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Control Rod Insertion Following a Cold Leg LOCA for Watts Bar Unit 1

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July 2008

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LIST OF ACRONYMS AND ABBREVIATIONS

AEP	American Electric Power
AISC	American Institute of Steel Construction
ASME	American Society of Mechanical Engineers
BWR	Boiling Water Reactor
BOT	Break-Opening-Time
CRDM	Control Rod Drive Mechanism
ECCS	Emergency Core Cooling System
GDC	General Design Criteria
GT	Guide Tube
ID	Inside Diameter
IFM	Intermediate Flow Mixer
LBB	Leak-Before-Break
LOCA	Loss-of-Coolant Accident
NFBU	Nuclear Fuel Business Unit
NP	Neutron Pad
NSAL	Nuclear Safety Advisory Letter
NSSS	Nuclear Steam Supply System
NRC	Nuclear Regulatory Commission
OD	Outside Diameter
PWR	Pressurized Water Reactor
RCCA	Rod Control Cluster Assembly
RCS	Reactor Coolant System
RFA	Robust Fuel Assembly
RPV	Reactor Pressure Vessel
SER	Safety Evaluation Report
SRSS	Square Root Sum of the Squares
SSE	Safe Shutdown Earthquake
TPBARs	Tritium Producing Burnable Absorber Rods
USNRC	United States Nuclear Regulatory Commission
V+P+	VANTAGE+ fuel with PERFORMANCE+ Fuel features
WCAP	Westinghouse Technical Report Number Preface (formerly Westinghouse Commercial Atomic Power)
WECAN	General Purpose Finite Element Computer Code (Westinghouse Electric Company ANalysis)
WOG	Westinghouse Owners Group

1 INTRODUCTION/BACKGROUND

In the year 2000, the WOG contracted Westinghouse to perform comprehensive analyses and evaluations to demonstrate that control rods will be inserted under LOCA+Seismic conditions following a design basis cold leg break for 3-loop and 4-loop Westinghouse plant designs. Once demonstrated, negative reactivity credit associated with control rods can be applied when evaluating recriticality at the time of switchover to hot leg ECCS recirculation. Recriticality at the time of switchover to hot leg ECCS recirculation was a concern in the scenario where the containment sump dilutes as boron builds up in the core. This program, which demonstrates that control rods will be inserted for cold leg breaks, called the WOG Control Rod Insertion Program, is documented in Reference 1.1.

For the WOG Control Rod Insertion Program, calculations were performed for the limiting cold leg breaks defined by the application of the LBB criteria consistent with the design basis for each of the plants included in the program. Bounding plant design parameters, seismic response spectra, and LOCA assumptions for the 3-loop and 4-loop Westinghouse plant designs were used throughout. Loads on distortion sensitive components, which may affect the ability to insert control rods, were considered. These included:

- RCCA Guide Tubes
- Fuel Assembly Grids
- Control Rod Driveline
- CRDM Seismic Platform Assembly

Since the focus of the WOG Control Rod Insertion Program was on the issue of sump dilution at the time of switchover to hot leg ECCS recirculation, and since sump dilution occurs only for cold leg breaks, only cold leg breaks were considered for that program.

Shortly after the start of the WOG Control Rod Insertion program, the NRC issued an SER for the D. C. Cook Control Rod Insertion Program (References 1.2 and 1.3). The D. C. Cook program considered limiting hot leg and cold leg breaks in the main loop piping as well as the largest branch lines (assuming LBB). In the SER, the NRC agreed that control rod insertion was adequately demonstrated to allow credit for the negative reactivity provided by the insertion of the rod cluster control assemblies into the reactor core following a design basis LOCA. Furthermore, the NRC agreed that the use of LBB is acceptable in calculating LOCA+Seismic loads while addressing control rod insertion.

Post-LOCA subcriticality is confirmed as a normal part of the Reload Safety Evaluation process. Starting in Cycle 6, Watts Bar Unit 1 has begun irradiating a limited number of TPBARs (Tritium Producing Burnable Absorber Rods). The post-LOCA subcriticality analyses for Cycle 6 conservatively assumed TPBAR failure and leaching of the lithium-6 for cold leg break LOCAs without the benefit of control rod credit. Currently, the Watts Bar Unit 1 minimum boron concentrations for the RWST and accumulators are 3,100 ppm and 3,000 ppm, respectively. To support the future use of larger numbers of TPBARs in Watts Bar Unit 1 while maintaining reasonable RWST and accumulator minimum boron concentrations, it is desirable to credit control rod insertion in the cold leg break post-LOCA subcriticality analyses.

The CRDM pressure housings and CRDM head adapter form the pressure boundary of the control rod driveline external to the reactor vessel head. Structural evaluations due to LOCA+seismic were not part of the original WOG program. In addition, a Nuclear Safety Advisory Letter, Reference 1.4 relating to an under-estimation of seismic loads used for the design basis structural evaluation of the CRDM head adapters, was not considered in the original WOG program.

The purpose of this report is to present the results of analyses and evaluations originally reported in Reference 1.1, including the supplemental analyses for CRDM pressure housings, CRDM head adapter and the NSAL that will demonstrate that control rods will be inserted following a design basis cold leg Loss-of-Coolant Accident for Watts Bar Unit 1. The intention of demonstrating control rod insertion is so the resulting negative reactivity credit can be applied in evaluating potential recriticality in a post-LOCA scenario.

2 ANALYSES DESCRIPTION

2.1 OBJECTIVE

The objective of these analyses and evaluations is to demonstrate that the control rods will insert in the Watts Bar Unit 1 plant under post-LOCA and seismic conditions for cold leg design basis breaks. Table 1 summarizes the applicability of evaluations and analyses with respect to control rod insertability for Watts Bar Unit 1.

2.2 ANALYSES ASSUMPTIONS

LBB Application

On February 1, 1984, the NRC issued a Safety Evaluation Report (Reference 2.1) on References 2.2 and 2.3 which addressed the use of 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 at the time). As a result of these studies, and subsequent plant-specific LBB analyses, double ended pipe ruptures of main coolant piping were no longer the design basis for all plants qualified under the LBB program. All Westinghouse domestic 3-loop and 4-loop plants now, to some extent, incorporate LBB in their licensing basis. With LBB, the limiting branch line breaks in the Westinghouse-designed plants are 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 2.4).

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
- Replacement Steam Generator design
- 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 1980s and has been described to the utilities in the engineering reports associated with the transition to VANTAGE 5 fuel. This application of LBB to the fuel assembly design had been discussed with the NRC in 1993 (documented in Reference 2.5), and most recently in an AEP/NRC meeting on May 6, 1999, Reference 2.6.

As mentioned earlier in this report, the use of LBB for the post-LOCA sump dilution issue was accepted by the NRC as discussed in an SER for the D. C. Cook Control Rod Insertion Program (References 1.2 and 1.3).

Table 1 Analysis/Evaluation Applicability Summary

	Plant Group	Head	Fuel	Guide Tubes	Fuel Integrity	Driveline Alignment	CRDM Platform
Watts Bar 1	4-Loop 17x17 NP	STD	W – V5H/RFA-2	Bounding Analysis	Bounding Analysis	Bounding Evaluation	Bounding Analysis

Plant Design Assumptions

Watts Bar Unit 1 plant design features were bounded in a model representing a Westinghouse 4 loop, 17x17 array fuel plant design. Specific Watts Bar Unit 1 plant design features such as upflow barrel/baffle region, neutron pad design, vessel support stiffness and upper internals design configuration were specifically modeled or conservatively addressed. The impact of other major plant design changes to Watts Bar Unit 1 since the WOG analysis, such as the replacement steam generator program and the introduction of RFA-2 fuel have also been considered and evaluated.

Break-Opening Time

The choice of BOT in the LOCA hydraulic forces analyses can have a significant impact on calculated vessel forces that are used in the vessel/internals/fuel dynamic analyses. The WOG Baffle-Barrel-Bolt Program (References 2.7 and 2.8) used a relaxed BOT using the justification in Reference 2.9. The analyses reported herein used a more conservative BOT of 1 millisecond. The more conservative one millisecond BOT was used since it had been anticipated that acceptable results could be achieved and margin associated with relaxed BOT could be retained.

Break Size/Location

The modeled break location in the LOCA hydraulic forces analyses can have a significant impact on calculated vessel forces that are used in the vessel/internals/fuel dynamic analyses. As mentioned earlier, the analyses reported herein considered a break of the limiting branch line in the cold leg, that is, the accumulator line. The limiting Watts Bar Unit 1 cold leg break was modeled as a break at the safe end of the accumulator branch line nozzle located near the connection to the main loop piping. The assumed break size was 0.4176 ft^2 (10" nominal schedule 140 piping), consistent with the Watts Bar Unit 1 loop piping design. The assumed break location was at a point along the cold leg, which is 7.50 feet from the reactor vessel inlet nozzle. This location conservatively represents the Watts Bar Unit 1 most limiting loop in this respect (Loop 1) where the accumulator line connects to the main loop piping at a location 7.6 feet from the reactor vessel inlet nozzle.

RCS Initial Conditions

The assumed RCS initial conditions in the LOCA hydraulic forces analyses can have a significant impact on calculated vessel forces that are used in the vessel/internals/fuel dynamic analyses. For cold leg breaks, the most significant assumption regarding RCS initial conditions is Tcold. Tcold was chosen to be bounding and consistent with those assumed in the WOG Baffle-Barrel-Bolt Program (Reference 2.8). The assumed Tcold was 511.7°F. This Tcold conservatively bounds the Tcold corresponding to the NSSS parameters for Watts Bar Unit 1.

Fuel Type

For the analyses discussed herein, the 17x17 RFA-2 (w/IFMs) and V5H (w/o IFMs) fuel types were analyzed for Watts Bar Unit 1. The homogenous core of 17X17 Robust Fuel Assemblies (RFA-2 w/IFMs) and the limiting mixed cores of the two type of the fuel assembly designs (17x17 RFA-2 w/IFMs and Vantage 5H, w/o IFMs) were evaluated.

For core reload Cycle 7, RFA-2 fuel design was introduced to the Watts Bar core. For core reload Cycle 9, a full core of RFA-2 fuel assemblies are in the Watts Bar Unit 1. The analyses and evaluations performed for RFA-2 (Reference 2.10) supplement the analyses and evaluations presented in Reference 1.1. Fuel integrity for RFA-2, as it pertains to control rod insertion, is assured by the evaluations in support of Reference 2.10. GT integrity and CRDM platform integrity for RFA-2 were also evaluated as part of the RFA-2 licensing submittal (Reference 2.10). As reported in Section 10, the conclusions of the Section 7 CRDM driveline alignment evaluation are unaffected by specific fuel design features and therefore are applicable to the RFA-2 fuel design.

Seismic Response Spectra

Plant specific Safe Shutdown Earthquake (SSE) response spectra in the horizontal and vertical direction at the reactor pressure vessel support elevation and the CRDM seismic platform elevation were used to generate synthesized time history acceleration input for the time history seismic analysis of the reactor vessel/internals/fuel model. The records of a real earthquake, TAFT, are the basis for the synthesized time history accelerations (Reference 2.11). The spectral characteristics of the synthesized time history accelerations are similar to the original TAFT earthquake records. In addition, response spectra corresponding to the synthesized time history acceleration meet the acceptance criteria given in US/NRC Safety Review Plan (SRP) 3.7.1 (Reference 2.12) and the suggested frequency intervals given in Regulatory Guide 1.122 (Reference 2.13).

The enveloped horizontal and vertical response spectra used in the Watts Bar Unit 1 analyses are shown in Figures 1 and 2.

2.3 ACCEPTANCE CRITERIA

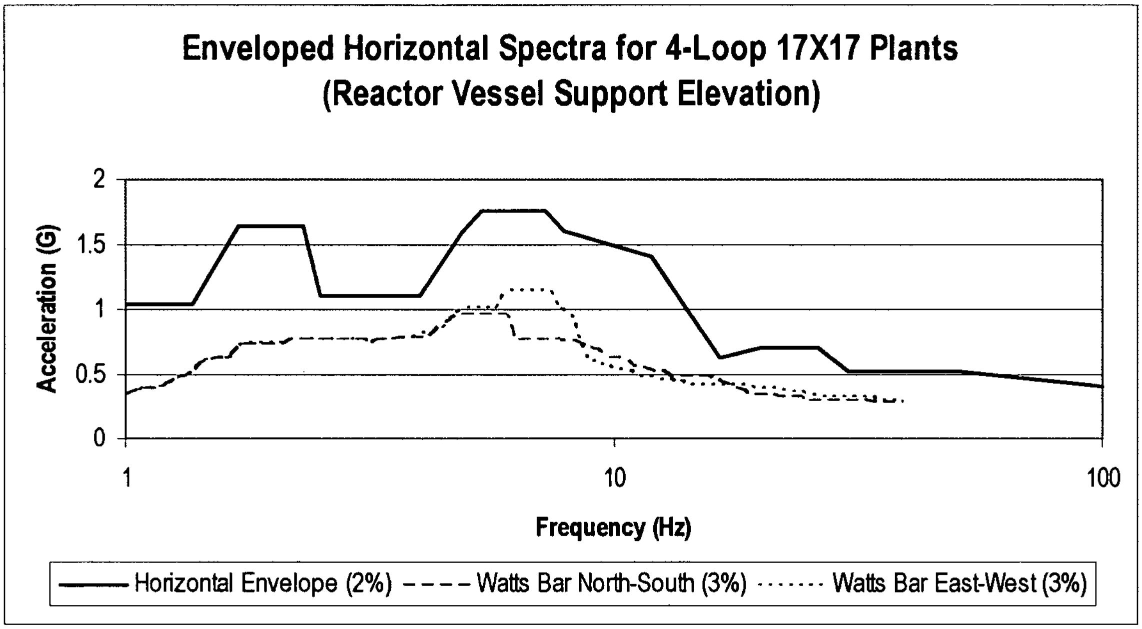
LOCA and seismic loads shall be conservatively combined using the SRSS method to determine the loads for comparison to the acceptable limits. LOCA loads shall be calculated for limiting breaks as determined by application of Leak-Before-Break criteria. Control rod insertability is considered to have been demonstrated if the following criteria are met.

1. RCCA Upper Internals Guide Tube loads calculated for the combined LOCA and SSE shall be less than design allowable values that have been shown to allow control rod insertion.

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.

3. Upper internals driveline displacement due to the combined LOCA and SSE forces shall be limited to values which will not cause driveline binding at the upper internals guide tube/driveline thermal sleeve interface.

4. CRDM support platform assembly performance for the combined LOCA and SSE forces shall meet ASME code allowable stress limits to retain its functional performance, or will otherwise be shown not to inhibit control rod insertion, in the prescribed loading environment.



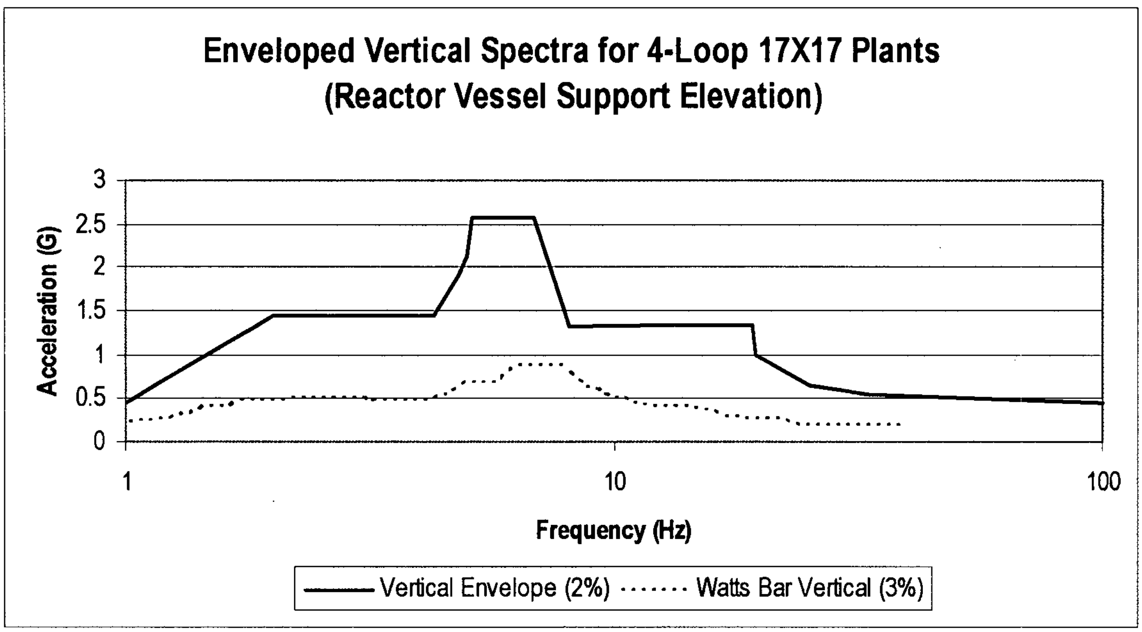
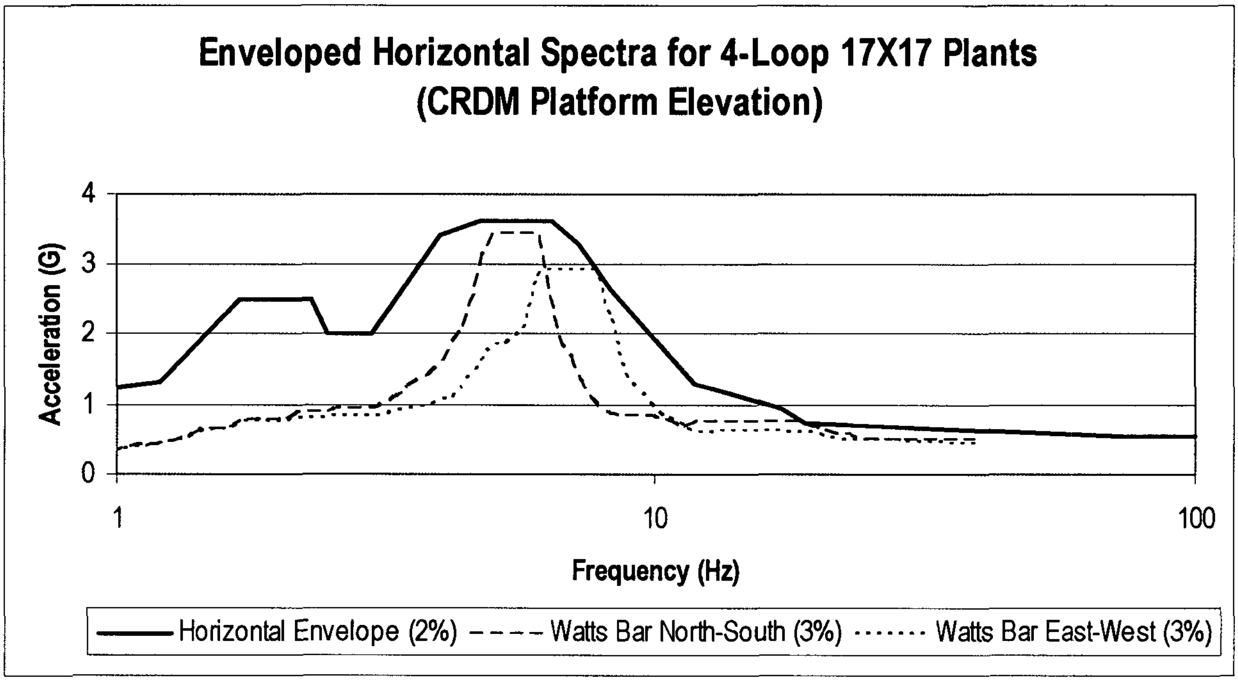


Figure 1 **Enveloped Horizontal and Vertical Spectra (Reactor Vessel Support Elevation) for 4-Loop 17x17 Plants**



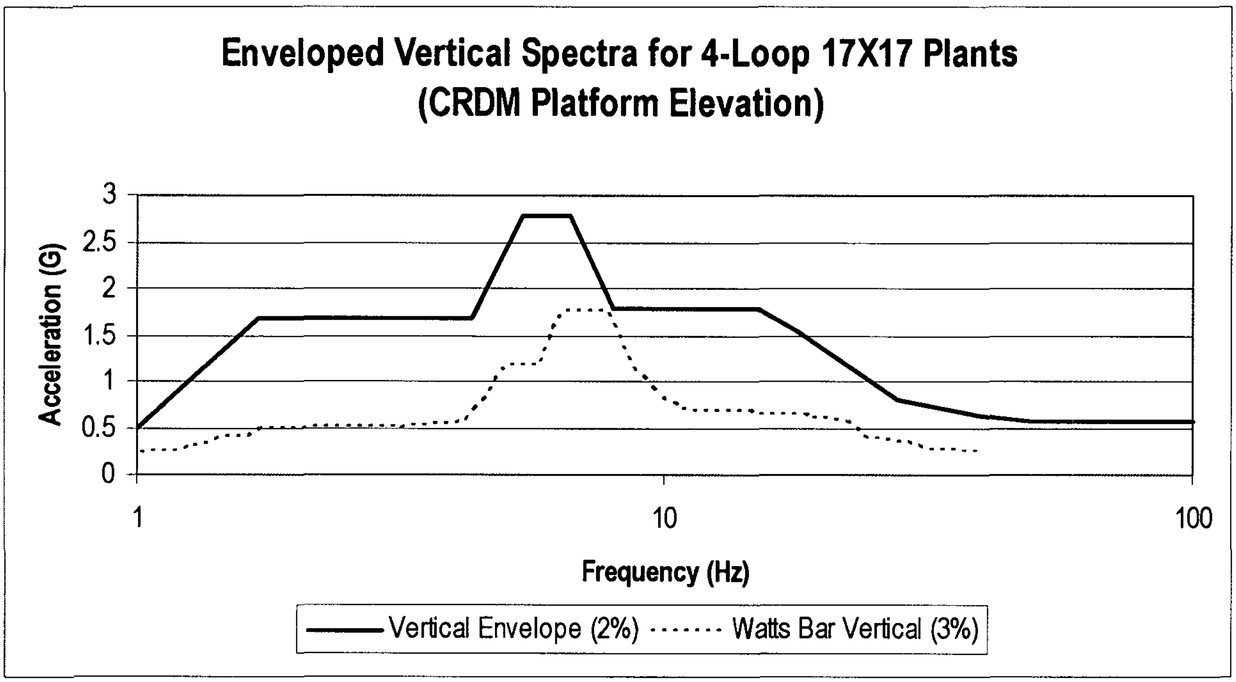


Figure 2 **Enveloped Horizontal and Vertical Spectra (CRDM Seismic Support Platform Elevation) for 4-Loop 17x17 Plants**

3 LOCA HYDRAULIC FORCES

3.1 LOCA HYDRAULIC FORCES ANALYSIS

The LOCA hydraulic forces analysis was performed through use of the MULTIFLEX code (Reference 3.1) previously accepted by the NRC for the WOG Baffle-Barrel-Bolt Program (References 2.7 and 2.8) and the D. C. Cook Control Rod Insertion Program (References 1.2 and 1.3). A 4-loop vessel model representative of Watts Bar Unit 1 was developed using a variation of the appropriated model from the WOG Baffle-Barrel-Bolt Program, Reference 2.7. Where specific differences did exist, e.g., fuel type, vessel support stiffness, etc., bounding values were applied. I.e., High vessel support stiffness is bounding for purposes of the LOCA hydraulic forces calculation, since vessel motion relative to the barrel would reduce the calculated hydraulic forces. As mentioned earlier, a conservative BOT of 1 millisecond was used.

LOCA hydraulic forces calculations were performed for an accumulator line break using assumptions indicated in Section 2. The results of the LOCA hydraulic forces calculations were used as input to the vessel/internals/fuel analyses and as input to the RCCA guide tube analyses. For the vessel/internals/fuel analyses, the LOCA hydraulic forces input consisted of electronic datasets of calculated horizontal and vertical forces at select points throughout the reactor vessel. For the RCCA guide tube analyses, the LOCA hydraulic forces input consisted of datasets of calculated flow-induced drag loads and acoustic wave loads on the limiting guide tube location.

3.2 ANALYSIS RESULTS

LOCA hydraulic forces analysis results applicable to Watts Bar Unit 1 are provided in Figure 3. The X direction refers to an axis coincident with the centerline of the broken leg. This is typically the direction of the maximum horizontal LOCA hydraulic forces. Figure 3 demonstrates that the peak core barrel horizontal forces occur in the first 0.040 seconds after the pipe break. This is also true for RCCA guide tube acoustic wave loads. Peak guide tube flow-induced drag forces (discussed in Section 5.1) occur somewhat later, typically at 0.080 seconds.

a,c

Figure 3 60 in^2 Accumulator Line Break Horizontal Forces on Core Barrel X Direction, High Vessel Support Stiffness

4 VESSEL/INTERNALS/FUEL DYNAMIC ANALYSES

4.1 MATHEMATICAL SYSTEM MODEL OF VESSEL/INTERNALS/FUEL

The vessel/internals/fuel system finite element model consisted of three concentric structural sub-models connected by nonlinear impact elements and linear stiffness matrices. The first sub-model, shown in Figure 4, represents the reactor vessel shell and associated support components. The reactor vessel of a 4-loop plant reactor vessel is restrained by four vessel supports located underneath alternate nozzles and the attached primary coolant piping.

The second sub-model, shown in Figure 5, represents the reactor core barrel assembly (thermal shield or neutron pad assembly), baffle-former assembly, lower support plate, tie plates and the secondary core support structure etc. These sub-models are physically located inside the first, and are connected by stiffness matrices at the vessel internals support ledges. Core barrel to reactor vessel shell impact is represented by nonlinear elements at the core barrel flange, core barrel outlet nozzles, and the lower radial restraints.

The third and innermost sub-model, shown in Figure 6, represents the upper support plate assembly consisting of guide tubes and upper support columns, upper and lower core plates, and the fuel. The third sub-model is connected to the first and second by stiffness matrices and nonlinear elements. Figure 7 shows the assembled representation of the vessel/internals/fuel system model.

Attached to the vessel head (Node 1) is the standard CRDM seismic support platform as shown in Figure 8.

4.2 ANALYSIS METHODOLOGY

The mathematical system model of the vessel/internals/fuel is a three-dimensional nonlinear finite element model that is described above. The only difference between the seismic and LOCA model is that, in the seismic model, fluid-solid interactions are represented by hydrodynamic mass matrices in the downcomer region (between the core barrel and reactor vessel). On the other hand, in LOCA analysis, the fluid-solid interactions are accounted for through the hydraulic forcing functions generated by the MULTIFLEX code, Reference 3.1.

The WECAN/PLUS computer code, a general purpose finite element code, is used to determine the dynamic responses of the vessel/internals/fuel system model subjected to LOCA and seismic loading using non-linear modal superposition method. The NRC-approved methodology established in Reference 4.1 is used in the dynamic analysis of the vessel/internals/fuel model. This methodology is the same as used and approved for the D. C. Cook analysis, References 1.2 and 1.3. A flow chart for the LOCA/seismic dynamic analysis is shown in Figure 9.

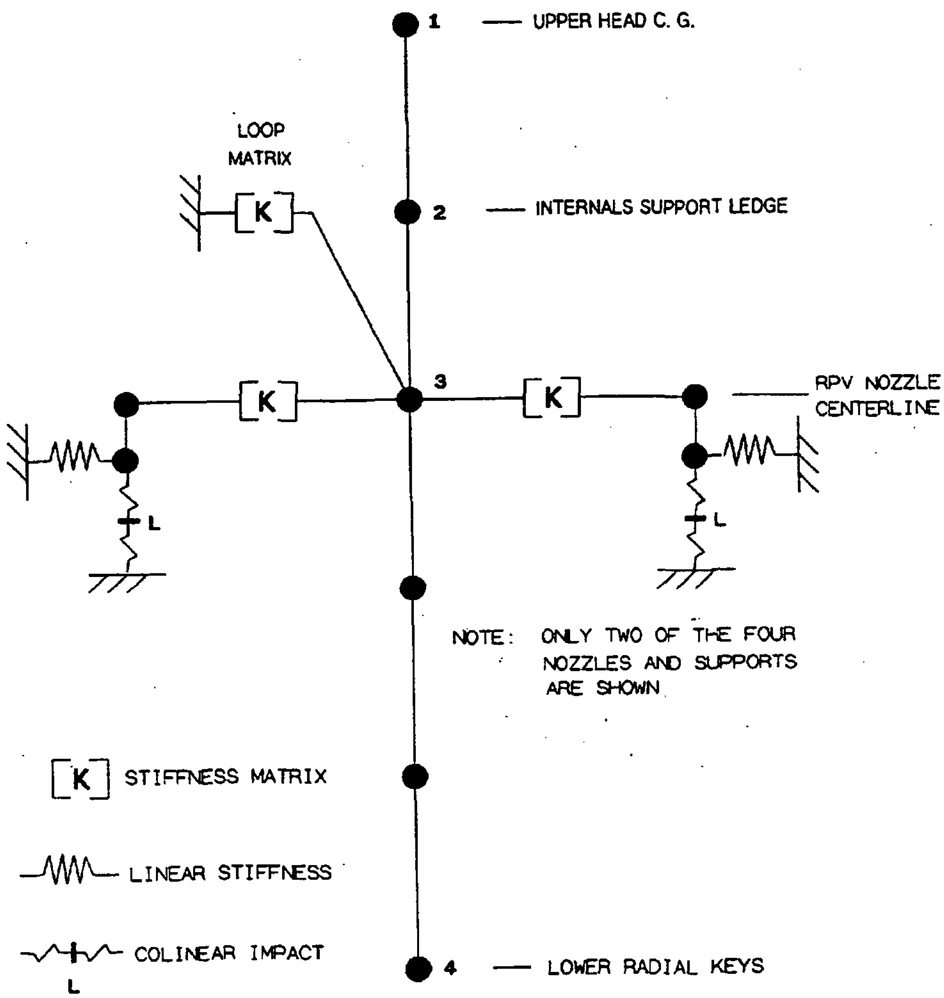


Figure 4 Reactor Pressure Vessel Shell and Typically Supported Nozzles

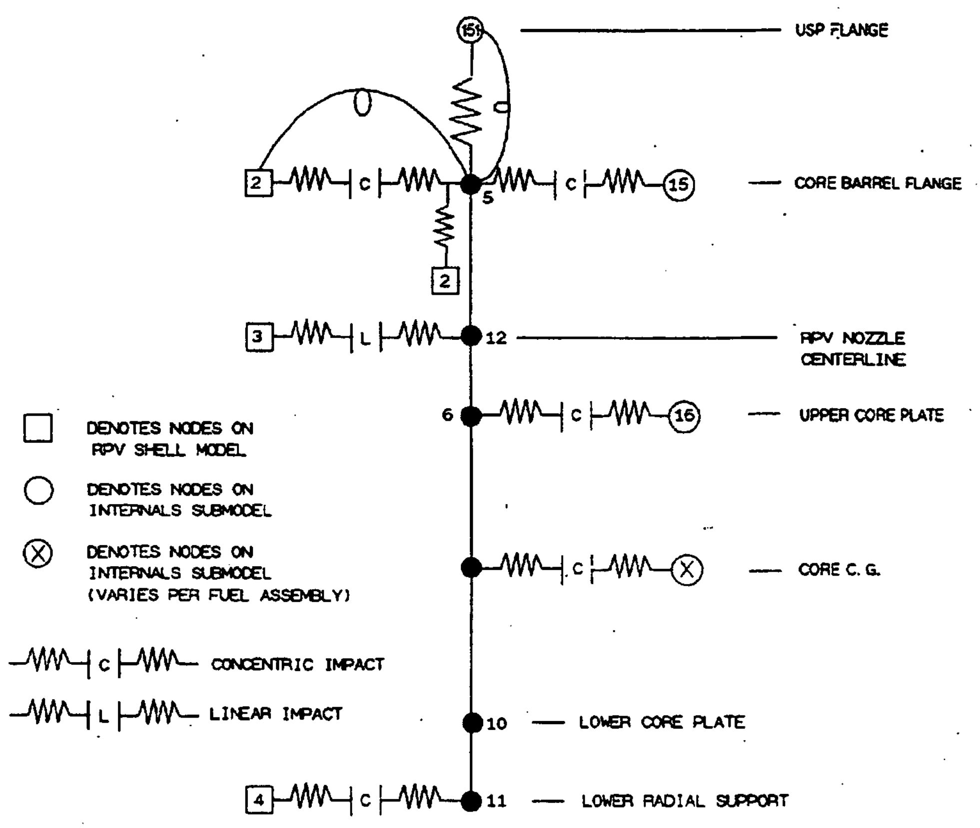


Figure 5 Core Barrel Sub-Model

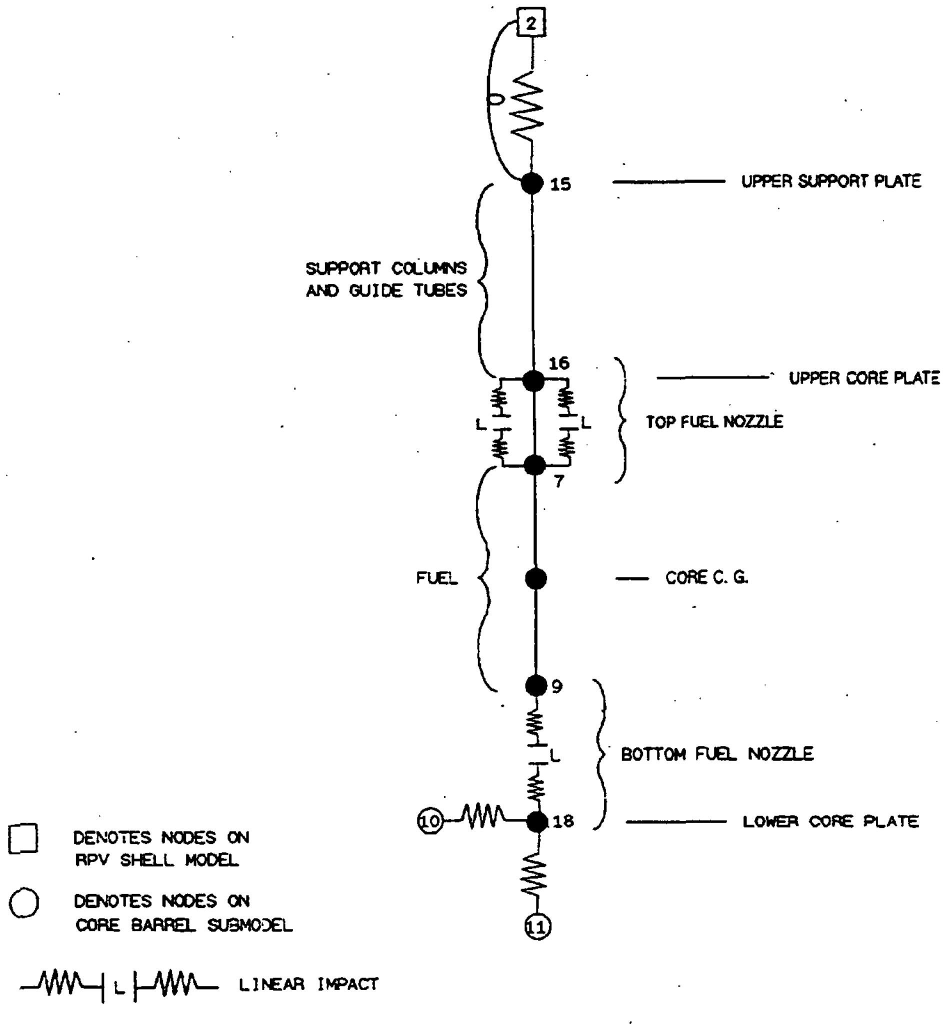


Figure 6 Internals (Innermost) Sub-Model

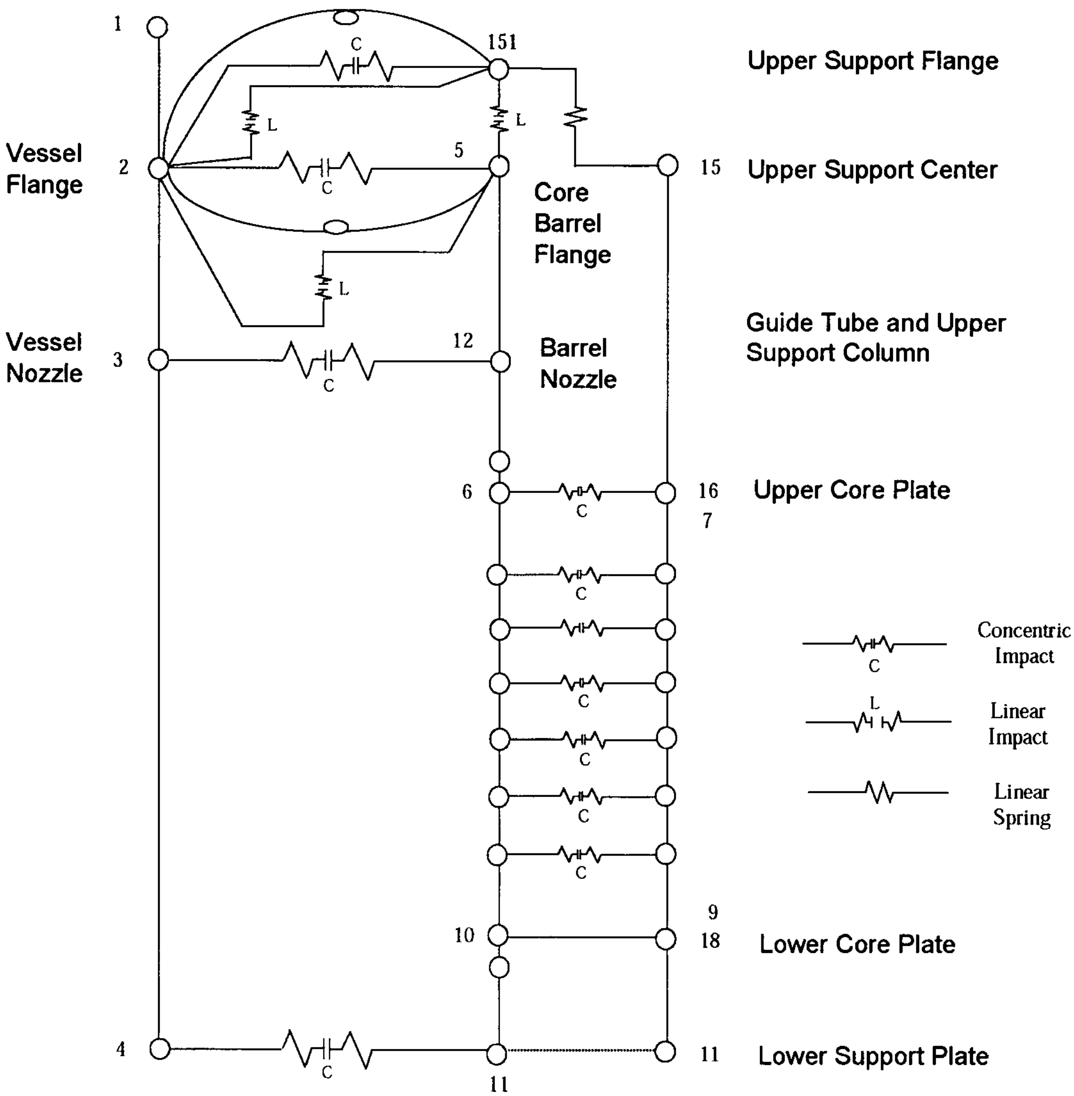


Figure 7 Vessel/Internals/Fuel Assembly Model

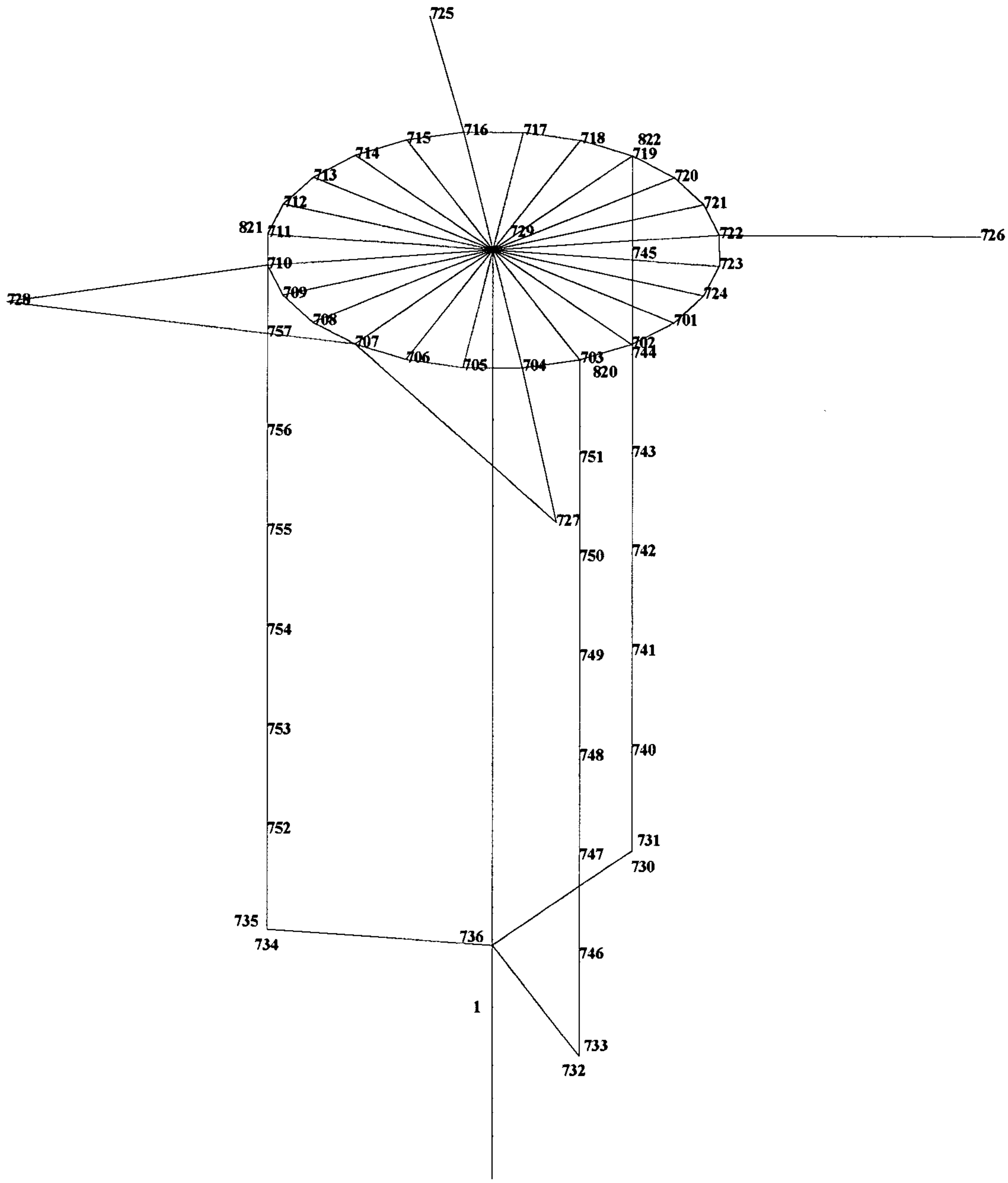


Figure 8 Standard CRDM Seismic Support Platform Model

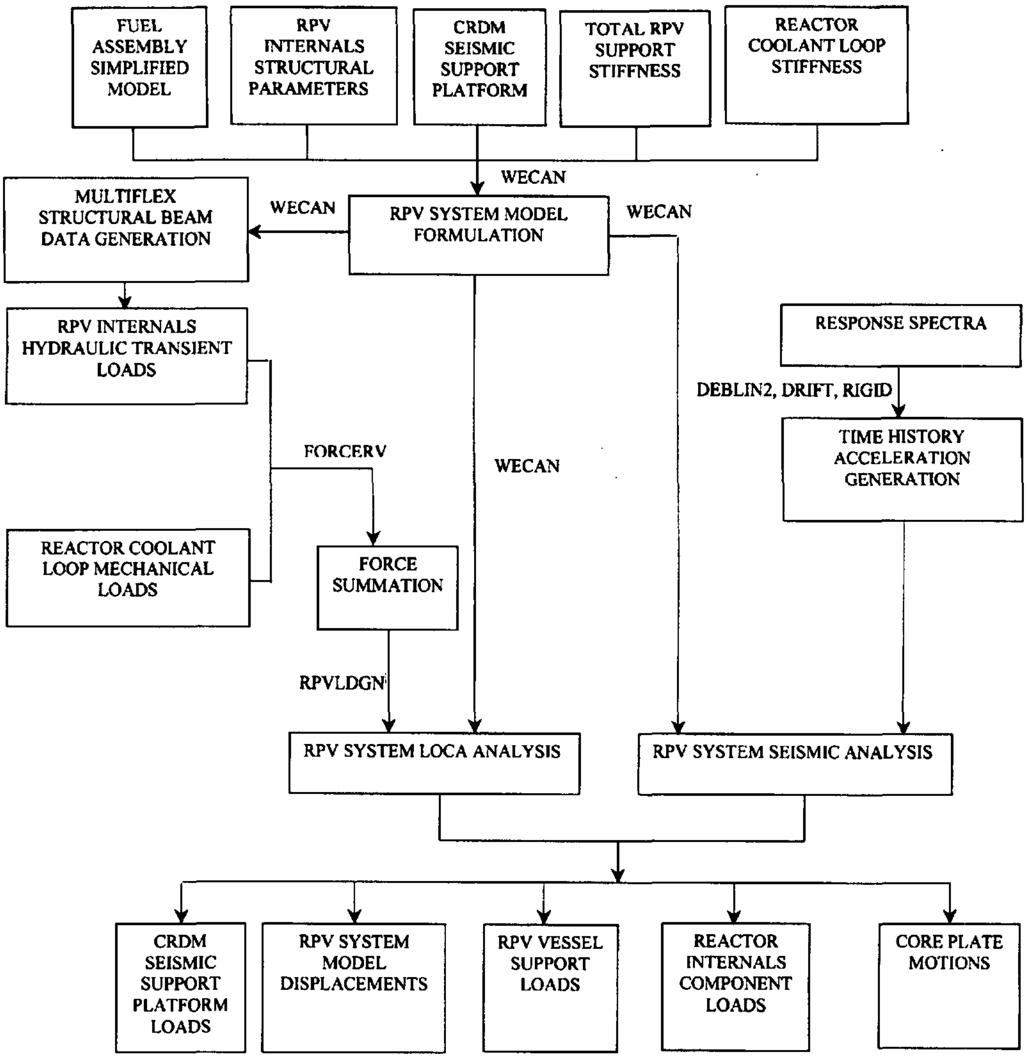


Figure 9 LOCA/Seismic Dynamic Analysis Flowchart

In addition to the WECAN/PLUS Computer Code, descriptions of other major codes such as FORCERV, RPVLDGEN, DEBLIN2, DRIFT AND RIGID are given below.

FORCERV code combines the horizontal and vertical hydraulic forces generated by the MULTIFLEX code with the loop release loads and cavity pressurization forces (if applicable) into a single time history forcing function.

RPVLDGEN code uses the single time history forcing function generated by FORCERV to create the corresponding WECAN/PLUS 9.0 series loading card image for the LOCA analysis.

DEBLIN2 computer code generates a synthesized time history acceleration from the design basis seismic response spectra by modifying the time history acceleration from the records of a real earthquake.

DRIFT code eliminates the drift from the time history displacement generated by the DEBLIN2 Code; and RIGID code determines the ground motion (displacement) from the synthesized time history acceleration generated by the DEBLIN2 code.

The development of LOCA hydraulic forces input to the model is discussed in Section 3 and the seismic loading input to the model is discussed in Section 2.

The results of the vessel/internals/fuel dynamic analysis include time history displacements and impact forces for all the major vessel internal components. The time history displacements are provided as input for the fuel assembly analysis and guide tube evaluation. The resulting loads on the CRDM seismic support assembly are used as input to the qualification of the lift legs, tie rods and the CRDM seismic platform components.

4.3 CORE PLATE MOTIONS

The 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 core plate motions were developed for the accumulator line break and bounding plant group seismic spectra and were provided to Nuclear Fuel Business Unit (NFBU) for their evaluation. The methodology used to perform these calculations is consistent with the methodology used in the studies associated with transition to Vantage 5 fuel, as documented in Reference 4.2 for example.

To address the variation in time phasing of the grid impact loads (LOCA versus seismic), the effects of LOCA and seismic core plate motions are calculated separately. The resulting peak grid impact forces from the fuel assembly analysis are then combined using the SRSS method.

4.4 EFFECT OF FUEL PROPERTIES ON VESSEL/INTERNALS/FUEL DYNAMIC ANALYSES

Calculated fuel grid loads are directly dependent on the vessel/internals/fuel assembly model and the fuel product that the model represents. Conversely, it is expected that the qualification of the guide tubes and the qualification of the CRDM platform assembly would not be adversely impacted by the specific fuel assembly model. Confirmatory sensitivity studies were performed to evaluate the effect of the differences in fuel design properties on GT and CRDM platform assembly loads. Using the limiting 3-loop analyses, three additional dissimilar fuel type variations were analyzed. In addition to V5H, the fuel variations included Vantage 5 (thin rod) with IFMs, RFA (fat rods with IFMs) and a representative fictitious non-Westinghouse fuel without IFMs.

Comparisons of the results in all four cases demonstrate that large differences in fuel properties that were considered in this investigation do not affect the calculated GT loads to an extent that will challenge the conclusions of the reported analyses. Therefore, the conclusions, relative to GT loads, can be extended to all typical Westinghouse or non-Westinghouse fuel design.

Similar results were seen with the calculated CRDM platform assembly loads. For each of the fuel variations considered, the calculated limiting CRDM platform assembly loads do not adversely impact the conclusion of the reported analysis. Furthermore, in cases where CRDM platform assembly load margins are relatively small, the inherent conservatisms of the bounding assumptions (plant design, break size/location, BOT, RCS initial conditions, enveloped seismic response spectra) provide sufficient margin to offset the effects of the differences in fuel properties. Therefore, it can be concluded that the conclusions of the bounding analyses, relative to the standard CRDM platform assembly loads, apply to any typical Westinghouse or non-Westinghouse fuel design.

4.5 EFFECT OF CRDM PLATFORM DESIGN ON VESSEL/INTERNALS/FUEL DYNAMIC ANALYSES

It is expected that the qualification of the fuel grid and GT loads will not be adversely impacted by the specific CRDM platform assembly design. Accordingly, confirmatory sensitivity studies were performed to evaluate the effect of the differences in CRDM platform assembly design on fuel grid and GT loads. Using a 3-loop analysis as a basis, variations in CRDM platform assembly modeling were analyzed. Comparisons of the results demonstrate that specific CRDM platform assembly modeling does not affect the calculated GT and fuel grid loads to an extent that will challenge the conclusions of the bounding analyses. Therefore, it can be concluded that the conclusions of the bounding analyses, relative to fuel grid and guide tube loads, apply to the Watts Bar CRDM platform design.

5 RCCA UPPER INTERNALS GUIDE TUBE INTEGRITY

5.1 RCCA UPPER INTERNALS GUIDE TUBE LOADS

Integrity of the upper internals guide tubes (as opposed to the fuel assembly guide tubes, or thimble tubes) is fundamental to successful control rod insertion. The total RCCA guide tube loading is a combination of seismic loads and a set of loads occurring during the LOCA transient. Three separate LOCA loads are calculated and combined to determine the total LOCA contribution to the combined loading.

- Hydraulic Flow Induced 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.

- System Loads (Inertial Accelerations)

  These loads result from the dynamic response of the vessel/internals/fuel model subjected to the vessel depressurization loads. They are determined by applying the LOCA forces obtained from the MULTIFLEX code to the WECAN vessel/internals/fuel model. These loads are not sensitive to the guide tube location, but are sensitive to break area, break location, and break opening time.

- 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 differential pressure, the effective guide tube area, and a dynamic load factor.

5.2 RCCA UPPER INTERNALS GUIDE TUBE LOAD COMBINATION

The hydraulic flow-induced loads and the acoustic loads described above originate from the same MULTIFLEX code calculation and thus it is appropriate to algebraically combine these loads as a function of time to obtain a peak combined load. This combined load is then algebraically added to the peak LOCA system load from the vessel/internals/fuel LOCA dynamic analysis. The resulting peak total LOCA load obtained in this fashion can then be combined with the peak seismic load on the guide tube using the SRSS method to obtain the total GT load.

5.3 ALLOWABLE LOADS FOR RCCA UPPER INTERNALS 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, Westinghouse has previously performed guide tube scram tests 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 5.1 and 5.2. Scram tests were conducted with full size guide tubes, with rod control clusters inside and fuel assemblies below. The guide tubes were mechanically loaded at four discrete elevations, illustrated in Figure 21 of Reference 5.1 for the 17x17-96" guide tubes which are in Watts Bar Unit 1, 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, was 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.

From the scram test results, the allowable loads, based on permanent loss of function, for the different styles of guide tubes is shown in Table 2. The 17x17-96 style of guide tubes have the largest allowable load and are the least critical of the three. Hence, bounding analyses were carried out using both the 15x15-150 and the 17x17-150 guide tubes which are the weaker than the 17x17-96 guide tubes

Table 2 Allowable Loads Determined from Scram Tests

Guide Tube Style	Allowable Load (lb)
17x17-96	51,000
17x17-150	29,600
15x15-150	10,000

5.4 ANALYSIS RESULTS

The results of the 4-loop bounding analyses are provided in Table 3. 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 LOCA+Seismic conditions. Thus, GT loads for breaks in the cold leg are well within the allowable limits and will in no way inhibit control rod insertion for the Watts Bar Unit 1 plant design.

a,c

Table 3	Reactor Internals Guide Tube Analyses Results	

As discussed in Section 4.4, these conclusions can be extended to all typical Westinghouse and non-Westinghouse fuel designs. This position is based on the large margins to acceptability and the fuel sensitivity studies which show that differences in fuel properties do not adversely affect the qualification of the calculated GT loads.

6 FUEL ASSEMBLY INTEGRITY

6.1 FUEL ASSEMBLY ANALYSES

The general analytical procedure for evaluating fuel assembly transient response to seismic and LOCA transients is shown schematically in Figure 10. 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 location of the RPV model. For the seismic analysis, the plant-specific design acceleration response spectra are specified based upon the plant site characteristics. For the analyses reported herein, the synthesized seismic time history accelerations used in the analysis are generated from the enveloped response spectra for the plants within a given plant group as discussed in Section 2. The LOCA hydraulic forcing functions and the seismic time-history accelerations are used as loading input to the dynamic analysis of the vessel/internals/fuel system model. The core plate and core barrel motions from the dynamic analysis of this model are obtained and then used as input to the Reactor Core Model.

The Reactor Core Model includes a number of fuel assembly array models (four for 4-loop cores) with varying row lengths and inter-assembly grid impact elements. A schematic of a typical reactor core array model is shown in Figure 11. The numbers of fuel assemblies in the array models are 7, 11, 13 and 15, which represent the number of fuel assemblies in each of the core planar arrays. In Figure 11, a total of 15 fuel assemblies represent 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 the different models (rows) in the X and Z directions, i.e., parallel to the reactor vessel horizontal cardinal axes. The seismic and LOCA analyses were performed for every array of fuel assemblies that are beneath CRDMs.

The maximum grid loads, obtained from SSE and LOCA loading analyses, are combined as required using the SRSS method. The results of the seismic and LOCA analyses of the maximum impact forces are compared to allowable grid distortion loads. 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.

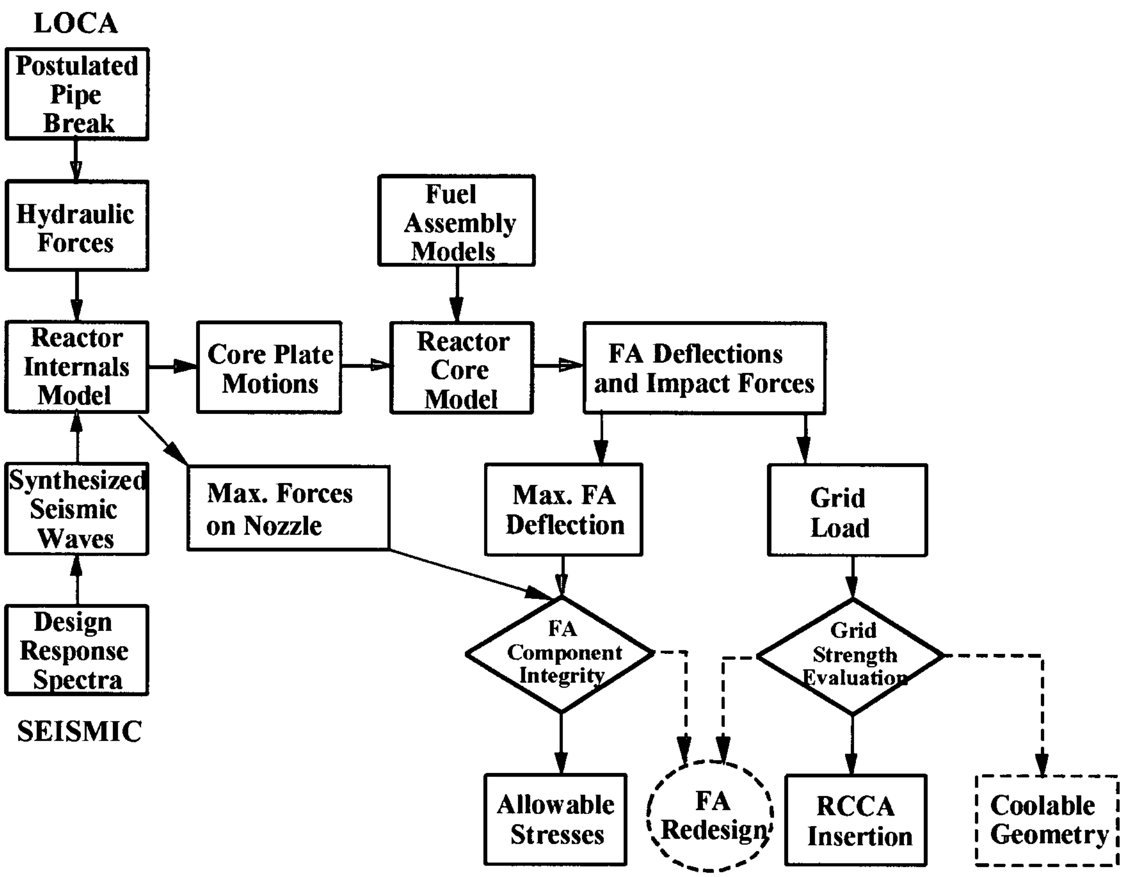


Figure 10 Seismic/LOCA Fuel Assembly Analysis Flow Chart

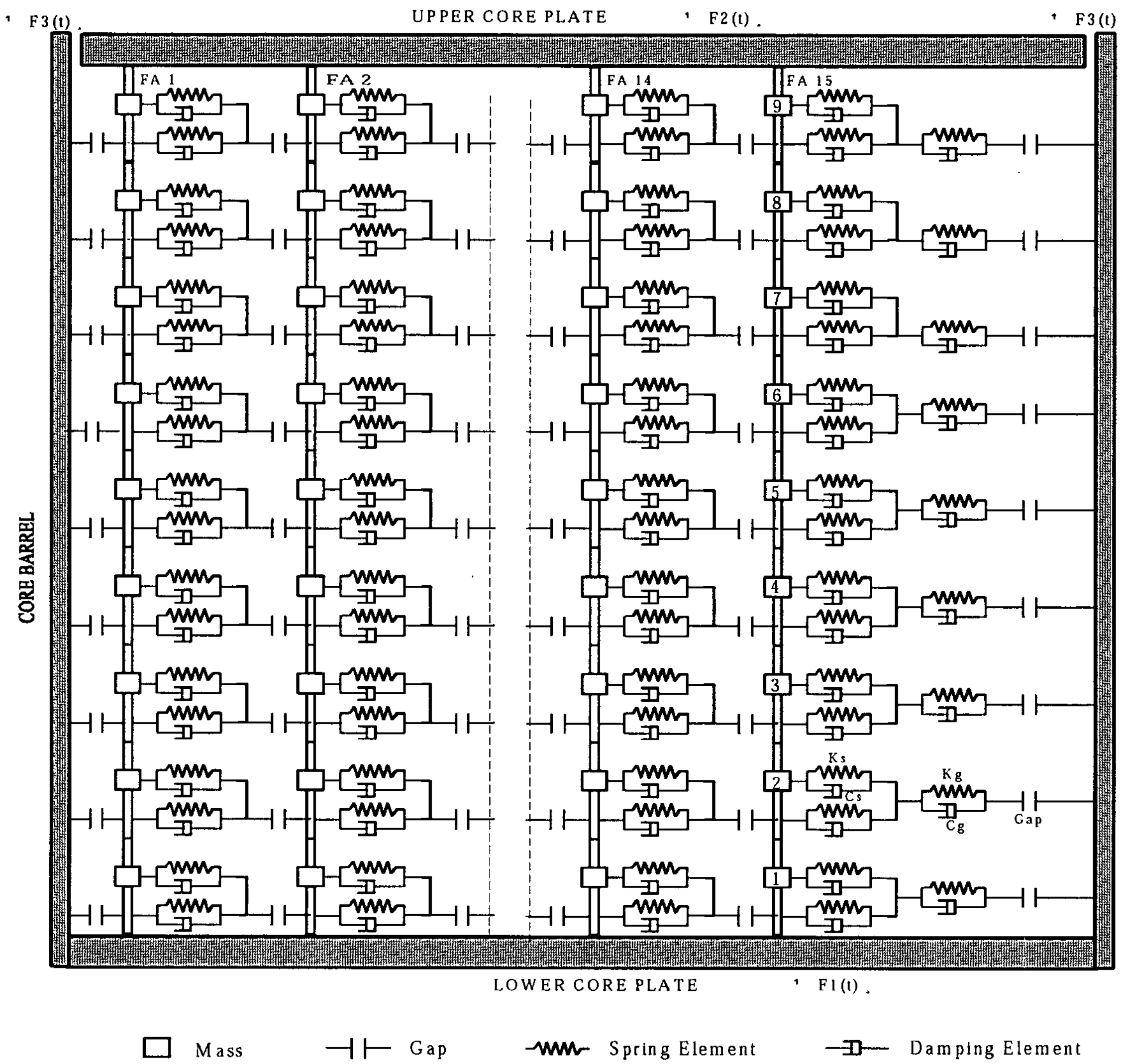


Figure 11 **Reactor Core Model for 17x17 RFA-2 Fuel Assembly Dynamic Analysis Row of 15 Fuel Assembly Array**

6.2 ALLOWABLE LOADS FOR FUEL GRIDS

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. The Westinghouse approach to evaluating fuel assembly grid strength is based upon the requirements defined in NUREG-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(critical) should be the 95% confidence level on the true mean as taken from the distribution of measurements on unirradiated production grids at (or corrected to) operating temperature. While P(critical) will increase with irradiation, ductility will be reduced. The extra margin in P(critical) for irradiated grids is thus assumed to offset the unknown deformation behavior of irradiated grids beyond P(critical)."

Similarly, the Westinghouse approach to evaluating other fuel assembly components is based upon the requirements defined in NUREG-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.

6.3 ANALYSIS RESULTS

The LOCA and seismic fuel grid impact loads for the Watts Bar Unit 1 bounding analyses are presented in Table 4. The maximum fuel grid loads, obtained from SSE and LOCA loading analyses, were combined as required using the SRSS method for each fuel grid. Note that in some cases, the maximum seismic-induced fuel grid load and the maximum LOCA-induced fuel grid load are at different elevations. The results of the seismic and LOCA analyses of the maximum impact forces for the structural grids are compared to allowable grid distortion loads. In all cases, no fuel assembly grid distortion was calculated for any fuel assembly, regardless of location and as a result, it was not necessary to consider the specific RCCA locations. Thus, it can be concluded that distorted fuel would not impede control rod insertion.

Table 4 17x17 RFA-2 Fuel Assembly Grid Distortion Analyses Results		
Grid Impact Forces	**Mid Grid**	**IFM Grid**

a,c

The fuel assembly stress analyses were performed for the most limiting (SSE and LOCA combined) cases. The thimble tube and fuel rod stresses for the combined maximum SSE and LOCA fuel assembly lateral deflection with the vertical LOCA impact load at the bottom nozzle were calculated. The analysis results applicable for Watts Bar Unit 1 show that the thimble tube and fuel rod stresses are less than the allowable values for all the cases.

7 CONTROL ROD DRIVELINE INTEGRITY

Lateral displacement of the reactor vessel upper internals with respect to the reactor vessel, resulting from LOCA and seismic forces, could theoretically cause a distortion of the control rod driveline alignment that might affect the ability to insert the control rods. This might occur as the driveline 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. See Figure 12 for a typical driveline layout of this area. 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. For the bounding designs at the end of blowdown (30 to 60 seconds into the LOCA transient), these gaps are 0.22 inches and 0.23 inches respectively. 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 motion of the upper internals package is assumed equal to the limiting vessel to upper support plate gap width at normal operating temperature.

The above lateral displacement 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 must be evaluated to address the transient temperature and thermal contraction of the reactor upper internals following the LOCA transient. The lack of cooling on the upper head surfaces and the potential for thermal distortion due to uneven cooldown of the reactor vessel head, internals, and vessel could impact the ability to insert the control rods. The insertion of control rods would occur during the LOCA transient or when the seismic and LOCA hydraulic forces have decreased following the accident. LOCA hydraulic forces peak within the first second of the break and are effectively nil 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 large cold leg break LOCAs, 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 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, combined with the vessel head assumed to remain at the operating temperature, provides a basis for the thermal misalignment evaluation. This thermal driveline misalignment is based upon the following assumptions:

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

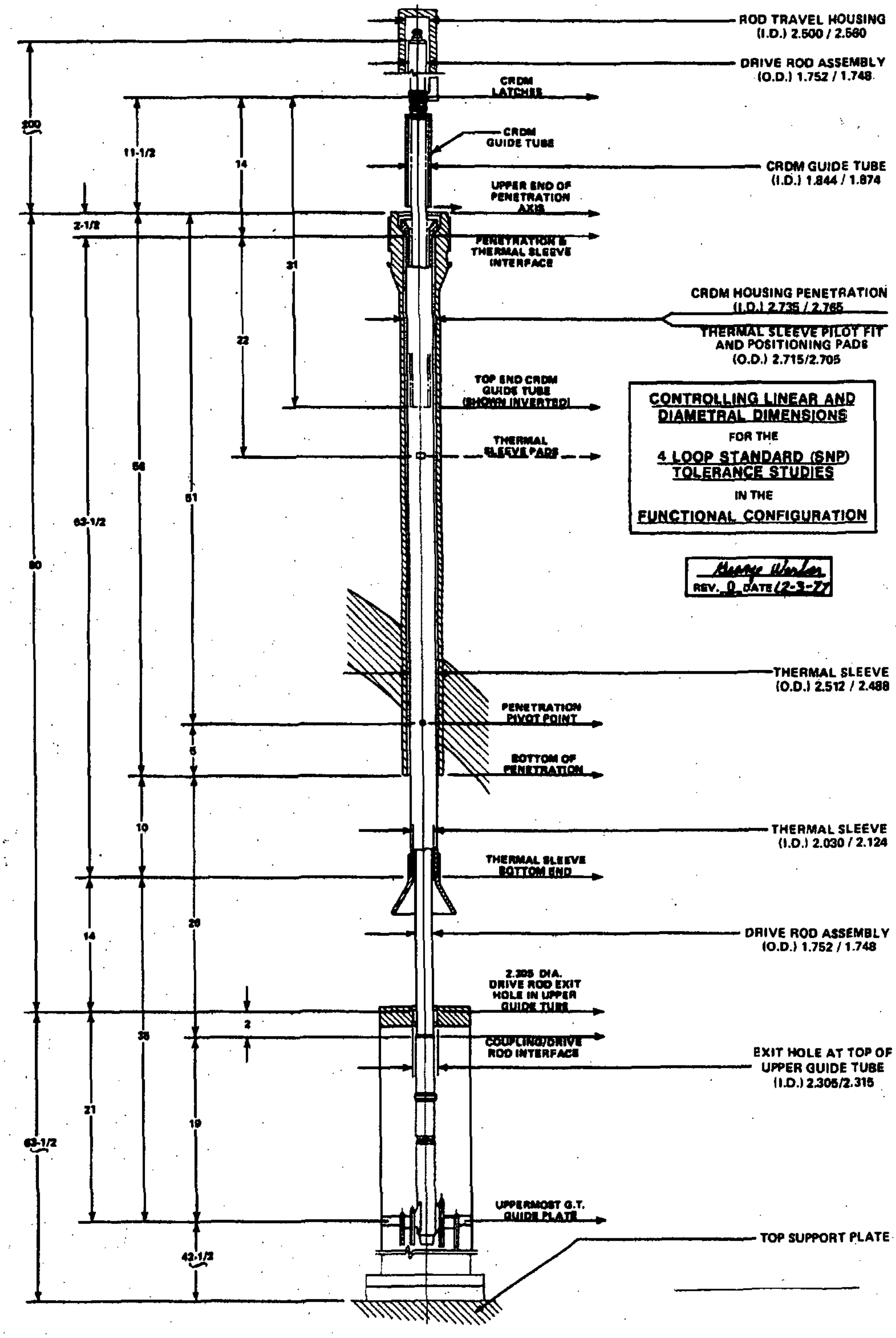


Figure 12 Typical CRDM Driveline Layout

2. The upper internals package is assumed to be "pinned" at the point where the upper internals support structure flange and reactor vessel surfaces are in contact.

3. 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.

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

5. Thermal and structural behavior of upper support structure is controlled by the thermal response of the upper support plate. Thermal and structural effects of any stiffening ligaments beneath the upper support plate may be neglected.

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

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

With the release of the 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 through the vessel head penetration from the top of the penetration to within a few inches of the top of the guide tube, Figure 12. 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 ID of the vessel head penetration and the OD 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.

Combining the bounding lateral upper package displacement with the conservative differential thermal contraction between the vessel and upper internals support structure shows that the driveline misalignment at the limiting guide tube location is less than the allowable value for a period in excess of [ ]^{a,c,e}. For the period up through [ ]^{a,c}, other guide tube locations, which are further from the point where the vessel support ledge and upper support package are in contact, have better driveline alignment because of the compensating effect of the support plate having more thermal expansion than the vessel head. Thus, the calculated bounding value for upper internals displacement plus the thermal effects can be accommodated by the allowable lateral displacement. It can be concluded that sufficient drive rod clearances are available for post-LOCA 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.

The CRDM pressure housing (latch housing and rod travel housing) and the CRDM head adapter form the CRDM driveline pressure boundary external to the reactor vessel. The structural integrity of these components is demonstrated by showing that the combined maximum SSE and LOCA loads are less than allowable loads that ensure ASME Code Section III NB allowable limits are satisfied. Details of the evaluation are provided in Section 8.3.

8 CRDM SEISMIC SUPPORT PLATFORM ASSEMBLY INTEGRITY

8.1 CRDM SEISMIC SUPPORT PLATFORM ASSEMBLY ANALYSIS

The CRDM Seismic Support Platform Assembly provides lateral support to the CRDM rod travel housings at their upper end through a set of seismic spacer plates installed on each of the CRDMs and a surrounding platform structure to restrain the motion of the plates. Gaps are provided between the platform and the plates and between each of the plates to minimize interaction during normal operation and design transient conditions. During seismic conditions, lateral motion of the rod travel housings is restricted to displacements that will preclude non-elastic deformation of the housings. The seismic support platform is supported vertically from the reactor vessel head by the head lift legs and supported laterally by a set of tie rods connected to the reactor cavity wall. Figure 13 illustrates a CRDM seismic support platform typical of the Watts Bar Unit 1 design.

The analysis of the CRDM Seismic Support Platform Assembly is being included in this program for two reasons:

1. If the lift leg buckling is not precluded for LOCA plus seismic loads, it must be shown that there is no plastic deformation of the CRDM rod travel housings or that the plastic deformation is insufficient to inhibit rod insertion. Demonstration of the adequacy of the platform assembly to perform its intended function was considered the preferred approach.

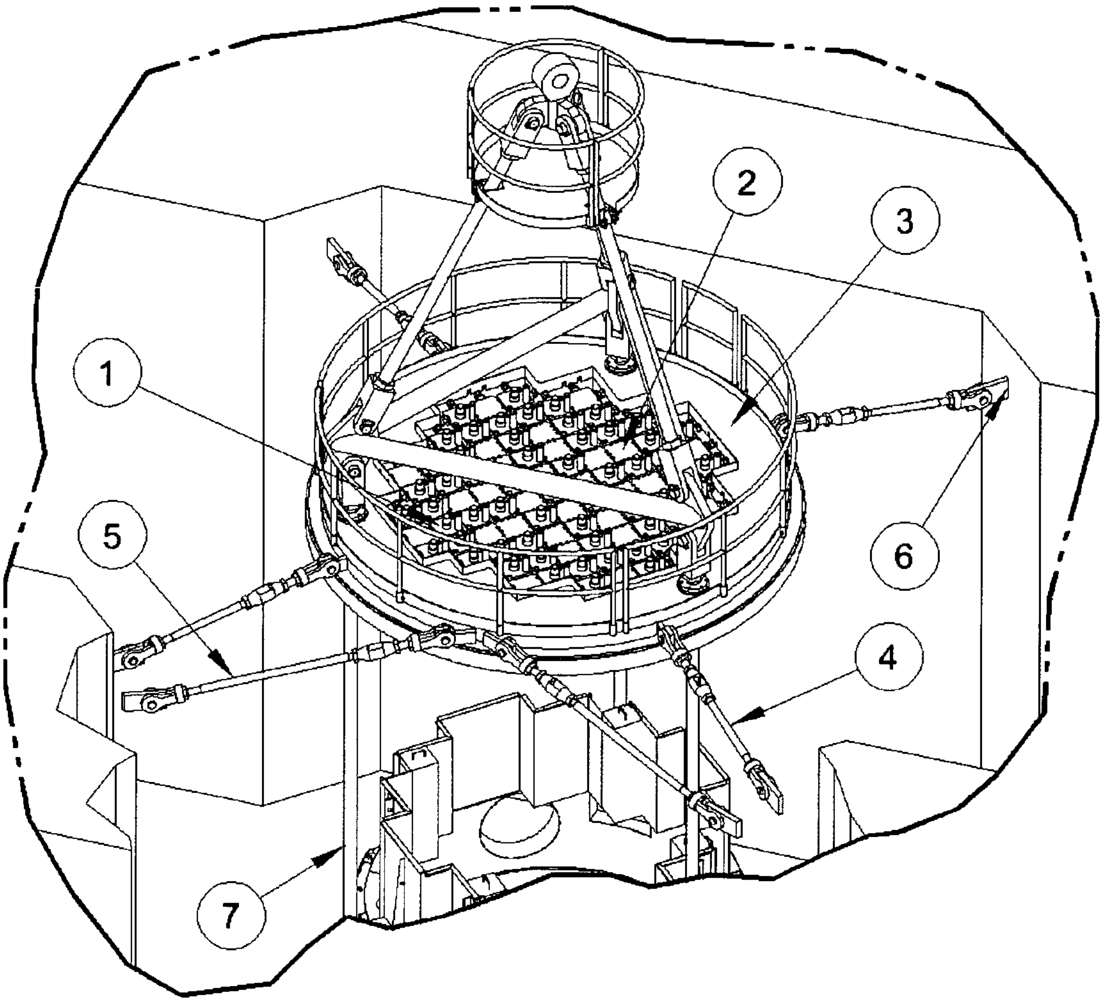
2. Analyses of the original standard platform assemblies are late 1960s and early 1970s vintage and were performed with methods, codes, and computer systems that may now be considered superseded for any new CRDM analysis work.

8.2 ANALYSIS RESULTS

The WECAN reactor vessel system model includes the standard CRDM support platform assembly with the vessel head lift legs, the platform tie rods, the basic platform, and the restraints for the CRDM rod travel housings. Gap elements were used, as appropriate, for the interface between the seismic platform and the CRDM housings.

The mathematical representation of the CRDM support platform assembly considers three major parts of the support structure – the circular support frame, the tie rod assemblies, and the lift legs. Figure 8 shows the finite element representation of the standard seismic support platform assembly. The seismic support platform model is connected to the reactor vessel head (Node 1). As described in Section 4.2, WECAN/PLUS computer code is used to determine the response of the entire system model subjected to LOCA and seismic loading using non-linear modal superposition method. Results of the system LOCA and seismic analyses include time history nodal displacements and non-linear impact forces for all major component interfaces. Component linear forces and moments of the CRDM seismic support platform are also obtained. The platform support frames, tie rods, and plant lift legs were qualified to ASME NF and Appendix F criteria using enveloped maximum loads.

SEISMIC SUPPORT ASSEMBLY



Item 1: RPI Top Plate
Item 2: Spacer Plate
Item 3: Seismic Support Platform
Item 4: Radial Tie Rod Assembly
Item 5: Tangential Tie Rod Assembly
Item 6: Containment Wall Lug Embedment
Item 7: Head Lifting Leg

Figure 13 Seismic Support Assembly

As shown in Table 5, the CRDM supports for Watts Bar Unit 1 have been shown to be acceptable for the Faulted loading condition. As a consequence, the CRDM support platform assemblies will be fully functional in the postulated environment and will provide the intended lateral support of the CRDM rod travel housings.

Table 5 CRDM Seismic Platform Assembly Analyses Results

Radial Tie Rod Assemblies		a,c
Calculated Stress (ksi)		
Allowable Stress (ksi)		
Interaction Ratio(1)		
Tangential Tie Rods		
Calculated Stress (ksi)		
Allowable Stress (ksi)		
Interaction Ratio(1)		
Platform Weldment		
Calculated Stress Intensity (ksi)		
Allowable Stress Intensity (ksi)		
Interaction Ratio(1)		
Lift Legs		
Calculated Stress, Axial (ksi)		
Calculated Stress, Bending (ksi)		
Allowable Stress		
Interaction Ratio(1)		

Note:
1. ASME Boiler & Pressure Vessel Code, Section III, Nuclear Power Plant Components, Subsection NF, Component Supports, 1974 Edition, Appendix VXII-2215.1 with Appendix F.

8.3 CRDM PRESSURE BOUNDARY AND HEAD ADAPTER INTEGRITY

For the CRDM pressure housing the combined SSE and LOCA (faulted) structural evaluation, the faulted bending moments are compared to allowable limits along the entire length of the CRDM pressure housings. The maximum percentage of allowable is 90% along the rod travel housing. As explained in Section 2.2 of this report, LOCA loads are now based on LBB methodology. The analysis of record for Watts Bar Unit 1 is based on main loop breaks. LBB methodology results in LOCA loads that are less conservative than the main loop pipe break. Since the LOCA loads considered for this WCAP are bounded by the main loop break loads used in the current analysis, the CRDM pressure housing structural integrity is maintained for the control rod insertion program.

NSAL-07-3 (Reference 1.4) identified under-estimation in the faulted loads used to qualify CRDM head adapters and CRDM pressure housings at various plants including Watts Bar Unit 1. The dominant faulted load in the CRDM head adapters is due to seismic excitation. This is due to the CRDM natural frequencies corresponding to the frequencies at which the seismic input is amplified. Seismic spectral comparisons with another plant at the reactor vessel support and tie rod elevations were completed to support the acceptability of the CRDM head adapter and CRDM pressure housing loads. The CRDM head adapter and CRDM pressure housings loads at this plant met the allowable load limits. A seismic spectral comparison between this plant and Watts Bar Unit 1 demonstrates that the CRDM head adapter and CRDM pressure housings loads at Watts Bar Unit 1 will meet allowable load limits.

9 RESULTS AND CONCLUSIONS

The analyses and evaluations documented herein have addressed the reactor vessel components whose structural distortion and integrity in a seismic/LOCA transient must be conservatively limited to insure control rod insertion in the post-LOCA environment. Bounding plant design parameters, seismic response spectra, and LOCA assumptions have been used throughout. The following results and conclusions apply to Watts Bar Unit 1.

1. The RCCA upper internals calculated GT loads are within the allowable limits as established by tests such that control rod insertion will not be precluded for the limiting LBB criteria cold leg break location.

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

3. The upper internals assembly motion, relative to the reactor vessel, associated with a large break LOCA (either cold leg or hot leg) and the associated bounding cooldown displacement provides a time window for control rod insertion in excess of 5 minutes.

4. The combined seismic SSE and LOCA loads for the Watts Bar Unit 1 CRDM Seismic Support Platform Assembly are within the allowable limits and have sufficient margin so that the platform assembly provides the intended lateral restraint to the CRDM rod travel housings.

5. The structural integrity of the Watts Bar CRDM pressure boundary (latch housing and rod travel housing) and the CRDM head adapter is maintained during combined SSE and LOCA loads.

These analyses and evaluations demonstrate that control rods will be inserted following a design basis cold leg Loss-of-Coolant Accident for Watts Bar Unit 1. Consequently, the resulting negative reactivity credit can be applied in evaluating potential recriticality in a post-LOCA scenario.

10 FUTURE FUEL DESIGNS

In order for the conclusions of this report to remain valid for future fuel product lines, it will be necessary to evaluate the effects of the fuel design changes on the results and conclusions presented herein. In order to assist future fuel evaluations, it is helpful to review the five qualification areas with respect to the impact of future fuel design changes on each.

1. RCCA Upper Internals Guide Tube Integrity

   As discussed in Sections 4.4 and 5.4, fuel sensitivity studies showed that the specific vessel/internals/fuel model does not adversely impact the qualification of the RCCA guide tube loads. Furthermore, there are large margins to the acceptable limits. Therefore, it would be expected that future fuel design changes would not challenge the conclusions of this report relative to GT loads for the type of LOCA break considered in this investigation.

2. CRDM Support Platform Assembly Integrity

   As discussed in Sections 4.4 and 8.2, fuel sensitivity studies showed that specific vessel/internals/fuel model does not adversely impact the CRDM platform load qualification. Nevertheless, since the demonstrated CRDM Platform lift leg load margins may be small for some plant or CRDM platform designs, it is recommended that the CRDM support platform lifting leg integrity, relative to control rod insertion for cold leg breaks be evaluated as part of future fuel design. When evaluating future fuel design against the results presented in this report, it can be noted that there is additional margin due to the inherent conservatisms of the bounding assumptions (plant design, break size/location, BOT, RCS initial conditions, enveloped seismic response spectra, etc.).

3. Fuel Integrity

   As discussed in Sections 4.2, calculated fuel grid loads are directly dependent on the Vessel/internals/fuel model and the fuel product that the model represents. Therefore future fuel design must be qualified for control rod insertion, that is, no grid crushing at the control rod locations using the assumptions and acceptance criteria presented in Section 2. This qualification would typically be the responsibility of the fuel supplier.

   It should, however, be noted that in the process of evaluating the fuel grid loads, loads on the guide tubes and the CRDM seismic support platform can be easily obtained from the entire RPV system model (vessel/internals/fuel/CRDM support assembly).

4. Driveline Alignment

   As discussed in Section 7, the evaluation of potential driveline misalignment made no assumptions relative to fuel type. Therefore, the conclusions of the driveline alignment evaluations presented in that section are unaffected by specific fuel design features.

5. CRDM Pressure Housing and CRDM Head Adapter

As discussed in Section 8.3 the structural integrity of the CRDM pressure housing and CRDM head adapter needs to be demonstrated under faulted conditions. Changes in fuel design that impact the seismic or LOCA response of the reactor vessel and its impact on the seismic and LOCA inputs used to confirm the CRDM pressure boundary structural integrity will need to be determined as part of future fuel design evaluations.

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