core plate locations.

## 3 DESCRIPTION OF APPLIED FORCES

A model of the building structure and the NSSS was subjected to a suite of five natural earthquakes to generate system time history responses. Seismic input is transferred, through soil structure interaction, from the free field to the building which excites the NSSS. The vessel motion is transmitted through its internal structure to the core support barrel and in turn to the core region. Response time histories from this system model were used as input forces to the core region. The final seismic input forces used in evaluating fuel system structural integrity were in the form of displacement time histories at the upper core support plate, the lower support plate and the core shroud (baffle) locations. Consistent transients were used for both the lateral and vertical evaluations.

Three soil conditions--soft, medium, and stiff--were evaluated for each of the five natural earthquake records considered--1954 Taft S69E, 1984 Coyote Lake S15W, 1971 Castaic N67W, 1979 El Centro Array #4 N00E, and 1979 El Centro S40E. The earthquake records were scaled to 0.2 g peak horizontal ground acceleration, corresponding to a safe shutdown earthquake. Each analysis was performed for a duration of 28 seconds to assure capture of the strong motion of each record.

Peak predicted lateral displacement of the reactor internals occurred for the Coyote Lake record in the soft soil condition. A plot of the upper bound horizontal response spectra at the approximate reactor pressure vessel support elevation, is presented in Figure 2. The first calculated fuel assembly in-reactor lateral response frequency is 2.93 Hz. Minimal dynamic amplification occurs at this first mode frequency. A review of these data indicates that the larger input displacement soft soil condition yields the worst case core region response. The core support plate inputs were taken directly from the system analysis linear beam model of the report (Wesley and Salmon, 1989).

## 4 TRUNCATION CRITERIA

Effects of the fifteen input transients were studied to see if an analysis-time duration less than 28 seconds was justified, and if the number of input cases could be reduced. The study used the single fuel assembly non-linear, eight impact spring model shown in Figure 4a, which included a representation of the core shroud. Allowance was made for possible impact between assemblies. The described non-linear model was subjected to earthquake strong motion for all 15 transient conditions using 12-second intervals. Peak impact force magnitude, location, and time of occurrence, were recorded and evaluated. Significant impact forces were obtained only for the Coyote Lake and the Castaic soft soil condition transients. The Coyote Lake soft soil input case was selected as the transient for use in the final evaluation; see Figure 4.

The time interval was selected by inspecting the displacement-time histories, i.e. the time interval selected for the Coyote Lake soft soil record was 6.25 seconds (Figure 4).

Shock spectra were plotted for both a 6.25 second interval and the entire 28 second earthquake Coyote Lake soft soil record. The shock spectra were essentially identical for the 6.25 and 28 second records. A 6.25 second interval was used in the final evaluation.

## 5 CORE REGION MODELS

The major purpose for developing the core region lateral models was to assess the condition of the fuel assemblies, contacting the core shroud (baffle) and other fuel assemblies. The vertical model of the assembly was developed to evaluate lift-off from the lower core plate and subsequent impact, along with fuel rod slippage. These contact and friction forces required that a fuel assembly non-linear models be used to monitor these forces and their effect on the fuel assembly overall response. Both the vertical and lateral models were modified to account for the in-reactor conditions of temperature and the fact that the assemblies were submerged in water during the transient. The multi-assembly lateral model incorporates several simplifying assumptions. The primary assumption is that the motion of the core is planar in response to seismic excitation and that this core response can be represented by a single row of fuel assemblies. It is further assumed that the effect of torsional motion of the fuel assemblies, barrel shell displacements and gap tolerances are negligible when determining overall core region lateral response. A more complete listing of fuel system analysis assumptions is presented as part of a US NRC study (Grubb, 1978).

The assembly model had to be as simple as possible, yet yield the appropriate response as indicated by full scale assembly testing. Therefore, several fuel assembly and core region models were required. The various models and their uses in the core region evaluation are described below:

1. A detailed single fuel assembly linear finite element model was used for the development of component loads and stress analysis (Hill and Patterson, 1982).
2. A single fuel assembly simplified modal synthesis (MacNeal, 1969) model was derived from the detailed model for use in critical parameter test verification and applied transient selection.
3. Two single fuel assembly non-linear models were used for model test verification and applied transient selection, Figure 3a.
4. A six-fuel assembly, non-linear model assembly was used to evaluate the fuel system mechanical response, Figure 3b.
5. A sixteen-fuel assembly, non-linear model was used to evaluate the fuel system mechanical response.
6. A single assembly non-linear fuel slip vertical model was used to evaluate assembly response if the vertical loads were large enough to cause rod slippage. This model allows for assembly lift-off.

## 6 DIRECT TRANSIENT ANALYSIS PROCEDURE

The final set of equations of motion in physical coordinates for the assembly models, which includes the spacer grid non-linear force is:

$$[M]\{\ddot{x}\} + [B]\{\dot{x}\} + [K]\{x\} = \{P_1(t)\} + \{P_n\}$$

where: $[M]$ = mass matrix, $[B]$ = damping matrix, $[K]$ = stiffness matrix
$\{x\}$ = a set of physical coordinates expressed as a column matrix of one term for each node point
$\{P_1(t)\}$ = the time-dependent applied force vector
$\{P_n\}$ = a vector of non-linear forces, functions of $\{x\}$ and $\{\dot{x}\}$, used to model the gaps and one-way impact damping

The Direct Transient Analysis capability of MSC/NASTRAN was used to solve the equations of motion. The applied non-linear forces, $P_n$, are evaluated with this option as the variations from a linear response for the spacer grid elements. This solution technique eliminates the traditional approach of reformulating the stiffness matrix for each successive time step since it uses the Newmark-Beta, (MacNeal, 1969) direct integration procedure. Analysis convergence and stability were checked by altering the time step of $\Delta t = 0.1$ ms $\pm$ 25%.

## 7 CORE REGION ANALYSIS RESULTS

Impact loads, deflections, and component loads were determined from core region and axial and lateral analyses. Detailed component stress evaluations were performed. Spacer grid impact loads and fuel assembly deflections represent the primary lateral analysis results. Impact loads occurred on both sides of the core region. A peak load of 1,134 newtons occurred at the spacer grid near the center of the left side peripheral assembly due to impact at the core shroud. A peak fuel assembly deflection of 19.8 mm was calculated for the assembly near the center of the core. This deflection value was used for cage stress evaluations. Impact forces and deflections determined for the sixteen-assembly are essentially the same as those presented for the six-assembly analysis. This is due to the fact that only the peripheral assemblies demonstrate contact and both the six and sixteen assembly models are identical in this region.

## 8 STRUCTURAL INTEGRITY AND FUNCTIONALITY

The criterion is to maintain a post-earthquake core geometry such that control blades could be inserted and the core cooled. Control blade insertion is assured if the fuel assemblies maintain their original positions, and no plastic deformation or geometric instabilities occur. Coolable geometry is maintained if insertability remains feasible and the spacer grids maintain the original fuel rod spacing. Component stress criteria were based on the ASME Boiler and Pressure Vessel Code, Section III, Appendix F, for component supports and on the USNRC Standard Review Plan. Additional assessments were made to assure that the core region would maintain mechanical function and coolability sufficient to enable control blade insertion and assembly lift-off. The stress limits were applied to fuel rods, guide bars, tie plates, and guide-bar to tie-plate cap screws. Spacer grids were evaluated to allowable load limits based on tests in which the grids were loaded to the point of permanent deformation. The allowable load was taken as the 95% confidence value of the mean of the test data for in-reactor temperature conditions.

Based on the results of the lateral and vertical seismic analyses, maximum stresses in the fuel assembly components were calculated in accordance with the stress criteria. All components met the stress or load criteria with adequate safety margins. The maximum vertical relative displacement of the fuel assembly from the lower core plate was calculated and compared to the length of the alignment pins, verifying that lateral support would be maintained, and that the assembly would return to its original position following the seismic event with no permanent deformation. Based on these findings, it is concluded that control blade insertability would be assured and that a coolable geometry would be maintained.

## 9 CONCLUSIONS

A cost effective evaluation, (which was found acceptable for this application by the US Nuclear Regulatory Commission) which includes non-linear analyses and a methodology for reducing the number of earthquakes and the analysis time interval for those earthquakes, has been completed. It has been demonstrated that the Siemens Power Corporation's High Thermal Performance Fuel Assemblies will maintain their structural integrity and functionality when subjected to a safe shutdown earthquake.

## REFERENCES

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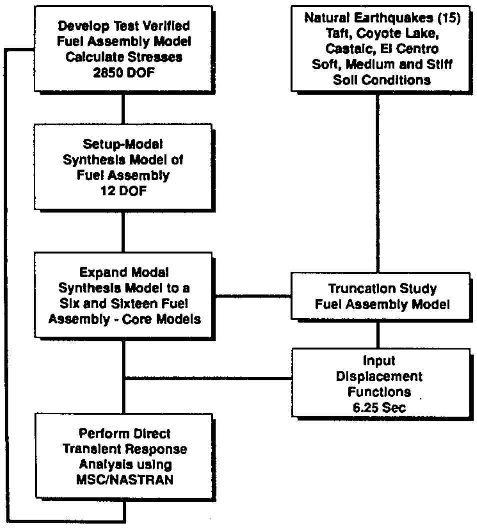


**Figure 1 Flow Chart - Dynamic Analysis**

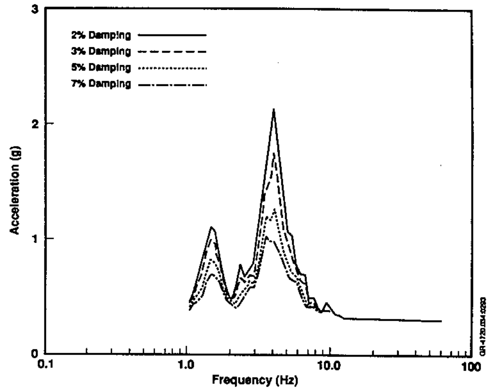


**Figure 2 Bounding Spectral Acceleration Soft Soil Case**

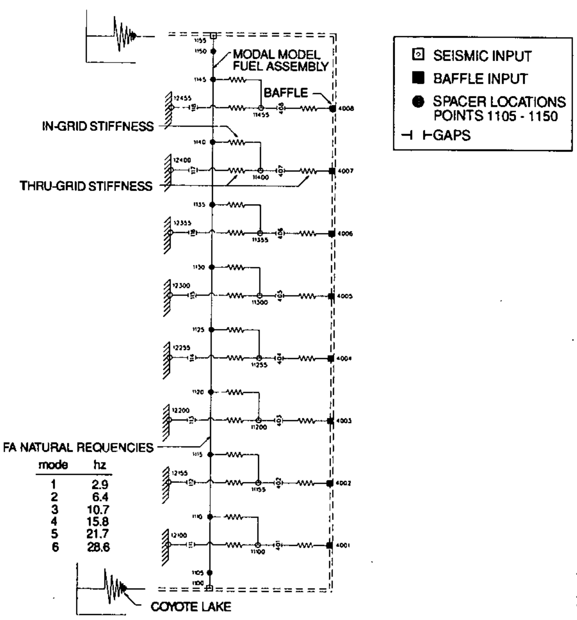


**Figure 3a Truncation FA Model**

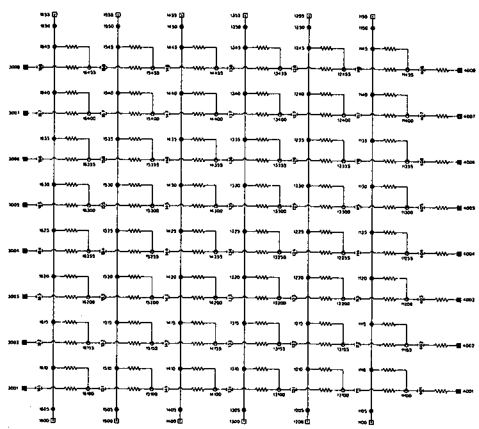

**Figure 3b Six FA Model**

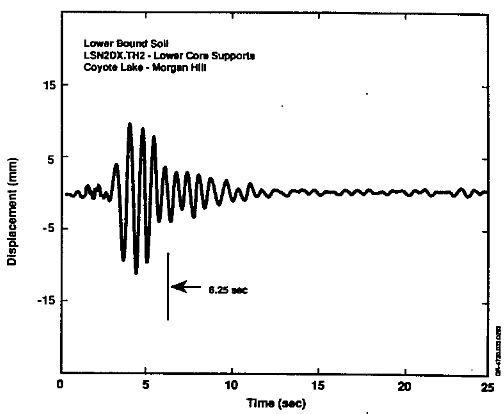


**Figure 4 Final - Input Time History**