Westinghouse Non-Proprietary Class 3

WCAP - 15246

Control Rod Insertion Following a Cold Leg LBLOCA, D. C. Cook, Units 1 and 2

Westinghouse Electric Company LLC

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TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

LIST OF FIGURES (cont.)

1 INTRODUCTION

Analyses have been performed for D. C. Cook, Units 1 & 2 to demonstrate that control rods will be inserted following the large cold leg Loss of Coolant Accident (LOCA) and that the negative reactivity credit can be applied in evaluating recriticality at the time of switch over to hot leg ECCS recirculation. Calculations were performed for both the Design Basis breaks and limiting breaks defined by application of the Leak-Before-Break (LBB) criteria. Loads on distortion sensitive components (RCCA Upper Internals Guide Tubes and Fuel Assembly Grids) which influence control rod insertion times were determined to be within allowable limits and no significant degradation of insertion times is indicated. This result applies to both design basis breaks and limiting breaks defined by application of the LBB criteria.

2 BACKGROUND

In June of 1992, the NRC identified concerns for possible core recriticality following a large break LOCA and notified the Westinghouse Owners Group of these concerns through SECY-92-208. Westinghouse routinely performs a check on sump boron level to preclude recriticality at the start of injection from the sump. However, in response to SECY-92-208, later portions of the transient were reviewed with respect to the boron concentration in the injected ECCS liquid. It was determined, that for large cold leg breaks, the buildup of high boron concentrations in the core could result in significant boron dilution in the sump and a potential recriticality condition could develop at the time of switchover of the ECCS to hot leg recirculation. In this scenario, the buildup of boron in the vessel causes an associated boron dilution in the sump and at the time of switchover to hot leg recirculation, insufficiently borated ECCS fluid could be introduced to the top of the core. This liquid was conservatively assumed to displace the more highly borated liquid and potentially lead to a core recriticality.

In response to a desire to simplify plant operation and to address this issue, the Westinghouse Owners Group (WOG) contracted Westinghouse to perform an ECCS Hot Leg Recirculation Elimination program. The objective of the program was to justify the elimination of ECCS hot leg recirculation following a LOCA event. The resulting analysis, documented in WCAP-14486, modeled RCS flow through the gap between the barrel and vessel at the hot leg nozzle location to prevent boron precipitation in the core following a LOCA event. The benefits of eliminating hot leg recirculation include elimination/relaxation of surveillance, tests and maintenance of the ECCS hot leg recirculation components, elimination of operator training to perform the switchover operation for hot leg recirculation, and increased reliability of the ECCS recirculation function following a LOCA event. Additionally, the safety issue of core recriticality at switchover of ECCS to hot leg recirculation would be resolved through this program by modifying EOPs to eliminate ECCS hot leg recirculation.

WCAP-14486, "ECCS Hot Leg Recirculation Elimination for Westinghouse 3 and 4 Loop Design NSSS," was submitted to the NRC for review in July 1996 (Ref. OG-96-054). The NRC has notified Westinghouse that during their review of WCAP-14486, they noted that the topical report did not address plugging of the hot leg nozzle gap with debris from the sump nor did the report address the risk impact of hot leg switchover elimination. The NRC also identified a number of other technical concerns related to WCAP-14486. After review of the WCAP by Westinghouse, it was concluded that the debris issue in the sump and potential for closing of the gap related to variability in long term vessel and barrel temperatures, made defense of the methodology impractical. Westinghouse recommended that the WCAP be withdrawn, the WOG concurred, and the WCAP was withdrawn in October 1998.

D.C. Cook Units 1 and 2 have used WCAP-14486 to address Westinghouse Nuclear Safety Advisory Letter (NSAL) 94-016, "Core Recriticality During LOCA Hot Leg Recirculation". NSAL 94-016 does provide an alternate approach to resolve this issue which is to demonstrate the plant is still within its licensing basis by taking credit for design margins. However, Units 1 and 2 did not appear to have sufficient margin and Westinghouse compiled a list of alternative approaches.

In addition to taking credit for existing plant margins and refinements in methodology for calculating those margins, the most promising alternative was demonstration of control rod insertion for Design Basis breaks or limiting cold leg breaks as defined by LBB methodology. Taking reactivity credit for control rod insertion for cold leg breaks, is feasible since the recriticality at hot leg switchover issue is a concern for cold leg breaks but not hot leg breaks. In the case of hot leg breaks, sump dilution during cold leg injection is precluded by spilling of core flow from the break. Control rod insertion for large break LOCA has been identified as a problem for hot leg breaks due to the proximity of the guide tubes in the upper plenum to the break. However, it was considered highly probable that control rod insertion could be demonstrated for cold leg breaks which would result in negative reactivity benefits on the order of 400 ppm or more (boron equivalent rod worth), depending upon plant type and design. This negative reactivity would be available to prevent re-criticality at the time of hot leg switchover. The review of the post-LOCA reactivity margins of D.C. Cook Units 1 and 2 has shown that it is difficult in some cases to demonstrate that sufficient reactivity margin is available at the start new fuel cycles. Consequently an analytical activity was initiated to demonstrate that credit could be taken for control rod insertion for the cold leg LBLOCA scenario.

3 ANALYSES DESCRIPTION

3.1 OBJECTIVE

Demonstrate through analysis and application of Leak-Before-Break criteria breaks and/or through use of Design Basis breaks that the control rods will be inserted at D.C. Cook, Units 1 and 2 under post-cold leg LBLOCA conditions.

3.2 BREAKS/LOCATIONS

The rod insertion analysis included a set of breaks which were selected based on standard Westinghouse analysis practice and the knowledge from past calculations that satisfactory results would be unlikely for the double ended guillotine hot leg breaks. The latter break was not evaluated since HLSO recriticality is only a cold leg break issue. The following breaks were analyzed.

60 in^2 Accumulator Line Break

98 in^2 Pressurizer Surge Line Break

144 in^2 Reactor Vessel Inlet Nozzle (RVIN) Break

144 in^2 Reactor Vessel Outlet Nozzle (RVON) Break

594 in^2 Reactor Coolant Pump Outlet Nozzle (RCPON) D. E. Guillotine Break

The first two breaks are the limiting break sizes and locations which result from application of the Leak-Before-Break criteria to the D. C. Cook Units. This criteria, as applied to D. C. Cook, eliminates consideration of breaks in the main RCS piping. Thus, the most limiting LBB breaks occur in branch lines and are breaks in the accumulator line and the pressurizer surge line adjacent to the main RCS piping. The 144 in^2 breaks at the reactor vessel inlet and outlet nozzles are Design Basis bounding generic breaks for which the break area is limited by the biological shield concrete and the piping supports. The 594 in^2 RCPON double ended guillotine break is the maximum area cold leg Design Basis break.

3.3 ACCEPTANCE CRITERIA

Control rod insertability is considered to have been demonstrated if the following criteria are met.

1. RCCA Upper Internals Guide Tube loads calculated for LOCA and the Safe Shutdown Earthquake (SSE) shall be less than design allowable values which have been shown to allow control rod insertion. Design values have been experimentally established for 150" 15x15 and 150" 17x17 upper internals guide tubes. (References 6.1 and 6.2)

2. No fuel assembly grid distortion shall be calculated to occur in fuel assemblies located beneath RCCA locations. Fuel assembly limits are based on fuel assembly specific, experimental grid load / distortion data.

LOCA and seismic loads shall be combined using the Square Root Sum of the Squares (SRSS) method to determine the loads for comparison to the above limits. LOCA loads shall be calculated for both design basis break areas and locations and also for limiting breaks as determined by application of Leak-Before-Break criteria.

4 LEAK-BEFORE-BREAK APPLICABILITY

On February 1, 1984, the NRC issued a Safety Evaluation Report, Reference 4.1, on References 4.2 and 4.3 which address the use of Leak-Before-Break (LBB) technology for eliminating double ended pipe ruptures of the main reactor coolant piping from the design basis of nuclear plants, as was defined in GDC-4. As a result of these studies performed for the Westinghouse Owners Group, double ended pipe ruptures of the RCS branch piping became the design basis for all plants qualified under the LBB program. D. C. Cook Units 1 and 2 are included in the qualified group. The limiting branch line breaks in the Westinghouse designed plants are now the Accumulator Line break in the cold leg and the Pressurizer Surge Line break in the hot leg. GDC-4 was subsequently modified in October 1987 to incorporate the provisions of LBB technology, Reference 4.4.

Since that time and based upon the guidance provided by the NRC in GDC-4, LBB based criteria have been incorporated as the design basis for a number of applications including:

Pipe whip restraint removal

Steam generator snubber reduction

Fuel assembly mechanical design

Selected reactor internals analyses

Of particular significance is the use of LBB in fuel assembly mechanical design. This approach has been applied to all new fuel designs since the mid '80s and has been described to the utilities in the Reload Transition Safety Reports (RTSRs) associated with the transition to VANTAGE 5 fuel. In the D. C. Cook Unit 2 transition to VANTAGE 5 fuel in the early '90s, the use of LBB for fuel structural evaluation was identified and documented in both the RTSR and reactor internals compatibility evaluation, Reference 7.1. In the SER for T/S Amendment No. 148 for Unit 1 and No. 134 for Unit 2, the NRC noted that core coolable geometry is maintained under design basis earthquake and asymmetric pipe rupture transients. In this statement, the application of asymmetric pipe rupture, i.e. LBB, is recognized. Thus, application of this methodology has been identified to the utilities and to the NRC.

This application of LBB to the fuel assembly design has been discussed with the NRC, most recently in a WOG / NRC meeting in 1993, as documented in Reference 4.5. At the time of the WOG / NRC meeting, the NRC indicated a concern for use of LBB for reactivity applications but took no action to discourage further consideration of this concept. The current proposed application of LBB to reactivity questions is only for cold leg breaks and is thus less aggressive than the previous proposal which included both hot and cold leg breaks. Approval does not provide the opportunity for RWST boron concentration reductions as did the previous proposed application.

The analysis results for D. C. Cook have shown that control rod insertion will occur for both the LBB criteria breaks and the Design Basis cold leg breaks. Thus, this report support two bases for licensing the application of control rod insertion for the large cold leg LOCA.

5 LOCA FORCES ANALYSES

The LOCA forces analysis was performed through use of the MULTIFLEX 3.0 Code which has been accepted by the NRC for the WOG Baffle-Barrel-Bolt Program, References 5.1 and 5.2, and incorporated the 4-loop vessel input model which was developed and validated on that program. The 4-loop input model was generally applicable to the D. C. Cook units since it was based upon an identical Westinghouse plant design. Where plant specific differences did exist, e.g. fuel type, vessel support stiffness, etc., the D. C. Cook values were applied. In contrast to the WOG Baffle-Barrel-Bolt Program an approved break opening time of 1 ms was used since NRC concurrence on the use of longer times has not been obtained for general application, Reference 5.3.

LOCA forces calculations were performed for the five breaks indicated in Section 3. Output of the calculations consisted of files of RCCA guide tube lateral forces and reactor vessel and internals horizontal and vertical forces. Identification numbers for these tape files are documented in Reference 5.4.

Typical results are provided in Figures 5.1 and 5.2 for the core barrel forces for the 144 in^2 Reactor Vessel Inlet Nozzle (RVIN) cold leg break and the accumulator line break. In these figures, the X direction refers to an axis coincident with the centerline of the broken leg which is always the direction of the maximum horizontal LOCA forces. Peak core barrel loads may be observed to occur in the first 0.04 seconds as do the upper internals guide tube forces and peak grid loads, shown in Sections 6 and 8. For the 144 in^2 Reactor Vessel Outlet Nozzle (RVON) hot leg break and the pressurizer surge line break, peak barrel forces occur somewhat later due to the transit time of the decompression waves in the piping, Figures 5.3 and 5.4. Conversely, the peak loads on the upper plenum guide tubes occur earlier and with a higher magnitude due to their proximity to the break. The peak core barrel load for the Reactor Coolant Pump Outlet Nozzle (RCPON) break is provided in Figure 5.5.

Flow loads and acoustic wave loads were calculated for the limiting upper internals guide tube location for both units. Flow loads and acoustic wave loads were provided for input to the structural analysis which is discussed in Section 6 and documented in Reference 5.4.

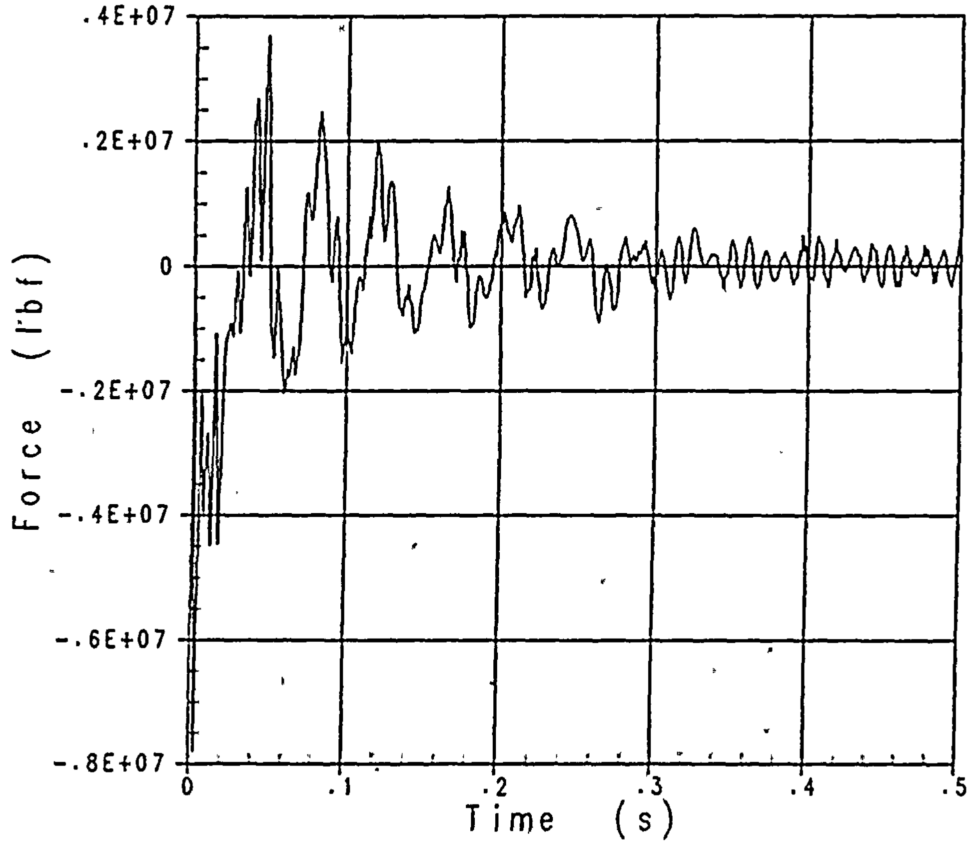


Figure 5.1 144 In^2 Reactor Vessel Inlet Nozzle Break Horizontal Forces on Core Barrel, X Direction

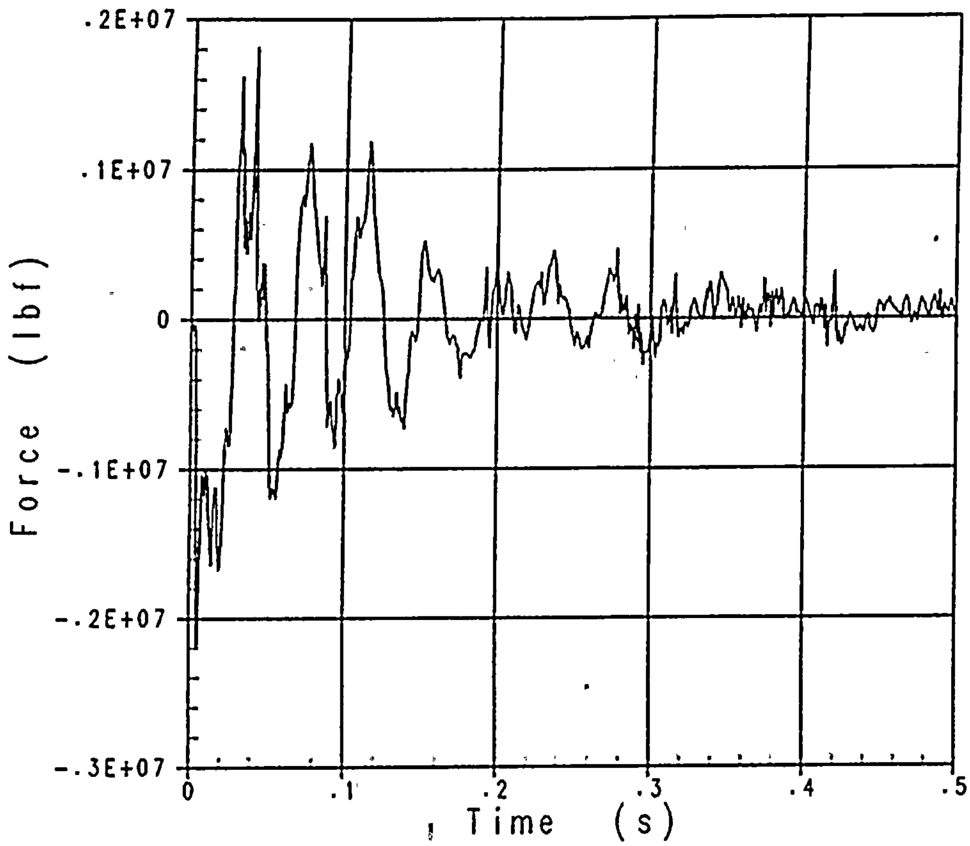


Figure 5.2 60 In² Accumulator Line Break Horizontal Forces on Core Barrel, X Direction

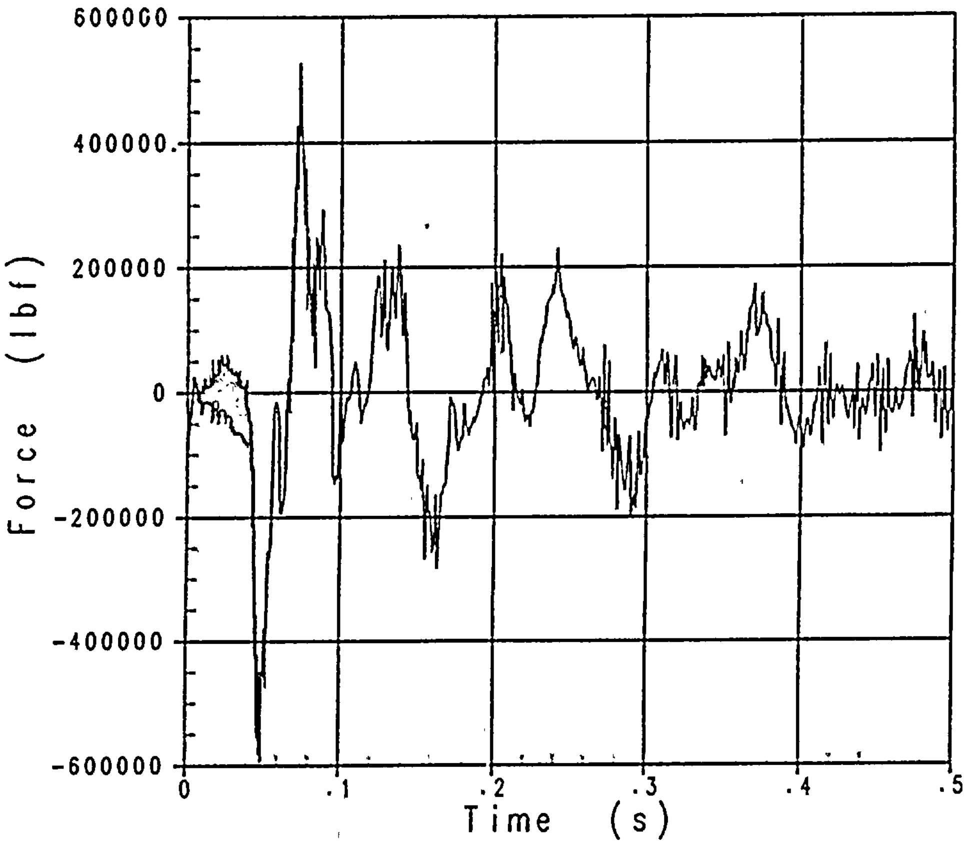


Figure 5.3 144 In² Reactor Vessel Outlet Nozzle Break Horizontal Forces on Core Barrel, X Direction

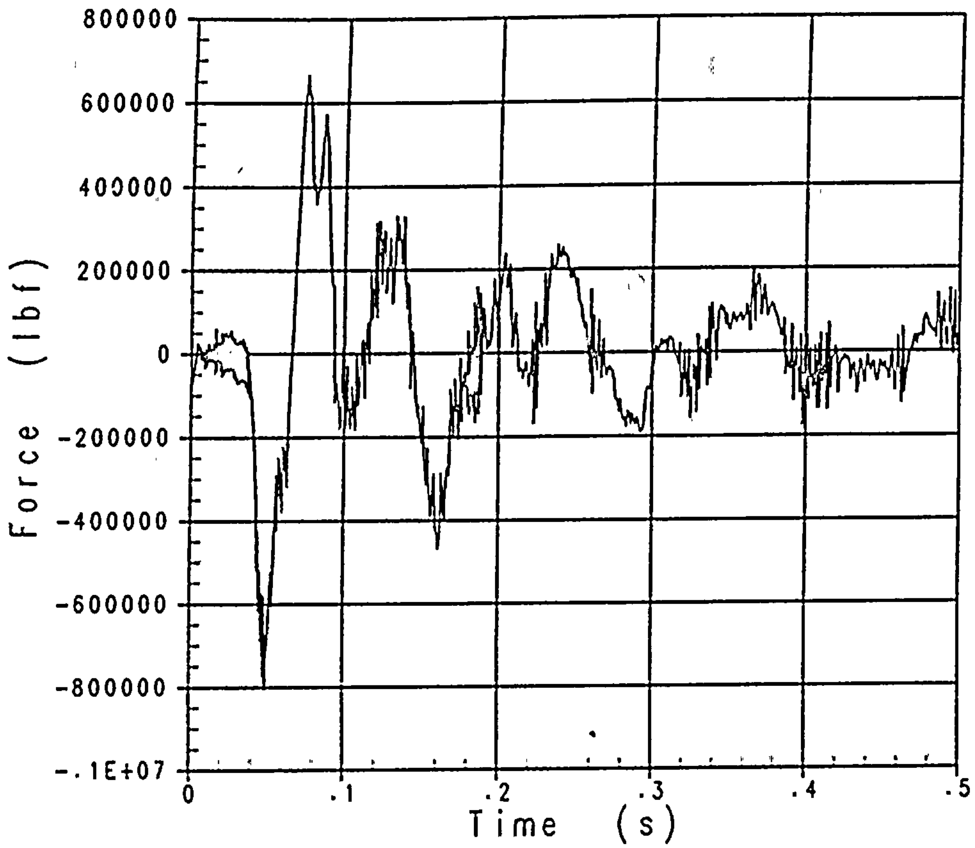


Figure 5.4 98 In^2 Pressurizer Surge Line Break Horizontal Forces on Core Barrel, X Direction

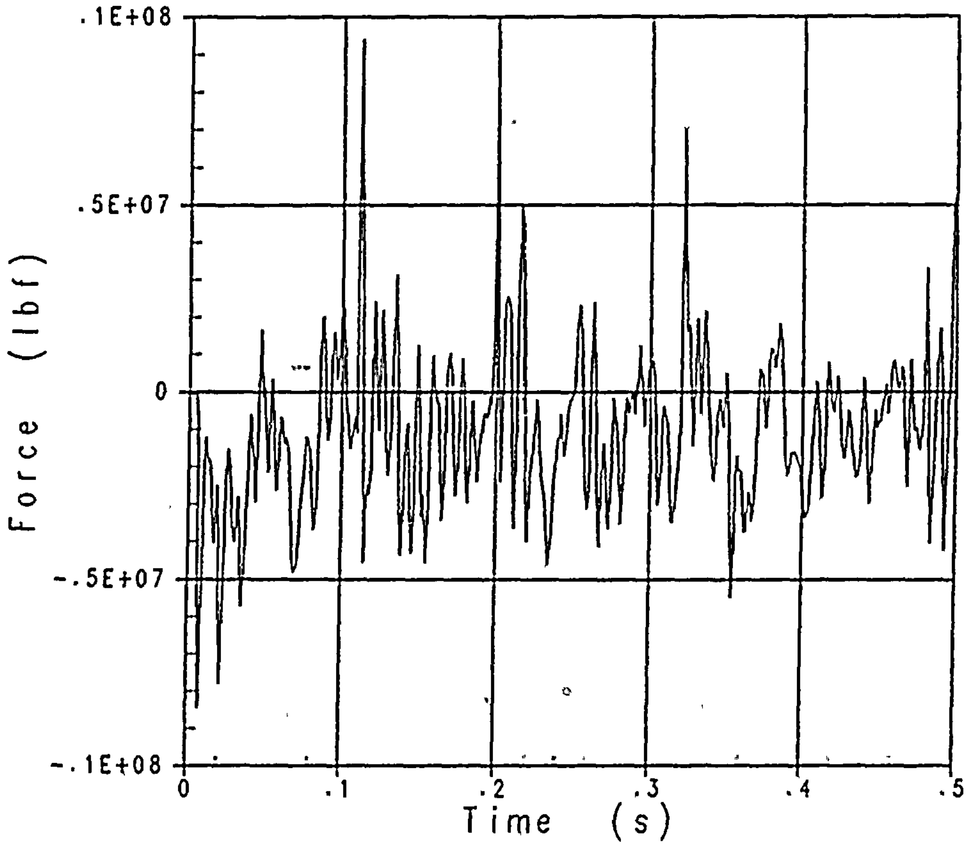


Figure 5.5 594 In² Reactor Coolant Pump Outlet Nozzle Break Horizontal Forces on Core Barrel, X Direction

6 RCCA UPPER INTERNALS GUIDE TUBE ANALYSIS

6.1 RCCA GUIDE TUBE LOADS

The total guide tube loading is a combination of seismic loads and a set of loads developed during the LOCA transient. Three separated LOCA loads are calculated and combined to determine the total LOCA contribution to the combined loading.

1. Hydraulic Cross Flow Loads (Drag Loads)

   These loads result from the effect of flow from the upper plenum toward the vessel outlet nozzle in the broken loop. This occurs for both cold leg and hot leg breaks, however, for the cold leg break it occurs later in the transient and has a significantly lower magnitude. Proprietary scale model and plant test measurements of guide tube strains coupled with hydraulic analyses of the upper plenum region and the LOCA blowdown force calculations form the basis for estimating these loads. A dynamic loading factor is applied which accounts for the natural frequency of the guide tubes and the time history variation of the crossflow loads,

2. System Loads (Inertial Accelerations)

   These loads result from the dynamic response of the reactor vessel and the internals to the vessel depressurization loads. They are calculated by using the LOCA force calculations from the MULTIFLEX combined with the WECAN structural model of the reactor system. These loads are not sensitive to guide tube location, but are sensitive to break area, break location, and break opening time.

3. Acoustic Loads (Pressure Gradient due to Decompression Wave)

   As the initial decompression wave from the break propagates through the upper plenum, differential pressure is applied to the guide tubes and lateral forces are developed. For the most highly loaded guide tubes near the vessel outlet nozzle, the acoustic load is a function of the maximum pressure differential, the effective guide tube area and a dynamic load factor.

6.2 GUIDE TUBE LOAD COMBINATION

The three LOCA guide tube loads originate from one MULTIFLEX Code calculation for each break considered and thus the time phasing of the loads are appropriate with respect to each other. Consequently, it is appropriate to linearly combine these loads as a function of time to obtain a total LOCA load transient. The peak LOCA load obtained in this fashion can then be combined using the Square Root Sum of the Squares method with the peak seismic load on the guide tube to obtain the total guide tube load.

6.3 ALLOWABLE LOADS FOR GUIDE TUBES

The control rod insertability is a function of the guide tube's deflection during a LOCA transient. As the amount of deflection increases, control rod insertion time will first be degraded and at sufficient deflection control rod insertion will be precluded. Since the guide tube is a rather complex structure and the motion of control rods are dependent on the amount of friction between the two components, it is difficult to determine control rod insertion though analytical means. For this reason, guide tube scram tests have been performed by Westinghouse in the past to experimentally determine the limits of control rod insertability. Guide tube scram tests have been performed on 96"-17x17, 150"-17x17, and 150"-15x15 guide tubes, References 6.1 and 6.2. Full size guide tubes, with rod control clusters, were mechanically loaded at discrete elevations to simulate flow loads experienced during a postulated LOCA transient. The insertability for the control rods as a function of the guide tube deflection, which in turn is a function of the applied mechanical loads, were recorded during the tests. The allowable load is then determined as the limiting applied mechanical load corresponding to the guide tube's permanent loss of function.

For example, Figure 6.1 which is from Reference 6.1, shows the results for one of the tests, in terms of the guide tube loads versus deflection for a 150"-17x17 guide tube. This graph determines the limits of guide tube deflection beyond which control rods can no longer be inserted.

6.4 GUIDE TUBE INSERTION RESULTS

The results of this analysis for Units 1 and 2 are provided in Table 6.1 for the five break sizes and locations considered in this study. The values reported are for the most highly loaded guide tube so that positive margin insures that all control rods will be inserted. It can be seen that substantial margin exists for all cases. The variation in margin between the two units is the direct result of the variations in the 15x15 and 17x17 guide tube structural design. The 17x17 guide tubes have a continuous enclosure from the upper core plate to the upper support plate and, thus, both a higher allowable load and higher natural frequencies which contribute to higher seismic loads. The minimum margin occurs for the Unit 1 Reactor Coolant Pump Outlet Nozzle break which has a margin of 24%. These margins insure insertion of all control rods. Thus, both LBB criteria breaks and Design Basis breaks in the cold leg are within allowable limits and pose no concern for control rod insertion at either of the D. C. Cook Units.

Table 6.1 RCCA Upper Internals Guide Tube Rod Insertion Margin

D. C. Cook Unit 1 (AEP) 150" - 15x15 Upper Internals Guide Tubes

Break Location/Size

	Acc. Line 60 in^2	Surge Line 98 in^2	RVIN 144 in^2	RVON 144 in^2	RCPON 594 in^2
LOCA Loads (lbs)					
SSE* Load (lbs)					
SRSS (lbs)					
Allowable Load (lbs)					
Margin**					

a,c

D. C. Cook Unit 2 (AMP) 150" - 17x17 Upper Internals Guide Tubes

Break Location/Size

	Acc. Line 60 in^2	Surge Line 98 in^2	RVIN 144 in^2	RVON 144 in^2	RCPON 594 in^2
LOCA Loads (lbs)					
SSE* Load (lbs)					
SRSS (lbs)					
Allowable Load (lbs)					
Margin**					

a,c

* SSE (Safe Shutdown Earthquake) is DBE (Design Basis Earthquake) for D. C. Cook

** Margin = $((F_{allowable} / F_{calculated}) - 1) \times 100\%$

a,c

Figure 6.1 Load vs. Deflection, 150" - 17x17 Guide Tube

7 REACTOR INTERNALS DISPLACEMENT ANALYSIS

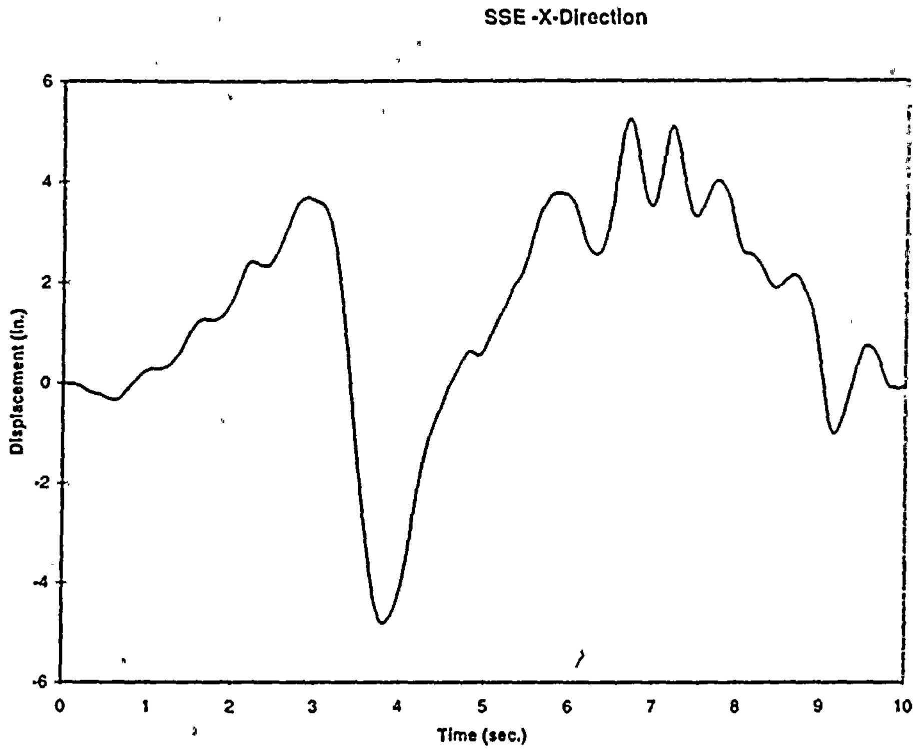
a,c

Analysis of the fuel assembly response to the combined LOCA and seismic loads requires as input, the upper core plate, lower core plate, and baffle assembly motions. These have been developed for the complete set of LOCA break sizes and locations indicated in Section 3 and the D. C. Cook plant specific seismic spectra. The methodology used to perform these calculations is identical to that used in the studies associated with the Unit 2 transition to Vantage 5 fuel in 1990, as documented in Reference 7.1.

Core plate and core barrel motion information, calculated with the Reactor Internals Model was provided to CNFD from WECAN Code output files as documented in Reference 7.2. Plots of Unit 1 seismic load induced core plate motions are provided in Figure 7.1 and the LOCA load induced core plate motions are provided in Figures 7.2 through 7.6. Similar plots for Unit 2 are provided in Figures 7.7 through 7.12. In these plots, the X and Z directions represent the coordinates which are parallel to the reactor vessel horizontal cardinal axes with Y being the vertical direction.

Seismic time histories are calculated from seismic response spectra for a ten second period to insure that the maximum motions and loads have been captured. As may be observed in the lower core plate motions, Figure 7.1 and 7.7, peak motions do not occur until several seconds into the transient. Note that in the seismic event, the reactor internals experience minimal relative motion and the core plates and core barrel move as a single unit with the same motion history. As may be seen for the 144 in^2 RVIN break Figures 7.2 and 7.8, the peak core plate motions due to LOCA loads occur within the first 0.10 seconds with a peak to peak upper core plate motion of 0.4 inches. The LOCA motions result in peak grid loads at approximately 0.05 seconds. To conservatively address the variation in time phasing of the loads, the effects of LOCA and seismic induced motion on grid impact forces are calculated separately and the peak forces are combined using the Square Root Sum of the Squares (SRSS) method.

Seismic motions of the core plates and core baffle assembly are provided to CNFD from WECAN Code output files as documented in Reference 7.2.



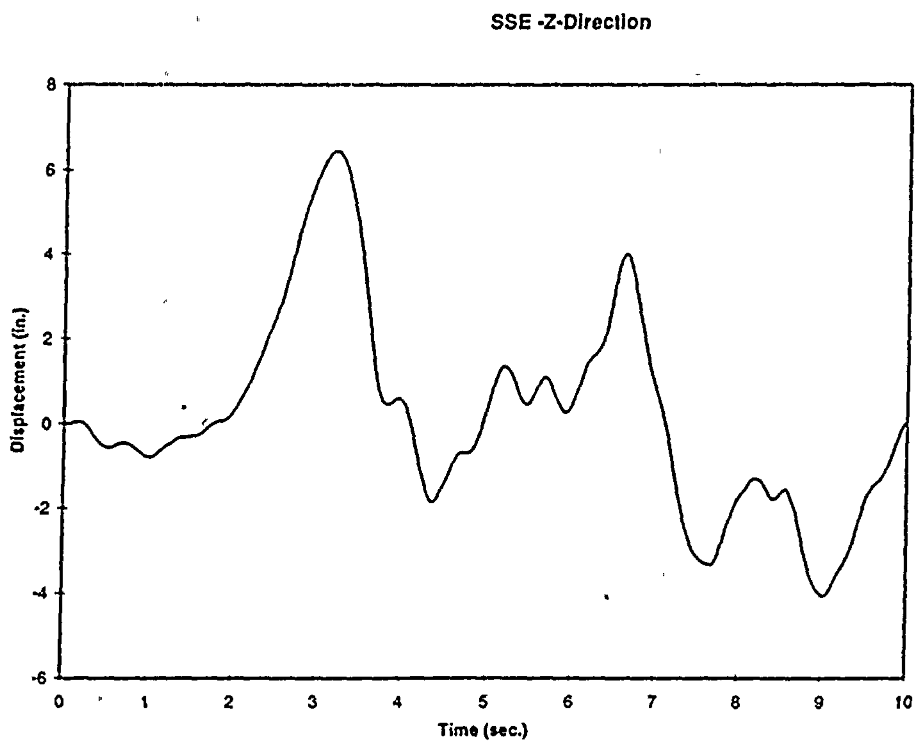
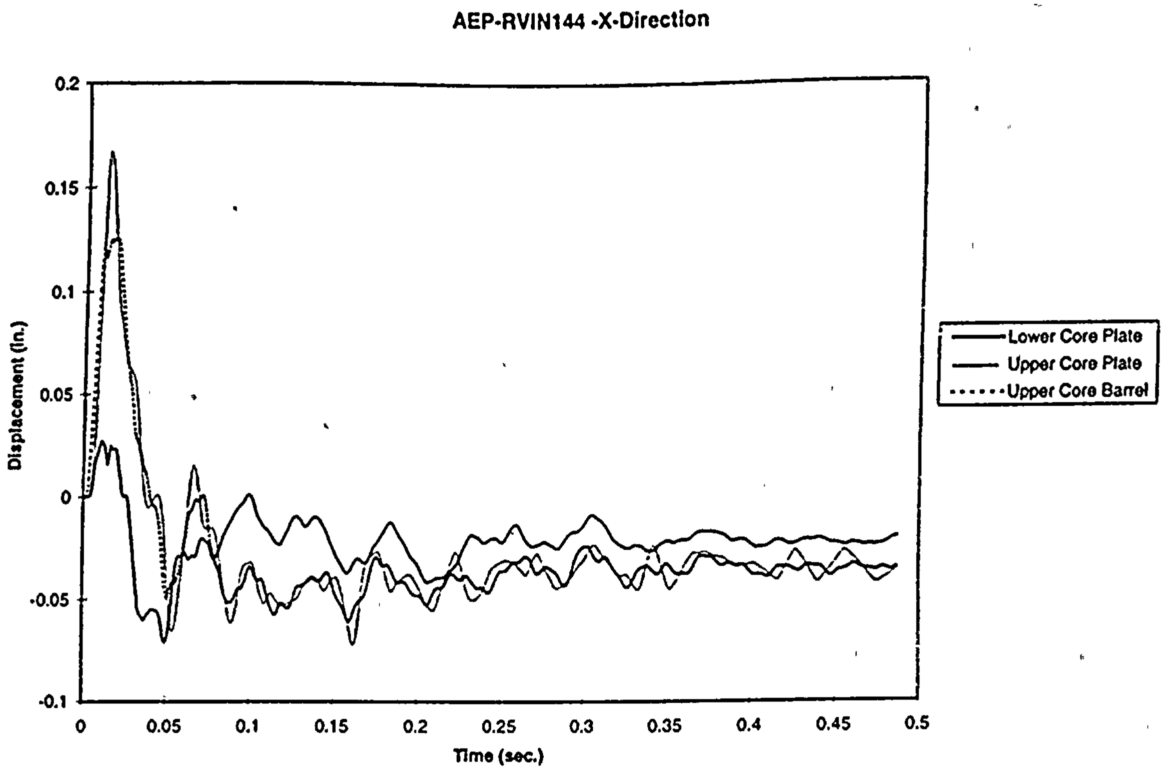


Figure 7.1 Seismic Lower Core Plate Displacements Safe Shutdown Earthquake X and Z Directions - D. C. Cook Unit 1 (AEP)



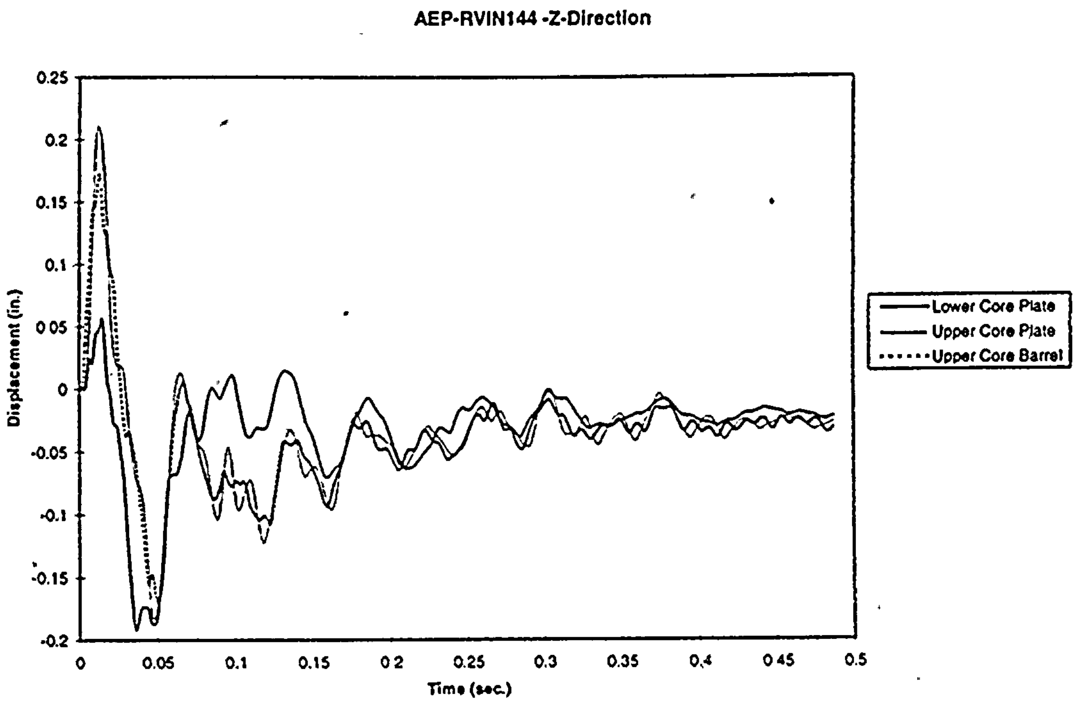
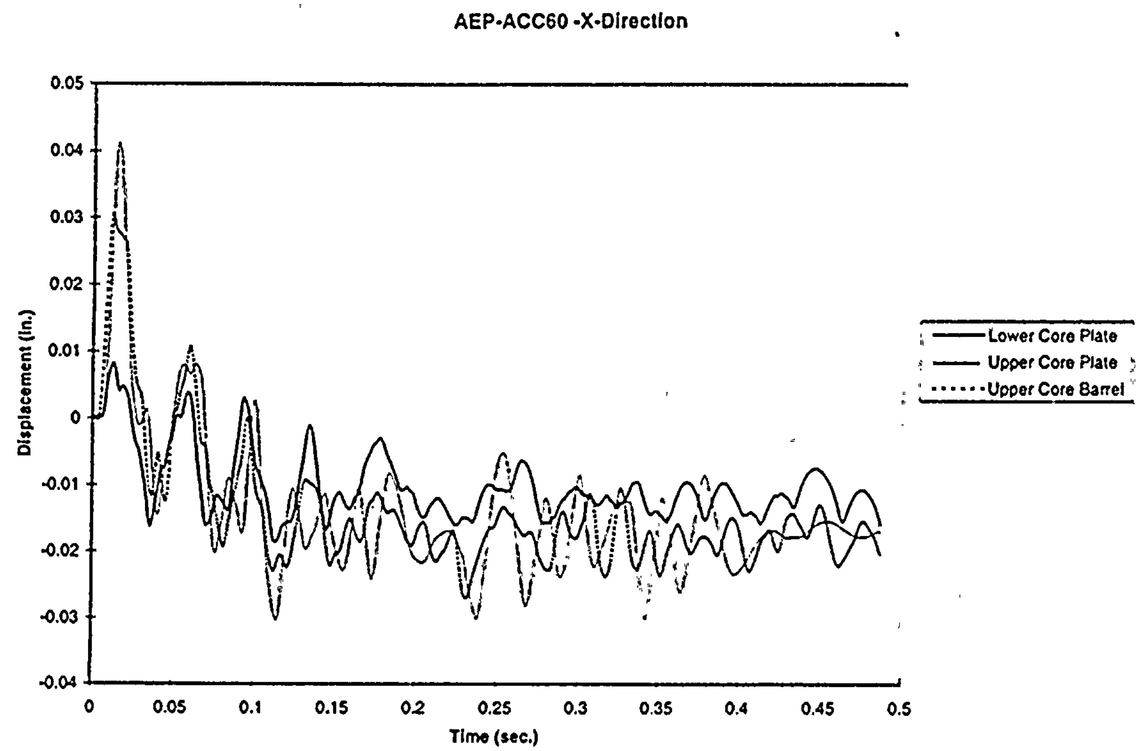


Figure 7.2 LOCA Core Plate Displacements Reactor Vessel Inlet Nozzle Pipe Break X and Z Directions - D. C. Cook Unit 1 (AEP)



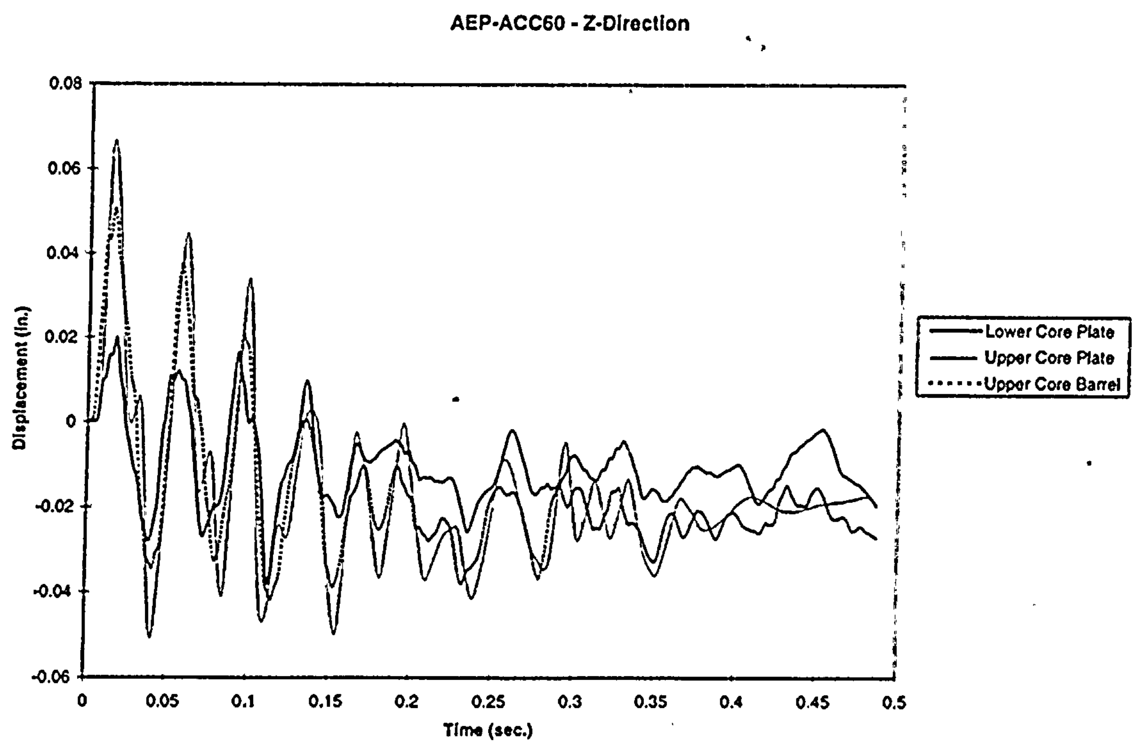
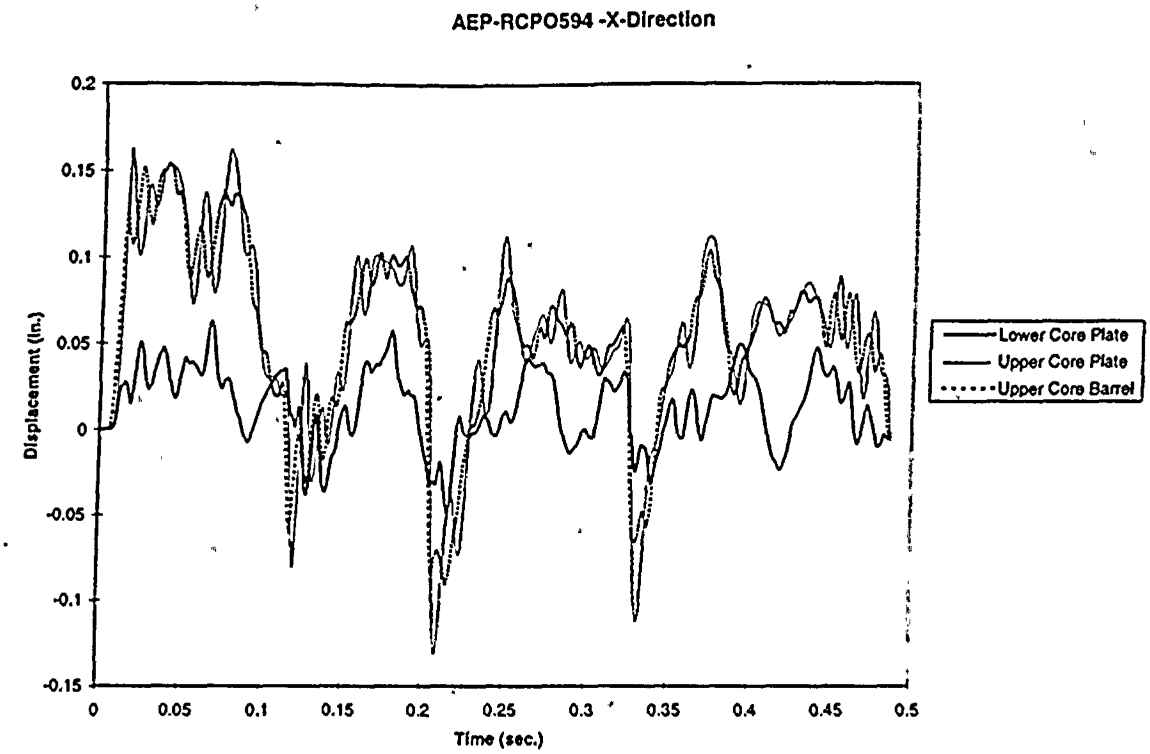


Figure 7.3 LOCA Core Plate Displacements Accumulator Line Break X and Z Directions - D. C. Cook Unit 1 (AEP)



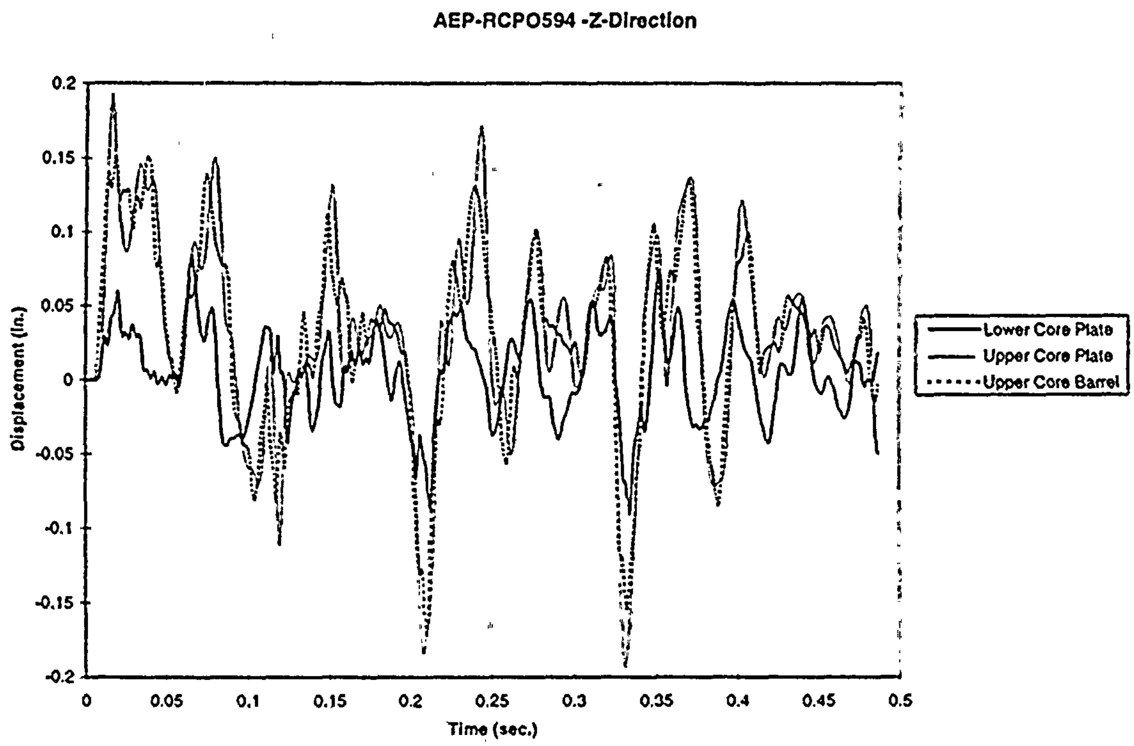
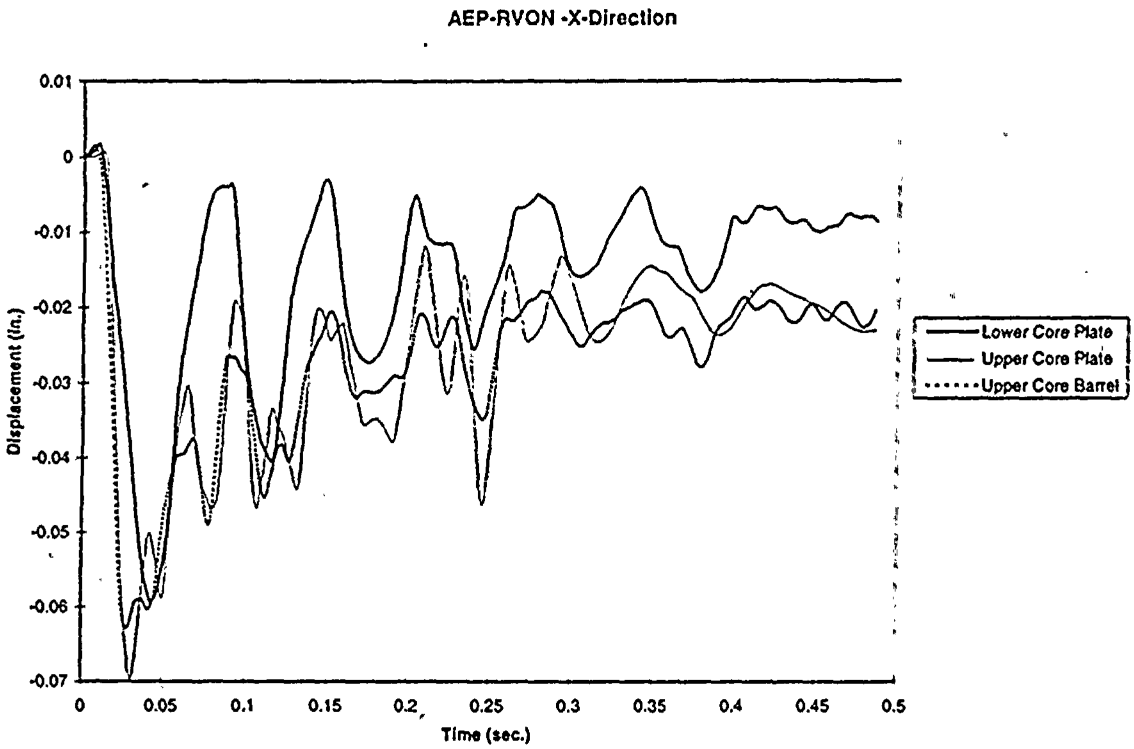


Figure 7.4 LOCA Core Plate Displacements Reactor Coolant Pump Outlet Nozzle Break X and Z Directions - D. C. Cook Unit 1 (AEP)



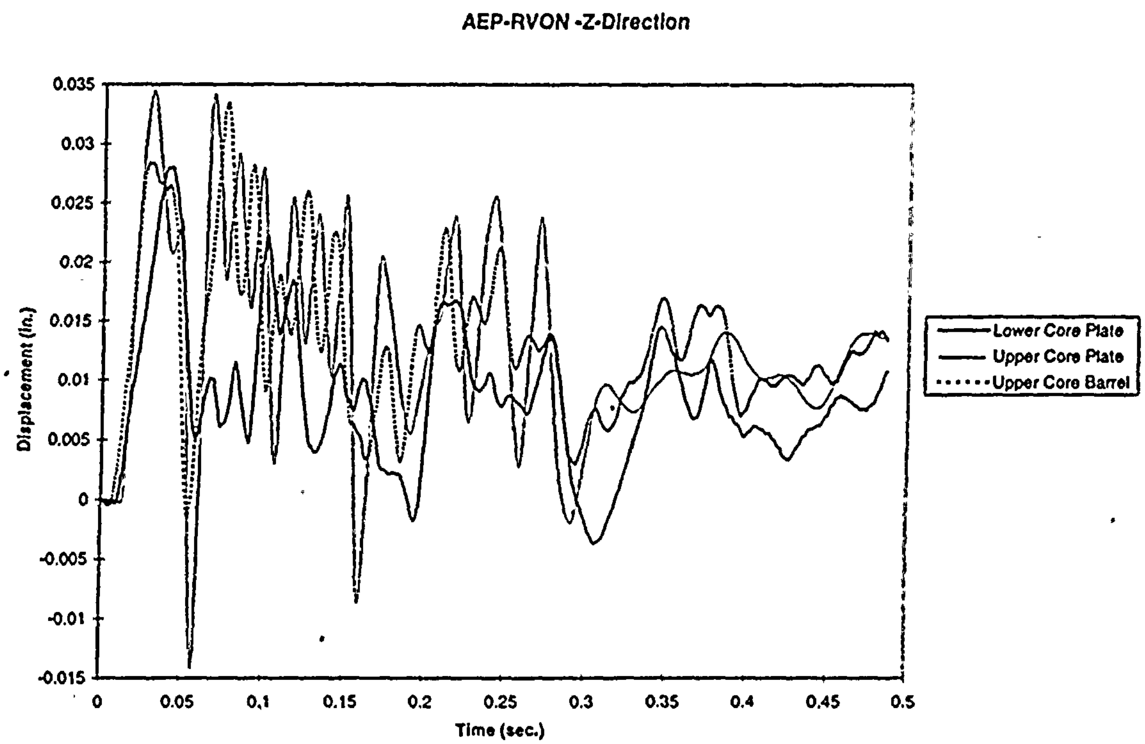
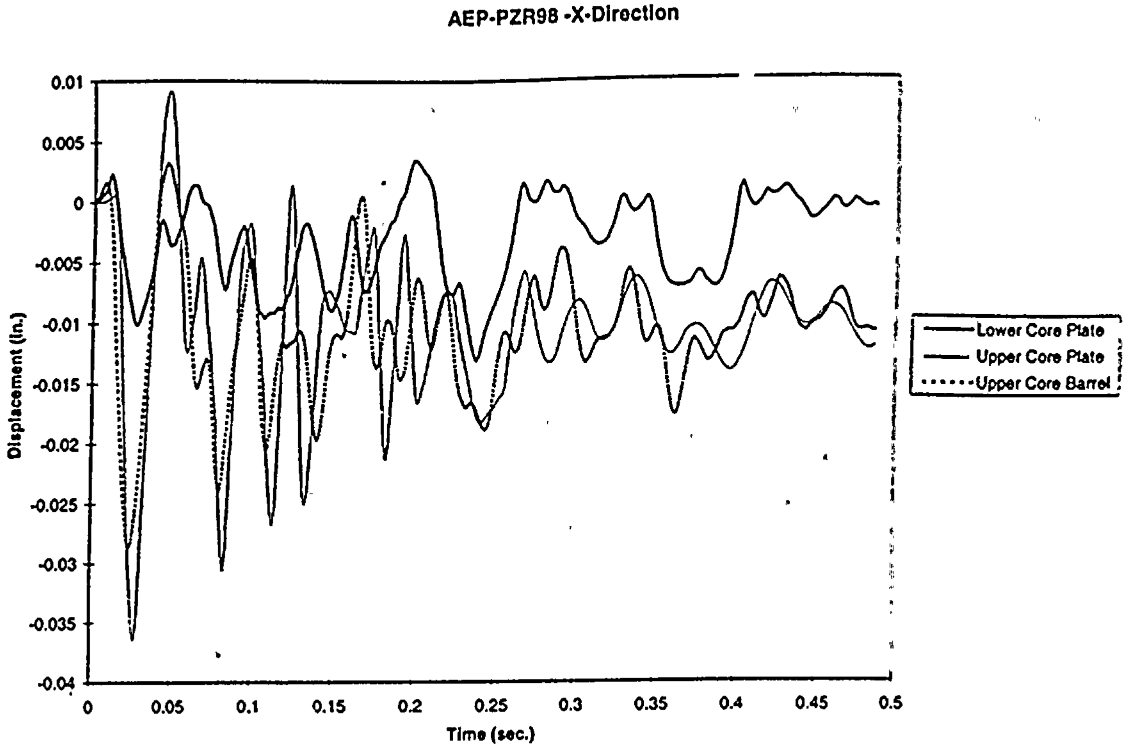


Figure 7.5 LOCA Core Plate Displacements Reactor Vessel Outlet Nozzle Pipe Break X and Z Directions - D. C. Cook Unit 1 (AEP)



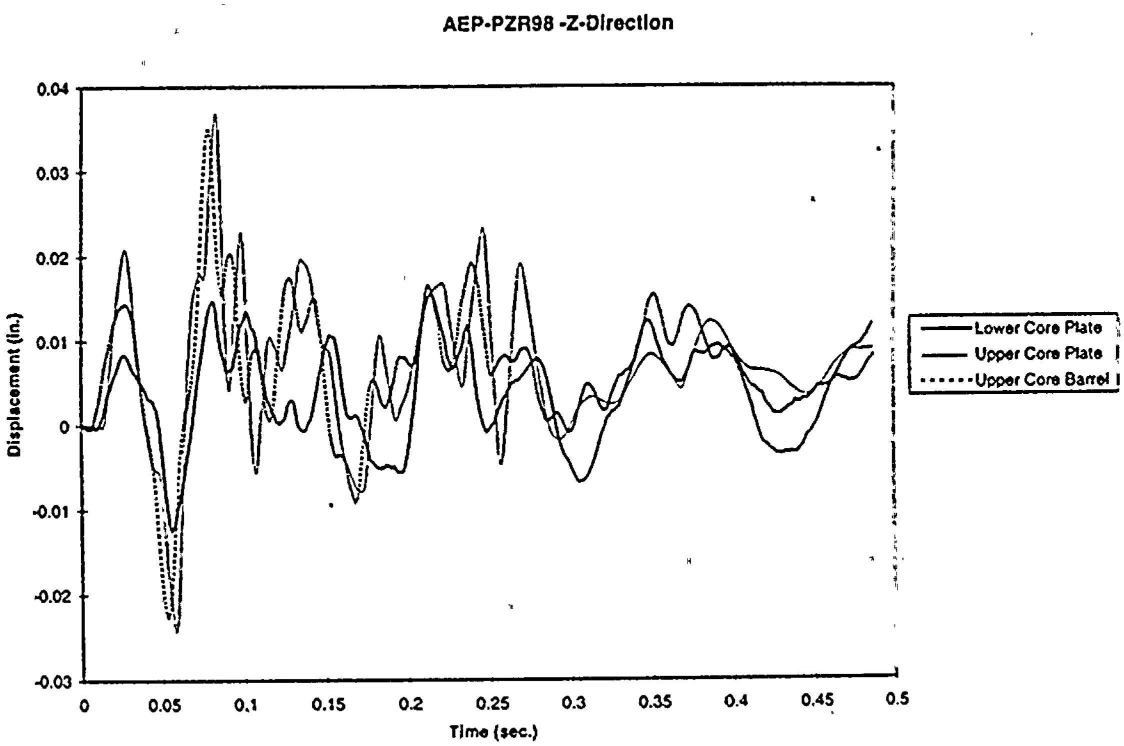
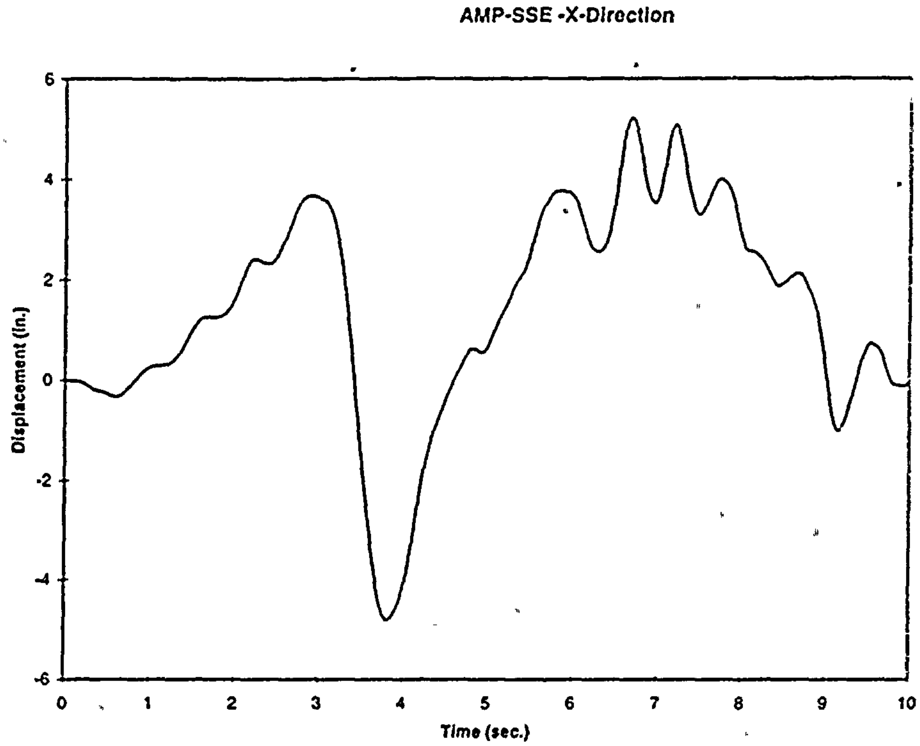


Figure 7.6 LOCA Core Plate Displacements Pressurizer Surge Line Break X and Z Directions - D. C. Cook Unit 1 (AEP)



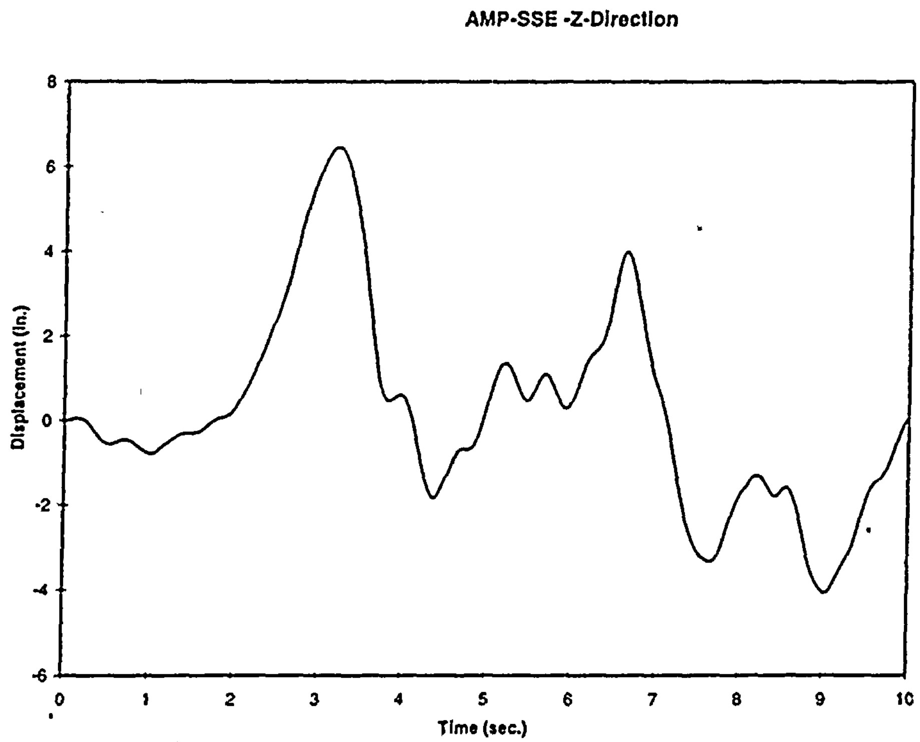
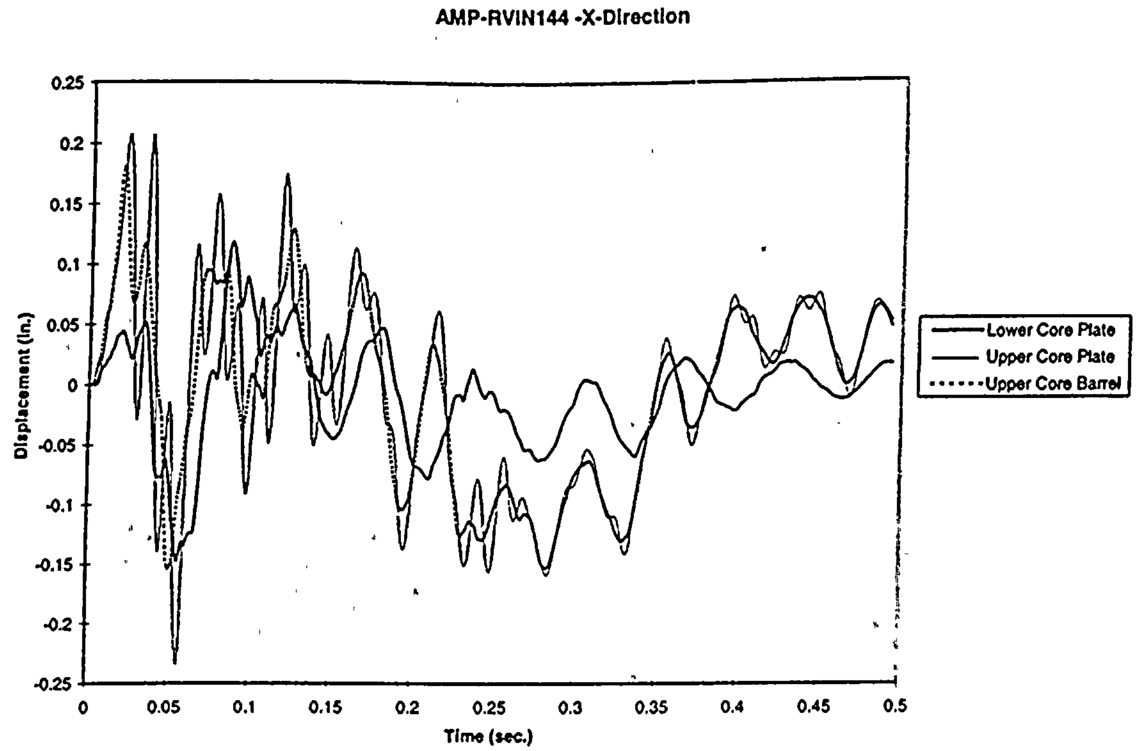


Figure 7.7 Seismic Lower Core Plate Displacements Safe Shutdown Earthquake X and Z Directions - D. C. Cook Unit 2 (AMP)



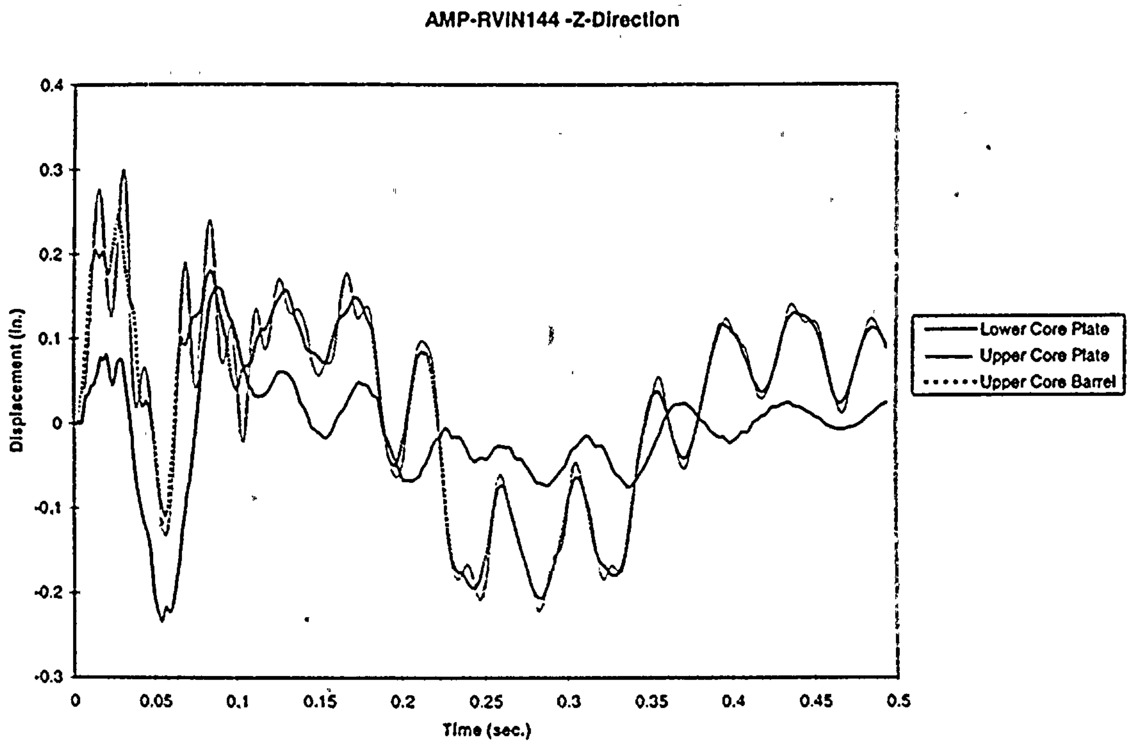
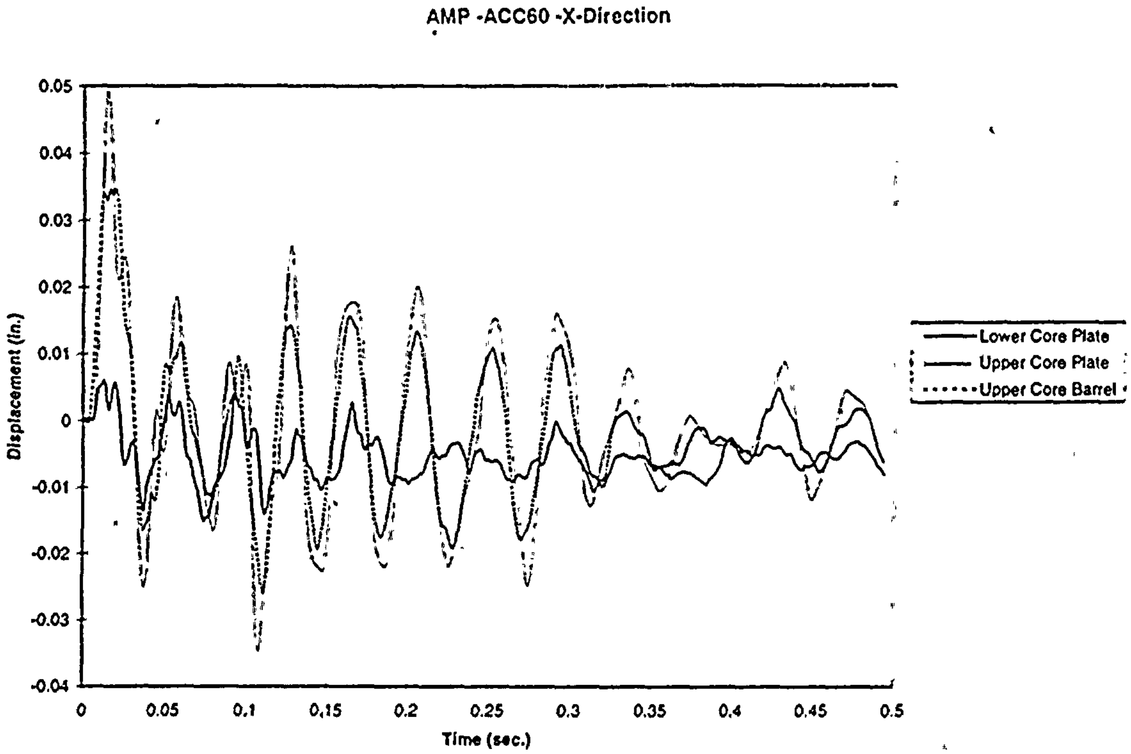


Figure 7.8 LOCA Core Plate Displacements Reactor Vessel Inlet Nozzle Pipe Break X and Z Directions - D. C. Cook Unit 2 (AMP)



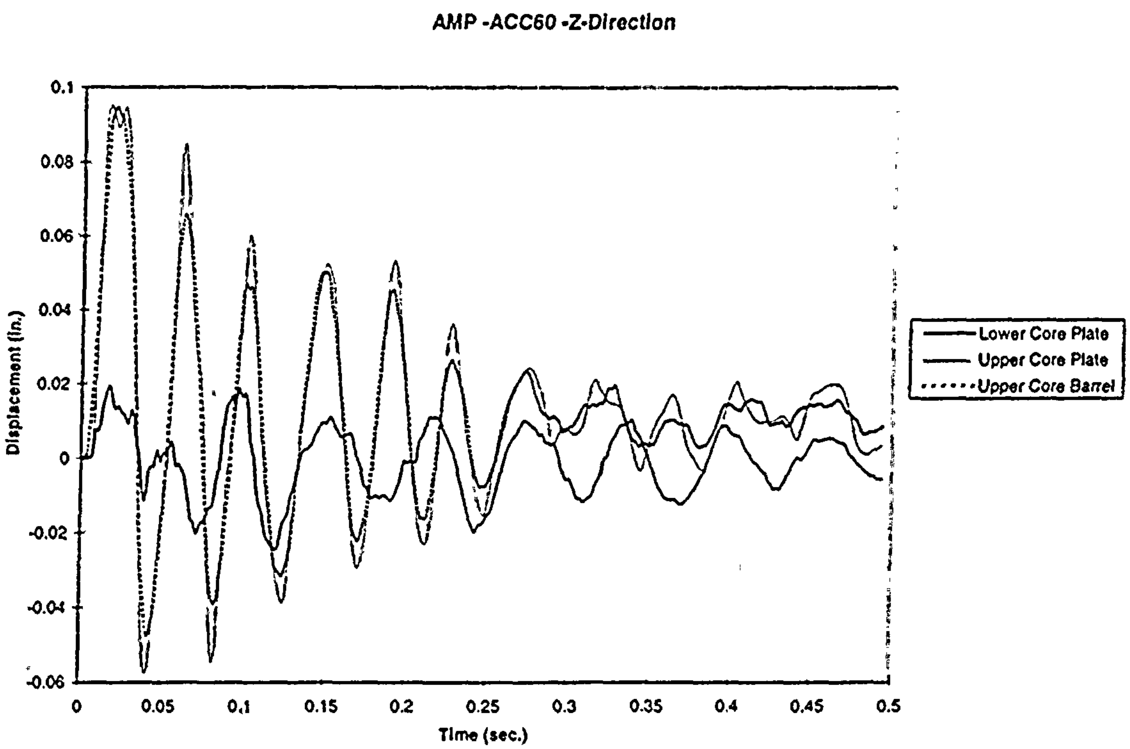
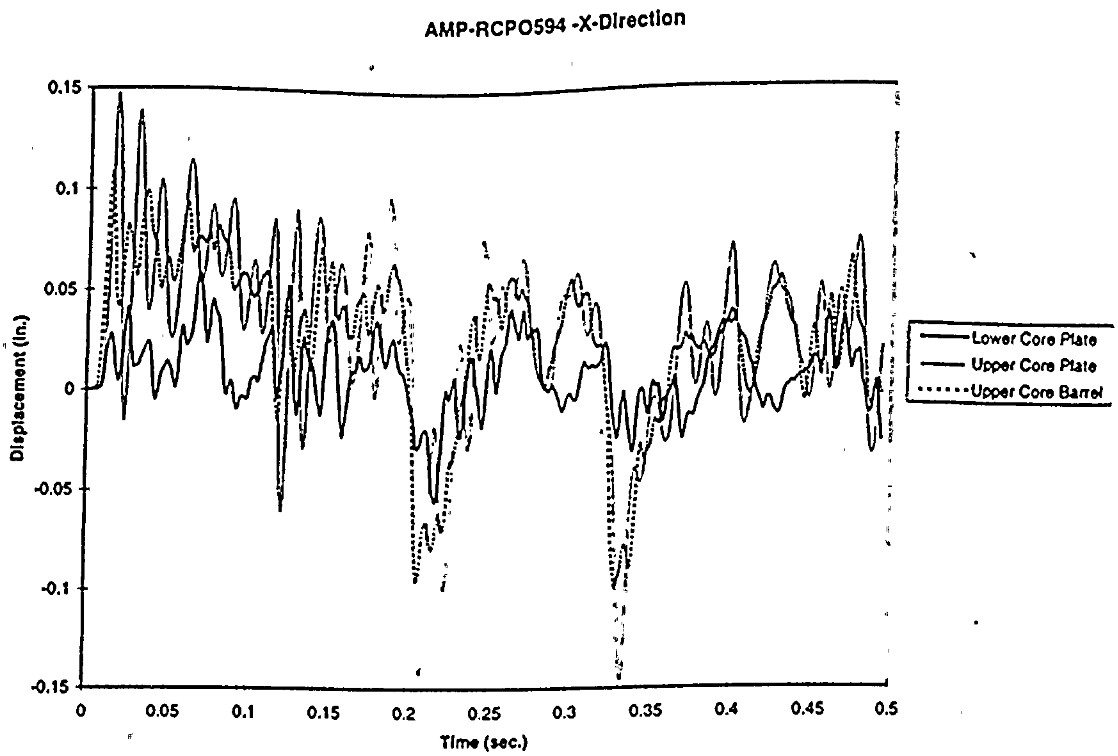


Figure 7.9 LOCA Core Plate Displacements Accumulator Line Break X and Z Directions - D. C. Cook Unit 2 (AMP)



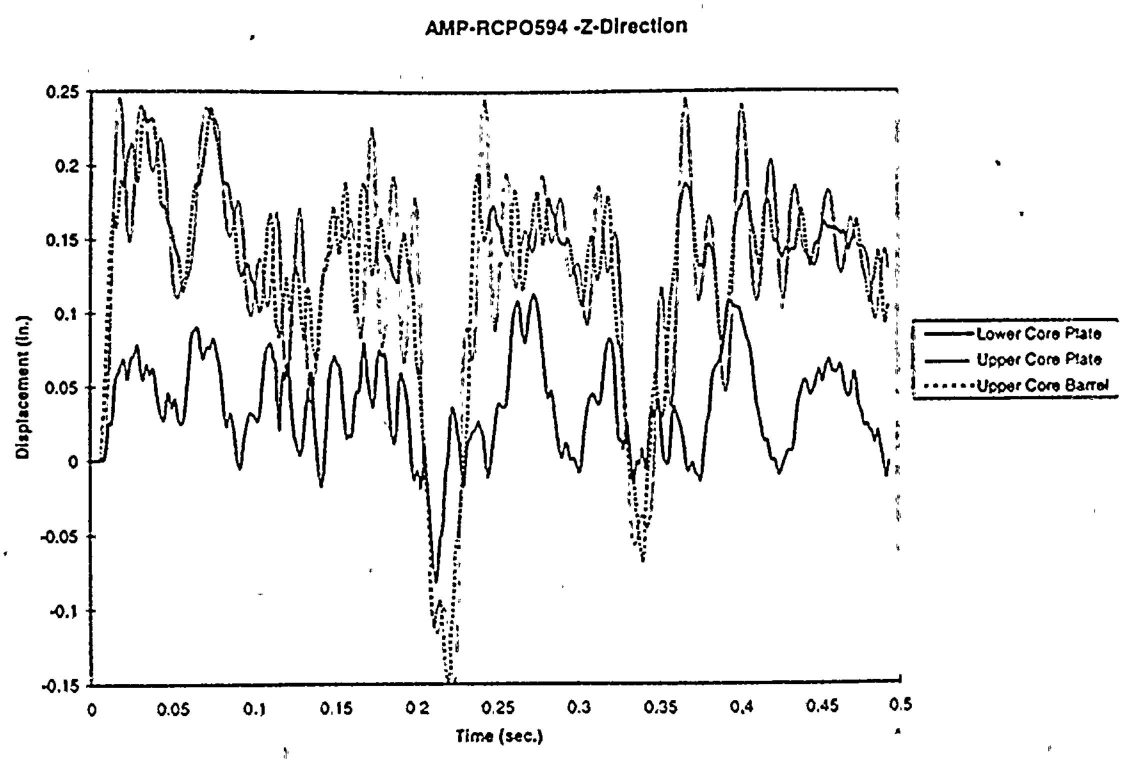
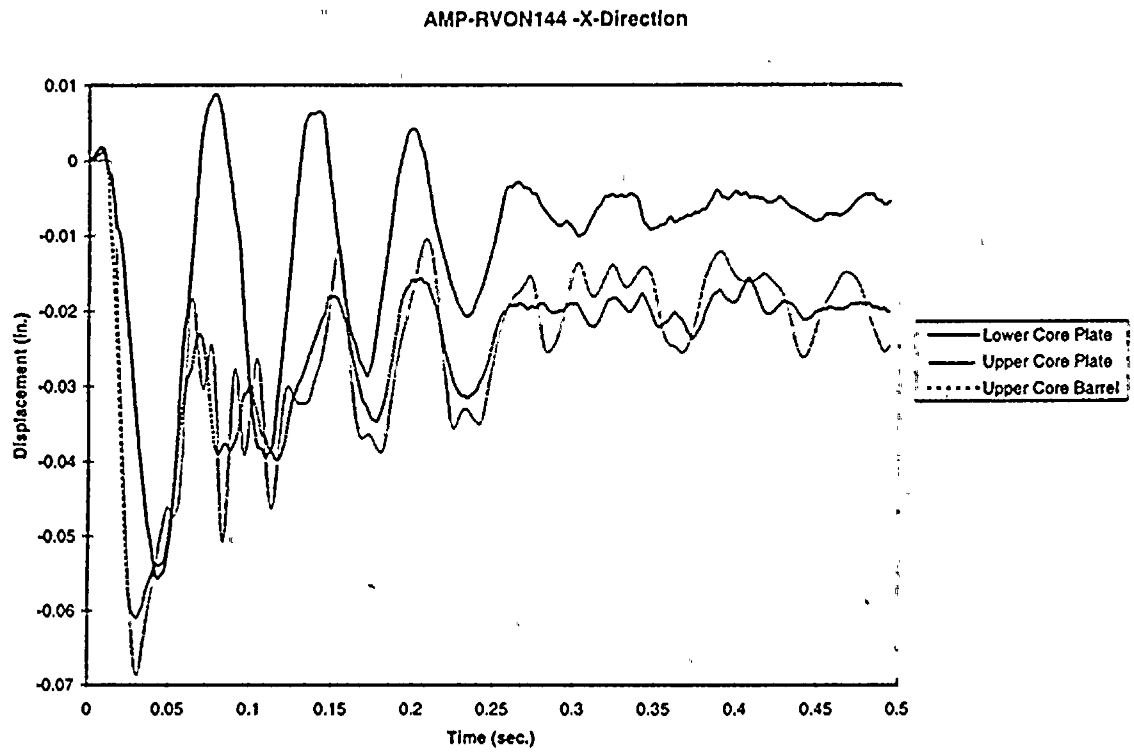


Figure 7.10 LOCA Core Plate Displacements Reactor Coolant Pump Outlet Pipe Break X and Z Directions - D. C. Cook Unit 2 (AMP)



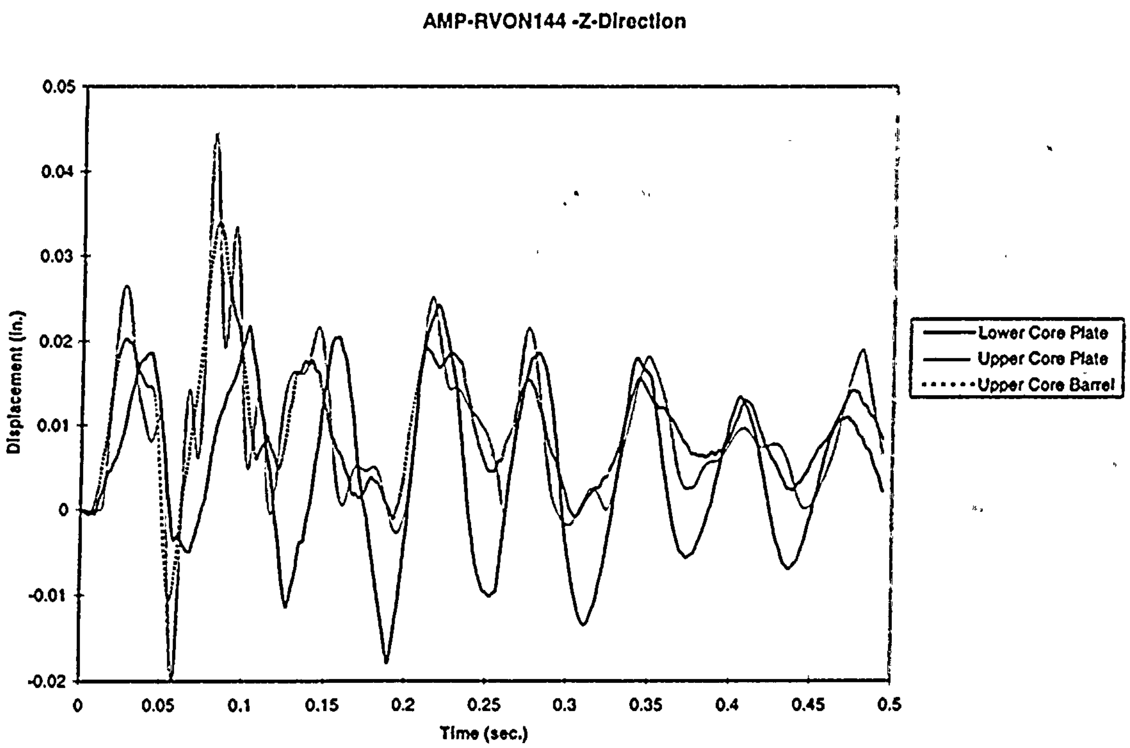
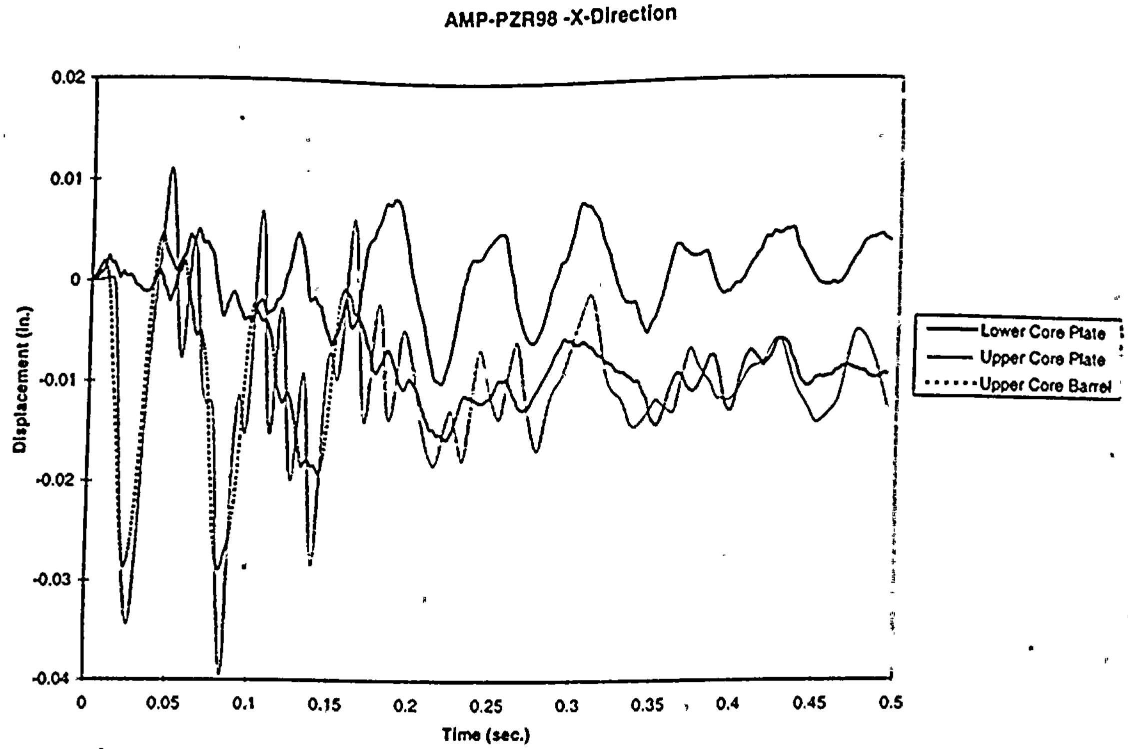


Figure 7.11 LOCA Core Plate Displacements Reactor Vessel Outlet Nozzle Pipe Break - X and Z Directions - D. C. Cook Unit 2 (AMP)



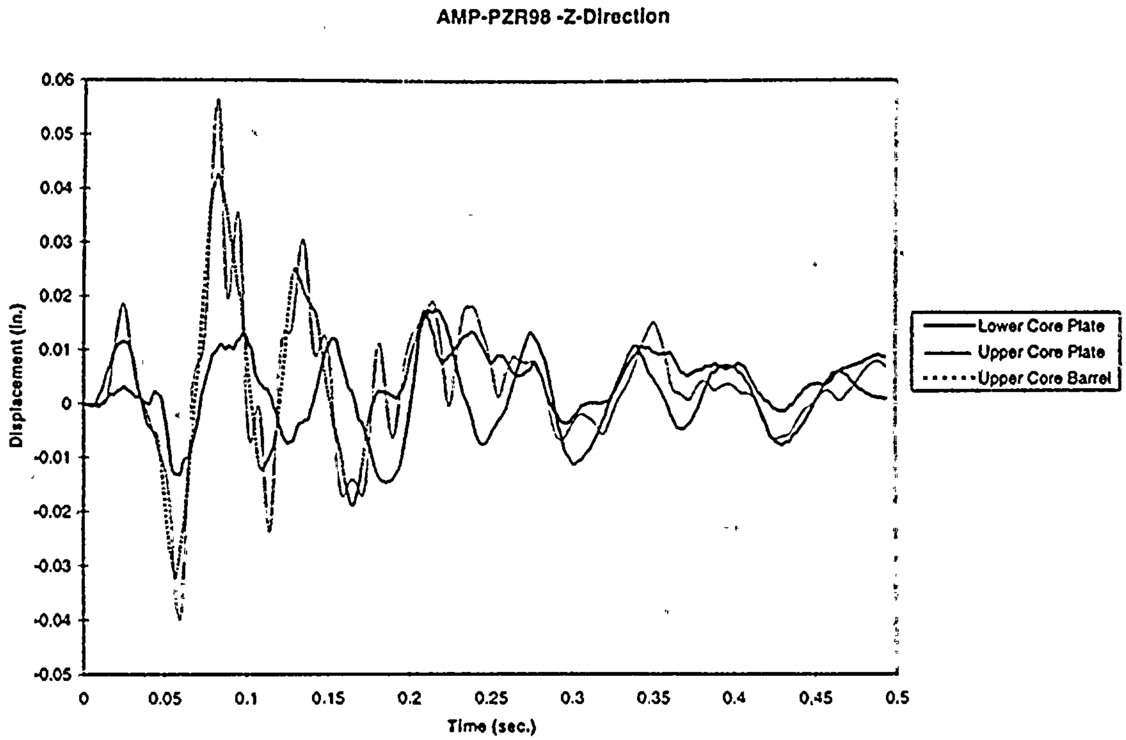


Figure 7.12 LOCA Core Plate Displacements Pressurizer Surge Line Break X and Z Directions - D. C. Cook Unit 2 (AMP)

8 FUEL ASSEMBLY GRID LOAD ANALYSIS

The general analytical procedure for evaluating fuel assembly transient response to seismic and LOCA transients is shown schematically in Figure 8.1. Forcing functions for the reactor internals model are based on postulated LOCA and seismic conditions. The hydraulic forces and loop mechanical loads resulting from a postulated LOCA pipe rupture are prescribed at appropriate locations of the Reactor Pressure Vessel (RPV) model. For the seismic analysis, the plant-specific design acceleration spectra are specified based upon the plant site characteristics. For the current analysis, the synthesized seismic time histories are calculated from the D. C. Cook plant specific acceleration response spectra envelope. These spectra are for the containment buildings at the 612.62 feet elevation and use the appropriate Design Basis damping as indicated in Reference 8.1. Both the LOCA and seismic time histories are applied to the Reactor Pressure Vessel system model. The core plate motions from the dynamic analysis of this model are obtained and are then input to the Reactor Core Model.

The Reactor Core Model includes four individual fuel assembly array models with varying row lengths and inter-assembly grid impact elements. A schematic of a typical reactor core array model is shown in Figure 8.2. The number of fuel assemblies in the array models for the D. C. Cook Units are 7, 11, 13 and 15, which represent the number of fuel assemblies in each of the core planar arrays. In Figure 8.2, a total of 15 VANTAGE 5 IFM fuel assemblies represents the maximum number of assemblies in a plane which is representative of the middle of the core. The peak grid loads for each LOCA and seismic transient are the maximum impact load obtained from four different models (rows) in the X and Z directions, i.e. parallel to the reactor vessel horizontal cardinal axes. The seismic and most limiting case of LOCA analyses were performed for every array of fuel assemblies. However, the non-limiting LOCA transient evaluations were performed only for the maximum and minimum number of fuel assembly rows of 15 and 7, respectively.

The limiting LOCA and seismic grid impact loads for homogeneous 15x15 OFA and 17x17 VANTAGE 5 IFM assembly cores are summarized in Table 8.1. The maximum grid loads, obtained from SSE and LOCA loading analyses, were combined as required using the SRSS method. The results of the seismic and LOCA analyses of the maximum impact forces for the 15x15 and 17x17 structural grids are compared to allowable grid distortion loads. These allowable grid loads are experimentally established as the 95 percent confidence level on the mean from the distribution of grid distortion data at normal plant operating temperature. Acceptability of the fuel (grid) performance for RCCA control rod insertion is verified by demonstrating that no grid deformation occurs in assemblies directly beneath control rod locations. For both Units 1 and 2, no fuel assembly grid distortion was calculated and thus control rod insertion will not be impeded by the fuel for either the limiting LBB criteria break locations or the design basis cold leg breaks. These results are documented in Reference 8.2.

Table 8.1 Fuel Assembly Grid Distortion Margin

D. C. Cook Unit 1 (AEP) - 15x15 OFA

Break Location/Size

	Acc. Line 60 in²	Surge Line 98 in²	RVIN 144 in²	RVON 144 in²	RCPON 594 in² a,c
LOCA Loads (lbs)					
SSE * Load (lbs)					
SRSS (lbs)					
Allowable Load (lbs)					
Margin **					

D. C. Cook Unit 2 (AMP) - 17x17 V5 with IFMs - Structural Grid

Break Location/Size

	Acc. Line 60 in²	Surge Line 98 in²	RVIN 144 in²	RVON 144 in²	RCPON 594 in² a,c
LOCA Loads (lbs)					
SSE * Load (lbs)					
SRSS (lbs)					
Allowable Load (lbs)					
Margin **					

D. C. Cook Unit 2 (AMP) - 17x17 V5 with IFMs - IFM Grid

Break Location/Size

	Acc. Line 60 in²	Surge Line 98 in²	RVIN 144 in²	RVON 144 in²	RCPON 594 in² a,c
LOCA Loads (lbs)					
SSE * Load (lbs)					
SRSS (lbs)					
Allowable Load (lbs)					
Margin **					

* SSE (Safe Shutdown Earthquake) is DBE (Design Basis Earthquake) for D. C. Cook

** Margin - $((F_{allowable} / F_{calculated}) - 1) \times 100\%$

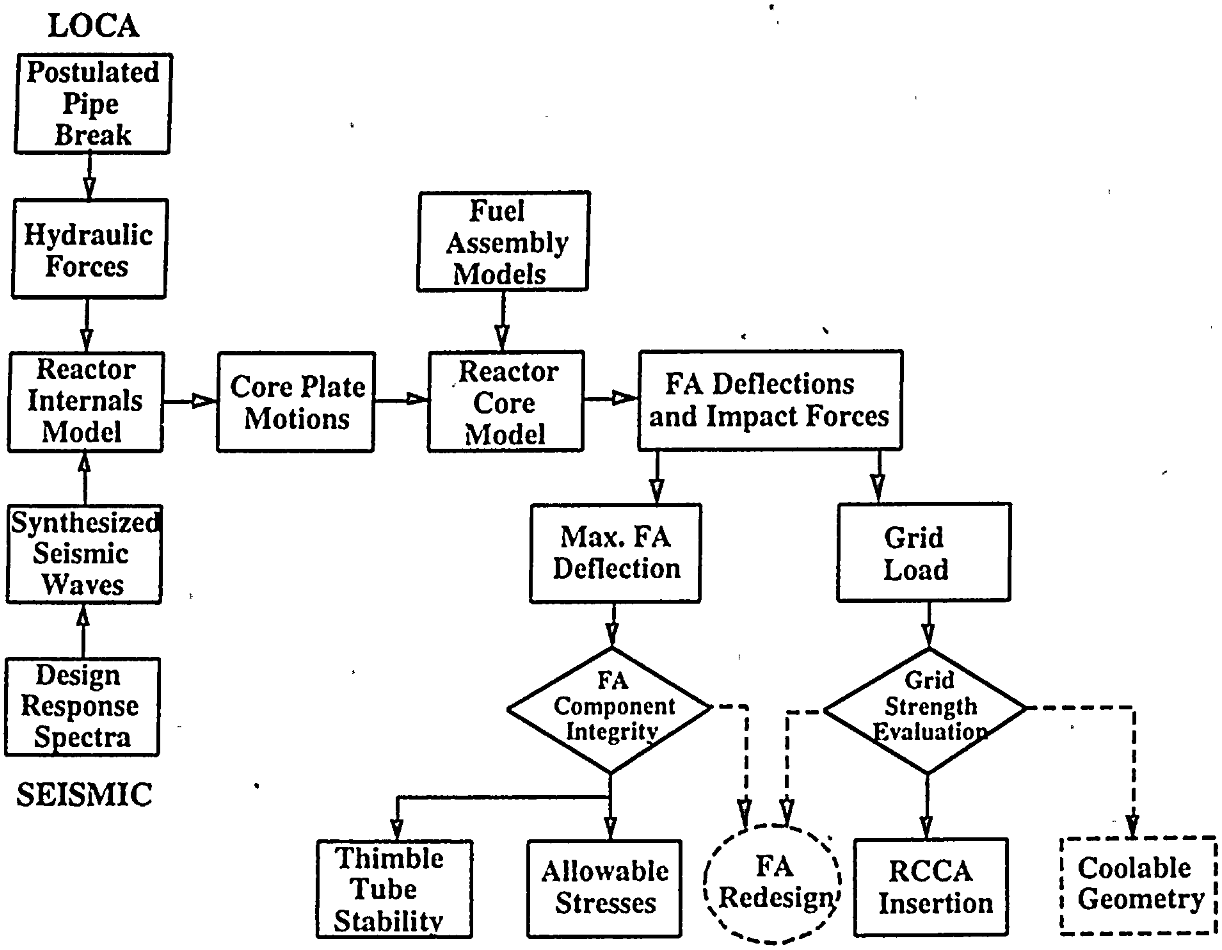


Figure 8.1 Seismic/LOCA Analysis Flow Chart

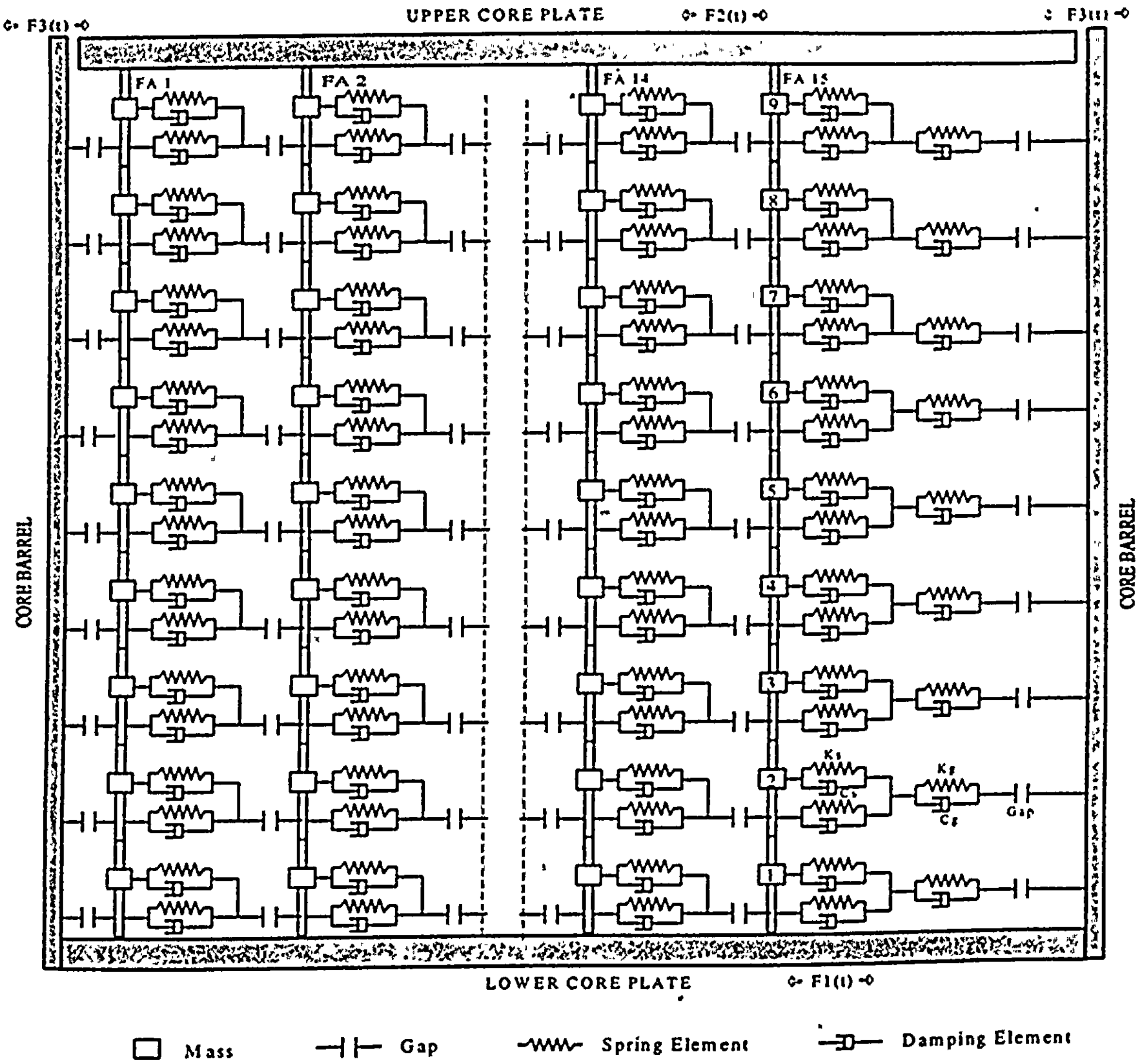


Figure 8.2 Reactor Core Model for Fuel Assembly Dynamic Analysis Row of 15 Vantage 5 IFM Fuel Assembly Array

9 RESULTS AND CONCLUSIONS

The analysis documented herein has addressed the reactor vessel components whose structural distortion in a seismic/LOCA transient environment must be conservatively limited to insure control rod insertion in the post-LBLOCA environment. Plant specific seismic response spectra and plant specific design parameters for the D. C. Cook Units 1 and 2 have been used throughout. The following results and conclusions have been obtained:

1. For both Units 1 and 2, the RCCA upper internals guide tube calculated loads are within allowable limits as established by tests such that control rod insertion will not be precluded for either the limiting LBB criteria break locations or the design basis cold leg breaks.

2. Fuel assembly grid distortion is not predicted for either Unit based upon calculated loads and measured grid load allowables such that control rod insertion will not be inhibited for either the limiting LBB criteria break locations or the design basis cold leg breaks.

Factors which contributed to the above results include the use of MULTIFLEX 3.0 with it's improved structural modeling, the relatively high reactor vessel support stiffness of the D. C. Cook Units, and the favorable total reactor vessel mass to break area ratio of the 4-loop design.

This analysis demonstrates that control rods will be inserted following the large cold leg Loss of Coolant Accident (LOCA) and the resulting negative reactivity credit can be applied in evaluating recriticality at the time of switchover to hot leg recirculation. The results provide two bases for crediting control rod insertion for post-LOCA recriticality, i.e., either the LBB criteria breaks or the Design Basis breaks may be used.

10 REFERENCES

4.1 "Safety Evaluation of Westinghouse Topical Reports Dealing With Elimination of Postulated Pipe Breaks in PWR Primary Main Loops," (Generic Letter 84-04), February 1, 1984.

4.2 "Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Through-Wall Crack," WCAP-9558, May 1981.

4.3 "Tensile and Toughness Properties of Primary Weld Metal for Use in Mechanistic Fracture Evaluation," WCAP-9787, May 1981.

4.4 "Modification of General Design Criteria 4 Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures," Federal Register, Volume 52, No. 207, October 27, 1987.

4.5 "Feasibility Study of Control Rod Insertability After a Large Break LOCA," WCAP-13834, August 1993.

5.1 "Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions," WCAP-15029, February 1998.

5.2 "Safety Evaluation of Topical Report WCAP-15029 'Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions,'", (TAC NO. MA1152), November 10, 1988.

5.3 "Safety Evaluation Related to Topical Report WCAP-14748/14749 - 'Justification for Increasing Break Opening times in Westinghouse PWRs'", (TAC NO. M98031), October 1, 1998.

5.4 "NR-98-197 Control Rod Insertability Issue - Phase 2 Evaluation LOCA Forces Results for D. C. Cook Unit 1 and 2 (AEP/AMP)," SAE-LIS-98-636, Rev. 1, February 1999.

6.1 "Scram Deflection Test Report, 17x17 Guide Tubes, 96 Inch and 150 Inch," WCAP-9251, December 1977.

6.2 "Summary Report on PGE Scrammability Test," CE-RID-223, February 17, 1969.

7.1 "Reactor Pressure Vessel and Internals System Evaluation for D. C. Cook Unit 2 Vantage 5 Fuel Upgrade with IFMs," WCAP-12828, December 1990.

7.2 "D. C. Cook Unit 1 & 2 (AEP/AMP) Control Rod Insertion - LOCA and Seismic Core Plate Motions," EDRE-EMT-951, February 1999.

8.1 AEP2-W/0049, September 12, 1989.

8.2 "Grid Load Evaluation for Fuel Assemblies in D. C. Cook Units," PD1-99-018, February 1999.

ATTACHMENT 8 TO C0999-11

I&M RESPONSES TO NRC
QUESTIONS FROM MAY 6, 1999 PUBLIC MEETING

6/30/99

Response to NRC Questions from AEP/NRC Meeting of 5/6/99

The questions raised at the meeting with the Staff in White Flint on May 6,1999 can be grouped into two categories; 1) items concerned with the effects of LOCA and seismic load induced driveline misalignments (mechanical and thermal) and 2) items concerned with fuel assembly burnup effects and their relation to control rod insertion. The responses are grouped in the same fashion.

1. Question: Would the movement of the internals packages within the vessel during the blowdown, given allowable clearances and tolerances, cause a distortion of the control rod driveline alignment that could impact the ability of the rods to insert?

   Response: Lateral displacement of the reactor vessel upper internals with respect to the reactor vessel, resulting from LBLOCA and seismic forces, could theoretically affect the ability to insert the control rods through binding of the control rod drivelines. This might occur as the drive line passes through the thermal sleeve on the reactor vessel head and subsequently through the upper section of guide tube attached to the upper internals support assembly. As discussed below, this is not a concern with the Westinghouse reactor design since clearances have been provided in the components adjacent to the driveline to prevent such an occurrence. Lateral displacement of the lower internals package does not directly affect the driveline alignment and is not a concern.

   The maximum lateral displacement of the upper internals package with respect to the vessel is limited by the gap between the upper support assembly flange and the reactor vessel, which for D. C. Cook Unit 1 at the end of blowdown (30 to 60 seconds into the LBLOCA transient) is 0.202 inches. The gap width for Unit 2 is less with a value of 0.163 inches. However, it is noted that the upper internals package does not experience significant lateral hydraulic forces during a cold leg break and could only be displaced by motion of the lower internals package which is transmitted to the upper package. This could occur through the coupling provided by the head and vessel alignment pins once the gaps surrounding the pins have closed. The lateral displacement of the lower internals is limited by contact at the vessel outlet nozzles as well as at the vessel support ledge. Thus, a number of factors indicate that the upper internals package would move less than the full gap width during a cold leg LOCA plus seismic event. However, as a conservative upper bound, the limiting vessel to upper support plate gap width at normal operating temperature is assumed.

   With the release of the Control Rod Drive Mechanism (CRDM) grippers following a reactor trip signal, the CRDM drive rod has significant radial gaps to accommodate lateral misalignments at the vessel head penetration and lower end of the thermal sleeve. In this area, the driveline is within the thermal sleeve which extends from the bottom of the CRDM housing to within a few inches of the top of the guide tube, Figure 1. There is a radial gap between the drive rod and the thermal sleeve within which the drive rod is free to move. In addition, the thermal sleeve is free to rotate at its upper end, such that at the bottom end there is radial gap between the I.D. of the

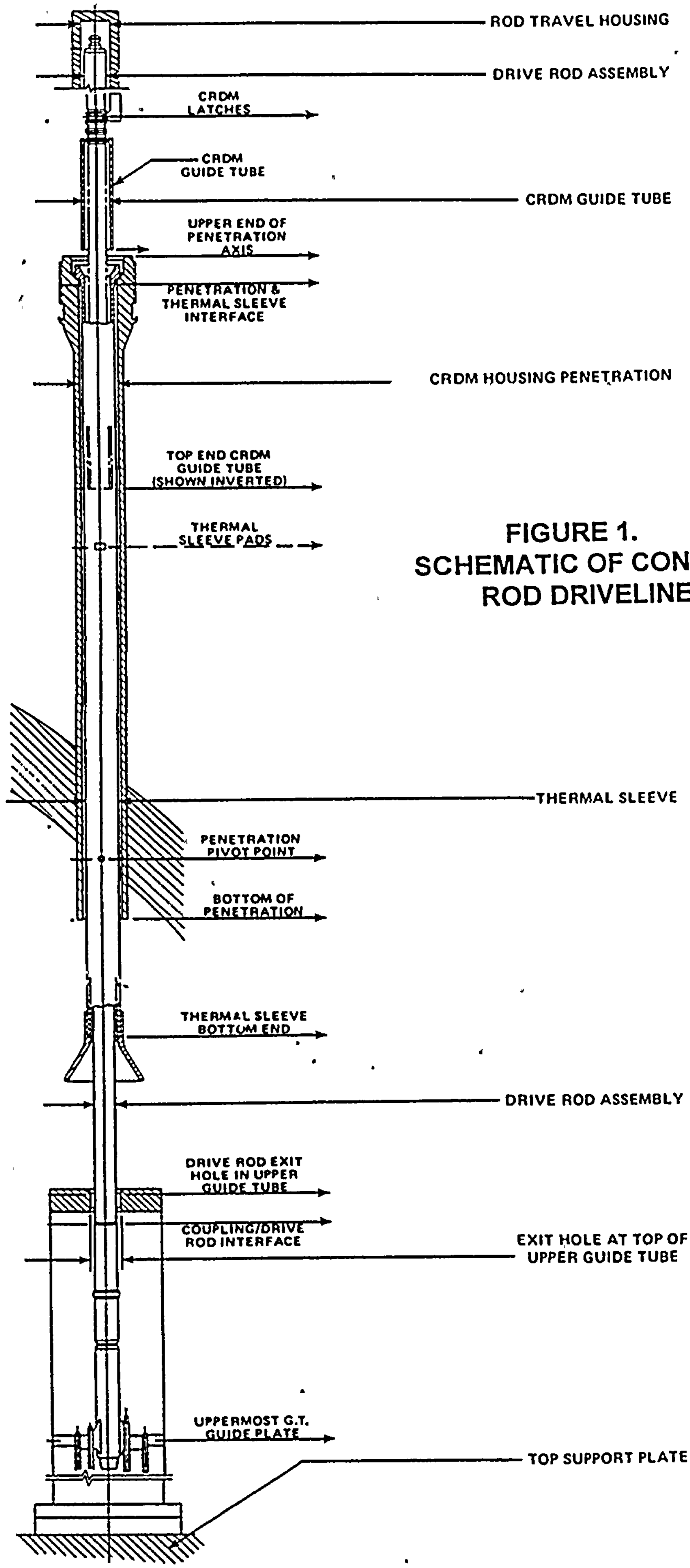


FIGURE 1. SCHEMATIC OF CONTROL ROD DRIVELINE

vessel head penetration and the O. D. of the thermal sleeve. This radial gap is available to accommodate lateral displacement of the upper internals package. Thus, the CRDM drive rod has two radial gaps available at the bottom end of the thermal sleeve to accommodate upper package displacement. The calculated bounding value for upper internals displacement inches can be accommodated by this allowable lateral displacement. Therefore, the occurrence of significant frictional forces which would retard control rod insertion will be precluded.

This evaluation assumes that the reactor vessel and internals are at the Normal Operating Temperature values at the initiation of the LOCA transient. An additional misalignment, as described below, is evaluated to address the transient temperature and thermal contraction of the reactor upper internals following the LOCA transient.

2. Question: Would the lack of cooling of the upper head region and the potential for thermal distortion due to uneven cooldown of the reactor vessel, internals, and vessel head impact the ability to insert the control rods? (The concern is that thermal distortion could impact rod insertion capability due to guide path alignment.)

   Response: While the need for negative reactivity from control rods occurs at the time of Hot Leg Switchover (4 hours to 12 hours after the accident), the insertion of control rods would occur during the LOCA transient or when the seismic and LOCA forces have decreased following the accident. LOCA forces peak within the first second of the break and are effectively zero within 10 seconds. Seismic forces are less deterministic with respect to duration, but may be assumed to occur for less than two minutes. For a large break LOCA, the rate of cooling of the reactor vessel head due to steam flow is not significant and an extended time at the normal operating temperature may be assumed. For the design basis cold leg LBLOCAs considered for D. C. Cook as well as the Leak-Before-Break size breaks, uncovery of the bottom surface of the upper internals support plate assembly (which positions the upper end of the guide tubes) occurs rapidly and cooling of this structure would then also be minimized by the steam environment. However, some liquid at saturated conditions would remain on the upper surface of the support plate assembly and provide a mechanism for rapid cooling of the structure. A conservatively high estimate of the cooldown rate of the upper support structure provides a basis for the misalignment evaluation with the vessel head, assumed to remain at the operating temperature. This thermal driveline misalignment is based upon the following assumptions:

   A. The internals/vessel thermal transient and resulting displacements are initiated with the upper internals support assembly and the lower core barrel flange assumed to be in contact with the vessel at the vessel support ledge; as a result of the LOCA forces.

   B. The upper internals package is assumed to be "pinned" at the point where the upper internals support structure flange and reactor vessel surface are in contact. (Thus, the differential thermal expansion of the upper internals and vessel head is maximized on the opposite side of the vessel)

C. The upper surface of the upper internals package remains covered with liquid and in a nucleate boiling heat transfer mode with a saturation temperature of 250 F, starting at the time of the break. This temperature corresponds to a saturation pressure of 30 psia, a conservatively low estimate of upper plenum pressure for the initial period following the break.

D. The lower surface of the support structure is exposed to steam in the upper plenum.

E. Thermal and structural behavior of upper support structure is controlled by the thermal response of the 5.0" thick upper support plate. Thermal and structural effects of the stiffening ligaments beneath the upper support plate are neglected.

F. Maximum initial reactor coolant temperatures bound the thermal effects and are assumed in the evaluation.

G. Seismic forces continue for a period of 2 minutes during which control rod insertion is conservatively assumed to be inhibited.

With these assumptions, the differential thermal contraction between the vessel and upper internals support structure is less than the allowable driveline misalignment at the limiting guide tube location for a period in excess of 5 minutes. Other guide tube locations, which are further from the point where the vessel support ledge and upper support package are in contact, have proportionately less differential expansion and better driveline alignment.

It can be concluded that sufficient drive rod clearances are available for post-LBLOCA control rod insertion based upon the worst assumptions for upper package displacement during the combined LOCA and seismic event and for the maximum cool down rate of the upper support package during the post-LOCA period.

3. Question: What are the effects of burnup on rod insertion considering warpage and distortion in the fuel assembly (thimble tubes)? Specifically address the Incomplete Rod Insertion issues.

   Response: This question addresses the interaction of Incomplete Rod Insertion (IRI) effects on control rod insertion during a LOCA/seismic event. Westinghouse considers that there is no coupling between these two issues since the potential impact of LOCA and seismic on control rod insertion, as related to fuel assembly distortion, is precluded by the fuel distortion acceptance criteria which has been selected. The criteria for control rod insertion states "No fuel assembly grid distortion shall be calculated to occur in fuel assemblies located beneath RCCA locations." Thus, the combined LOCA and seismic loads are not allowed to degrade the ability of the rodded assemblies to accept control rod insertion.

   In addition, the D. C. Cook Units have low core average fluid temperatures in comparison to the plants that actually have exhibited the IRI event. Industry testing has shown that most high drag occurs in high temperature plants (i.e., with Tout > 610°F). Both Cook Units operate with vessel outlet temperature less than 610°F. Also, Unit 2 has a full core of fuel with IFM grids and Unit 1 will begin introducing fuel with IFM grids in Cycle 17. Plant control rod drag measurements have consistently shown that fuel assemblies with IFM grids exhibit lower control rod drag at high burnup than fuel without IFM grids. This has been related to lower axial loads on the guide thimbles in the lower half of the core due to the IFM pressure drop and, consequently, less tendency for thimble bowing. As a result, incidents of Incomplete Rod Insertion would not be predicted for the D. C. Cook units.

4. Question: What are the effects of burnup on rod insertion considering grid crush and distortion in the fuel assembly (thimble tubes)?

   Response: The Westinghouse approach to evaluating fuel assembly grid strength is based upon the requirements defined in NEUREG 0800, Appendix A, Section C.1 which permits the use of unirradiated grids tested at operating temperatures. As defined in paragraph 2 of that section, "The consequences of grid deformation are small. Gross deformation of grids in many PWR assemblies would be needed to interfere with control rod insertion during an SSE (i.e., buckling of a few isolated grids could not displace guide tubes significantly from their proper location). In a LOCA, gross deformation of the hot channel in either a PWR or a BWR would result in only small increases in peak cladding temperature. Therefore, average (test) values are appropriate, and the allowable crushing load P(crit) should be the 95% confidence level on the true mean as taken from the distribution of measurements on <u>unirradiated production grids at (or corrected to) operating temperature</u>. While P(crit) will increase with irradiation, ductility will be reduced. The extra margin in P(crit) for irradiated grids is thus assumed to offset the unknown deformation behavior of irradiated grids beyond P(crit)."

   Similarly, the Westinghouse approach to evaluating other fuel assembly components is based upon the requirements defined in NEUREG 0800, Appendix A, Section C.2.

"ASME Boiler and Pressure Vessel Code values and procedures may be used where appropriate for determining yield and ultimate strengths. Specifications of allowable values may follow the ASME Code requirements and should include consideration of buckling and fatigue effects." This is applied to fuel assembly guide tubes and assembly nozzles.

5. Question: What is the impact of reactor core barrel vent valves on the ability of control rods to insert.

   Response: This is a design feature of the B&W vessel at Crystal River but is not part of any Westinghouse reactor internals design. Neither B&W nor Westinghouse claim that rods are inserted during the de-pressurization (blowdown) period of the LOCA transient during which time the B&W core barrel vent valves are required to function. However, Westinghouse cannot comment on what effects a vent valve design would have on LOCA forces and the related ability to demonstrate rod insertion.

6. Question: What is the basis for selecting the cold leg break locations for the Design Basis breaks and for those developed from LBB criteria?

   Response: The Westinghouse basis for selecting Reactor Coolant System break sizes and locations was established in the 1970 through 1975 period and is documented in WCAP-8082-P-A. This topical report presents the Westinghouse implementation of the ANS criteria for the Westinghouse primary reactor coolant loop design. The postulated locations and types of break are derived on the basis of stress and fatigue analysis, for the Normal Operation and Condition II system transients. The report presents the basis for the pressure and thermal transients used in the loop analysis to derive stress and fatigue usage factors. The results of transient analysis are summarized to illustrate the influence of the pressure, thermal and seismic transients. Based on the calculations performed, the limited displacement reactor vessel inlet nozzle break and the double ended guillotine reactor coolant pump outlet nozzle break were identified as the design basis breaks in the cold leg, WCAP-8082-P-A, Table II.D-2.

   On February 1, 1984, the NRC issued a Safety Evaluation Report, Generic Letter 84-04, on WCAP-9558 and WCAP-9787 which address the use of Leak-Before-Break (LBB) technology for eliminating double ended pipe ruptures of the main reactor coolant piping from the design basis of nuclear plants, as was defined in GDC-4. As a result of these studies performed for the Westinghouse Owners Group, double ended pipe ruptures of the RCS branch piping became the design basis for all plants qualified under the LBB program. D. C. Cook Units 1 and 2 are included in the qualified group. The limiting branch line breaks in the Westinghouse designed plants are now the Accumulator Line break in the cold leg and the Pressurizer Surge Line break in the hot leg.

7. Question: Was MULTIFLEX 3.0 modified to perform the control rod insertability analysis and if not why is it directly applicable to this analysis?

   Response: MULTIFLEX 3.0 input files which were used for the control rod insertion calculations were identical to those used for the Baffle-Barrel-Bolt program with the exception of a change of break opening time from 20 ms to 1 ms. This change was made to be in conformance with the SER for WCAP-14748 / 14749, "Justification for Increasing Break Opening Times in Westinghouse PWRs," in which it was stated that "the staff's approval of the subject WCAP is limited to its application to the BBB program." The objective to the Baffle-Barrel-Bolt program is to determine the magnitude of bolt deterioration which would lead to unacceptable levels of fuel assembly grid distortion. Thus, the same type of calculations are required in both studies and the same code inputs are appropriate. Additional calculations performed for the Baffle-Barrel-Bolt program to determine the pressure within the baffle former region were not required for the rod insertion program and were not performed.

ATTACHMENT 9 TO C0999-11

SCIENTECH INC. REVIEW REPORT, DAP-43-99:

INDEPENDENT REVIEW OF CONTROL ROD INSERTION FOLLOWING A COLD LEG LBLOCA, D. C. COOK, UNITS 1 AND 2

NON-PROPRIETARY VERSION

SCIENTECH, INC.

11821 PARKLAWN DRIVE ■ ROCKVILLE, MD 20852 ■ PHONE: 301-468-6425 ■ FAX: 301-468-0883

August 26, 1999

Mr. Gregory J. Hill
American Electric Power Nuclear Generation
500 Circle Drive
Buchanan, MI 49107

Subject: DAP-43-99: Transmittal of Final Non-proprietary Report: "Independent Review of Control Rod Insertion Following a Cold Leg LBLOCA, D. C. Cook, Units 1 and 2"

Dear Mr. Hill:

Enclosed please find two (2) copies of the final report "Independent Review of Control Rod Insertion Following a Cold Leg LBLOCA, D. C. Cook, Units 1 and 2", which SCIENTECH prepared as a deliverable for work under AEP Contract No. A-11520, dated July 12, 1999. This final report has been revised to exclude Westinghouse proprietary information.

SCIENTECH has incorporated the revisions which you recommended in order to exclude Westinghouse proprietary information. Should you have questions on the enclosed review, please call either Len Ward at (301) 255-2279 or me at (301) 255-2273.

Very truly yours,

Dan Prelewicz

Dan Prelewicz
Project Manager

ccs: E. Hollis, SCIENTECH
L. Ward, SCIENTECH
File: 17089-001