APPENDIX 3.9E

ASYMMETRIC LOADS

➔(DRN 03-2056, R14)

This appendix was written with respect to main coolant loop breaks (MCLBs), which were subsequently eliminated from consideration of mechanical (dynamic) effects via leak-before-break (LBB) arguments. Elimination of MCLBs by LBB and replacement with branch line pipe breaks (BLPBs) is discussed in Section 3.6.2. LBB methodology is discussed in Section 3.6.3.

➔(EC-8458, R307)

This appendix has been retained for the historical record because most of the Design loads that are retained in the RCS component specifications for extended power uprate to 3716 MWt and the Replacement Steam Generators are due to the postulated MCLBs. For extended power uprate to 3716 MWt and the Replacement Steam Generators, branch line pipe breaks (BLPBs) were postulated, and their effects on the RCS, its supports and interfacing components were compared to those from MCLBs. Most loads due to BLPBs are bounded by loads due to MCLBs. Where new bounding component loads were generated by BLPBs under power uprate conditions, the component was re-evaluated with respect to the criteria of Section 3.9.3 to demonstrate continued acceptability.

🡐(DRN 03-2056, R14; EC-8458, R307)

1.0 INTRODUCTION

In February 1979, the NRC, in Question 110.1, required LP&L to expand FSAR Subsection 3.9.1.5 to more specifically address the consideration of asymmetric loads on reactor coolant system components and supports which could result from postulated reactor coolant pipe breaks within cavities located inside the containment. Enclosure 1, attached to the question, described the information required.

LP&L had tried unsuccessfully, on several past occasions, to obtain clarification of this particular question. The clarification was warranted because of the following:

a) The probability of the type of pipe failures required to be assumed for these analyses is very low;

b) The nature of the analyses required to perform the stated assessment can vary in sophistication to a significant degree so that the time and expenditures required for the performance of said assessment can also vary quite substantially.

c) LP&L has already modified the plant original design to severely limit the area of the breaks that could be postulated to occur in both the reactor and steam generator/pump compartment and has increased the load carrying capacity of the fuel spacer grids and guide tubes; and

d) an analysis of the response of the primary system to break at locations specified in Enclosure 1 was performed and reported in FSAR Appendix 5.4A, and no formal feedback has been obtained to date form the Commission for that submittal.

On August 13, 1980, LP&L personnel and LP&L consultants met with the NRC MEB staff in Bethesda to present LP&L's intended approach in responding to Question 110.1. At this meeting, LP&L indicated that several of the components listed in Enclosure 1 to Question 110.1 had already been assessed by either specific consideration of asymmetric loads or in the design of the component/structure. LP&L further indicated that it intended, to the extend deemed technically feasible, to demonstrate the adequacy of the remaining components/structures by comparison type analyses with other similar plants which had been previously analyzed.

On October 1, 1980, at the draft SER Review Meeting between the NRC MEB Staff, MEB Consultants, LP&L Staff and LP&L's Consultants, LP&L's Consultants presented the basis and the results of an evaluation of the various components/structures listed in Enclosure 1 to Question 110.1. At the conclusion of the meeting and as indicated in the "Action Items" of the meeting minutes (letter from LP&L to R.J. Bosnak, dated October 24, 1980), LP&L was to provide a written report of the evaluation. Accordingly, such a report containing the information presented at the meeting and responses to the MEB Staff/Consultants questions was submitted to the NRC on November 14, 1980.

Initial MEB Staff review of the report generated a set of informal questions and requests for additional information. Responses to the questions and requests were discussed at a meeting with the NRC in Bethesda on January 22, 1981. At this meeting the NRC MEB Staff recommended that LP&L respond to the asymmetric load questions in a manner and format consistent with NUREG-0609, a copy of which was made available to LP&L Consultants (Summary of meeting on LOCA Asymmetric Load Analysis, Docket 50-382, dated February 14, 1981).

Accordingly, the subsequent sections which present the basis, analytical methods and results of the evaluation of the capability of the various components/structures to withstand asymmetric loads are organized in the same way as NUREG-0609.

Replies to questions raised by the NRC Staff and their Consultants, and additional information they requested to enable them to conduct expeditious review, are incorporated either in the subsequent sections or in attachments at the end of the report.

Table 1.1 summarizes the status of the assessment of each component/structure requested in Question 110.1. Details of the assessment are presented in the following sections.

2.0 BACKGROUND

During a postulated loss-of-coolant accident, thermodynamic and hydrodynamic induced loads occur through the primary system. When the loss-of-coolant accident is in the form of a circumferential pipe rupture at the inlet nozzle of the reactor pressure vessel, a decompression of the reactor pressure vessel occurs within a short period of time. Decompression waves which originate at the postulated break travel around the inlet plenum and propagate downward along the downcomer annulus. The finite time required by the decompression disturbances to travel about the vessel causes a transient pressure differential field t be created across the core support barrel (CSB) and the vessel inner surface. This field imposes a transient asymmetric loading on the core support barrel as well as the vessel itself. Since the postulated pipe break is located within the biological shield wall, the blow down fluid flashing into the reactor cavity also causes a transient pressurization acting externally on the vessel. This external pressurization is also asymmetric. The internal asymmetric loading (IAL) and external asymmetric loading (EAL) act in the same direction for breaks occurring in the cold leg piping. For breaks in the hot legs, the internal asymmetric load is virtually absent in the horizontal direction, hence the two loads are additive in the vertical direction only. These loadings are transmitted to the reactor vessel support system. The resultant reaction forces at the support interfaces must be considered in the evaluation of the adequacy of the support system together with the thrust load resulting from the break and other operating loads. Normal operating loads, together with postulated seismic loads, as well as the EAL and IAL for both cold leg and hot leg breaks at the vessel nozzle had been previously analyzed in FSAR Appendix 5.4A with regard to vessel support.

Breaks outside the reactor cavity can result in IAL imposed on the reactor pressure vessel and internals, and in EAL on the reactor coolant pump and steam generators. For the breaks outside the cavity, the adequacy of the primary system supports has been assessed for those breaks at appropriate primary system locations listed in Appendix 3.6A, as part of the design requirement.

3.0 DEVELOPMENT OF LOADING FUNCTIONS

3.1 DETERMINATION OF POSTULATED BREAK SIZES

The evaluation of the primary system for LOCA loads requires definition and development of the loading transients associated with the postulated pipe ruptures. In turn these transients depend on the postulated break locations, and the break conditions (i.e., break area and opening time), which depend on the primary system configuration and the presence of restraints to limit the size of the break. The Waterford 3 primary system has been provided with system restraints specifically designed to limit the size of the breaks in the main piping and such that there will be no unacceptable deformation in the RCS piping.

The Waterford 3 primary system piping was subjected to the a detailed stress survey. Implementation of the stress survey criteria indicated in FSAR Subsection 3.6.2.1.1.1, which criteria are consistent with the criteria delineated in Reference 3, results in the postulation of a pipe break location at the terminal ends of each leg of the RCS main loop piping and at two of the three intermediate elbows in each of the pump suction legs. Implementation of the detailed finite element stress analysis criteria of Subsection 3.6.2.1.1.1 results in the postulation of a longitudinal type pipe break at each elbow break location, oriented within 90 degrees circumferentially from the crotch of the elbow.

The design basis pipe break sizes are as presented in Reference 3, except as discussed in Subsection 3.6.2.1.1. The set of pipe stops and system restraints used in Waterford 3 are within a range of parameters for which these break sizes were confirmed using Reference 3 methodology.

The break opening times and the break sizes are also the same as given in CENPD 168 since they result from the detailed analyses presented in that document for a system which is identical to Waterford 3.

The circumferential break opening times and break sizes for Waterford summarized in Table 3.1. Longitudinal breaks are reported in FSAR Table 5.4A-3.

Figure 17 and 22 and Table 4-1 of CENPD 168 (Reference 3) describe the Waterford 3 RCS restraints and pipe stops.

The time varying forcing functions used and the time varying responses of the RCS main loop piping for Waterford are presented on Figures 15, 16 and 27 through 30 of CENPD 168 (Reference 3).

Each of the postulated breaks produces transient loading conditions on the primary system. These loads can be characterized into four components:

a) Subcooled blowdown loads - dynamic hydraulic forces within the primary system due to rapid subcooled depressurization of the system.

b) Cavity pressure loads - resulting from pressurization of the space in the reactor cavity or in the steam generator compartment around the components.

c) Strain Energy Loads - due to the release of strain energy where the rupture occurs.

d) Jet Impingement and Jet Thrust Loads - which act at the break location.

Each of these load components have been addressed in the Waterford 3 asymmetric load assessment.

3.2 INTERNAL SUBCOOLED BLOWDOWN LOADS

A thermohydraulic analysis was performed in accordance with the procedures and models developed in Reference 4 for the controlling breaks of those described in Subsection 3.1 of this report. The thermohydraulic analyses for this evaluation used an approach which anteceded the presently approved CE approach embodied in their Topical Report CENPD-252-P as referenced in NUREG-0609. The approach used in the Waterford 3 analyses, which are explained in Appendix 5.4A, produces subcooled loads which are larger than those which would be produced by using the computer program CEFLASH-4B described in the topical above. The methodology of Reference 4 utilized the computer program WATERHAMMER and CEFLASH-4 for the analysis of cavity breaks only.

In the methodology of Reference 4, both programs were used since WATERHAMMER predicted asymmetric pressure differentials during the initial subcooled portion of the blowdown that were larger than the corresponding differentials predicted by CEFLASH-4.

WATERHAMMER results were thus employed in the subcooled regime and joined to CEFLASH-4 results which were then used for subsequent pressure difference distribution calculations by CE.

Reference 4 provides comparisons of this analytical approach with experiments and also the models used for WATERHAMMER and CEFLASH-4 to represent both hot leg and cold leg breaks.

The pressure and flow parameters resulting from the application of the above thermohydraulic models were used to compute three-dimensional time history forcing functions acting on the reactor vessel and the internals, using the break sizes and opening times described in Subsection 3.1

The loading histories thus derived were combined with the cavity pressure loads strain-release loads and thrust loads, and applied to the structural model of the primary system.

3.3 CAVITY PRESSURE LOADS

Analyses to calculate the asymmetric subcompartment pressures were performed by Ebasco using mass and energy release data provided by Combustion Engineering.

The CEFLASH-4A computer program was used to calculate the mass and energy release rates from the postulated reactor coolant system pipe ruptures. This program is described in Subsection 6.2.1.1.4 of CESSAR (Reference 5). The reactor coolant system nodalization scheme utilized to compute the mass and energy releases is given in FSAR Figure 6.2-20.

FSAR Subsection 6.2.1.2.3 describes the assumptions made in order to maximize the short-term blowdown rates to the the subcompartments.

The same section also describes the assumptions made in analyzing the pressure response of the reactor cavity and steam generator subcompartments.

The analyses of these subcompartments utilized the RELAP-3 Mod 68 computer code (Reference 6), modified to allow the use of a multiplier in the Moody choked flow correlation of 0.6. The RELAP-3 Mod 68 code was the best code available at the time of this analysis, and predated the RELAP-4 Mod 5 currently utilized at Ebasco for such analyses. The coefficient of 0.6 was used for physical junctions.

Further description of the modeling assumptions used in the analyses are provided in FSAR Subsection 6.2.1.2.3.

The asymmetric pressure loads calculated by these analyses were converted to forces time histories across the reactor vessel and steam generators, which combined with the asymmetric internal subcooled decompression loads, strain energy release loads and thrust loads, were applied to the structural model of the Reactor Coolant System. The asymmetric pressure time histories were also applied to the biological shield wall and a the steam generator subcompartment walls as equivalent static loads using peak differential pressures calculated in the different regions of the pertinent subcompartment models.

4.0 STRUCTURAL ANALYSIS

4.1 PRIMARY SYSTEM ANALYTICAL MODEL

The major components of the Reactor Coolant System are designed to withstand the forces associated with the design basis pipe breaks discussed in Section 3.1 of this report in combination with the safe shutdown earthquake and normal operating conditions. The forces associated with the postulated pipe breaks include pipe thrust forces and tension release forces at the break location as developed in CENPD 168 (Reference 3), resultant subcompartment differential pressurization forces and hydraulic forces acting on the reactor internals and the reactor pressure vessel.

The model utilized for the dynamic analysis is a lumped parameter model discussed in CENPD 168 (Reference 3) which includes details of the reactor vessel and supports, major connected piping and components, and the reactor internals. This model is also described in FSAR Appendix 5.4A.

Appendix 5.4A also describes the approach taken to reduce the detailed model of the RCS and reactor internals to the condensed model used to evaluate the overall system response to the LOCA loads.

4.1.1 REACTOR AND STEAM GENERATOR

The reactor vessel representation is shown on Figure 5.4A-9 of FSAR Appendix 5.4A. The internal and external asymmetric forces are applied at the indicated nodes. The steam generator vessel representation is shown in the flexibility analysis model of Figure 5.4A-7 of FSAR Appendix 5.4A.

4.1.2 PRIMARY COOLANT PIPING

The model of the primary piping is also shown on Figure 5.4A-7 of FSAR Appendix 5.4A.

4.1.3 COMPONENT SUPPORTS

The Waterford 3 component supports are modeled as springs with gaps. The stiffnesses of the springs for the pump and steam generator supports are defined in Table 4-1 of Reference 3 along with the hot gap allowed. The vessel support stiffnesses and gaps are as follows.

In the vertical direction there is no gap between the vessel and supporting ring girder except a maximum hot gap of 0.015 inches at the hold-down studs is provided. The support stiffness is 2.255×10^7 lb/in upward and 1.087×10^8 lb/in in the downward vertical direction.

Horizontally, a minimum stiffness of 2.864×10^7 lb/in exists in the tangential direction. The nominal horizontal gap between the reactor vessel and the cavity wall is less than 0.035 inches in the tangential direction.

A detailed description of the reactor vessel steam generator and pumps support and restraint system is given in FSAR Subsection 3.8.3.1.5. FSAR Figure 3.8-34 illustrates the reactor vessel support system. Figures 3.8-35 and 3.8-36 illustrate the steam generator supports and Figure 3.8-37 shows the reactor coolant pumps supports. Figures 3.8-38 and 3.8-39 show details of the pipe restraints designed to limit the break sizes.

4.1.4 INTERNALS

The model of the internals is shown on Figure 5.4A-8, of FSAR Appendix 5.4A.

4.1.5 FUEL

The fuel model is shown on Figure 5.4A-8 of FSAR Appendix 5.4A. Additional modeling of the fuel for the purposes of its analysis is addressed by the Response to Question 231.2.

4.1.6 CONTROL ELEMENT DRIVE MECHANISMS (CEDMs)

CEDMs were not specifically modeled for Waterford. For a discussion of the CEDM analysis please refer to Subsection 5.1.8 of this report.

4.1.7 PIPING ATTACHED TO THE RCS

Two different models have been used for the analysis of RCS attached piping in Waterford 3. One model is used for a dynamic non-linear analysis of one of the lines. One line only was analyzed in this fashion since the computer execution cost is very high. The model that was utilized is shown on Figure 4.1. The computer program utilized in the analysis is PLAST which is described in detail in FSAR Appendix 3.6B. The program allows consideration of geometric (i.e., gaps) as well as material (i.e., plasticity) non-linearities in both the piping and the supports/restraints placed on the pipe.

The analysis performed with PLAST as evidenced in the results discussed in Subsection 5.1.7 of this report did not reveal any material non-linearities and hence it was in effect an elastic, non-linear analysis.

The remaining lines were analyzed using the computer program PIPESTRESS2010 described in FSAR Subsection 3.9.1.2.1, expanded to permit computation of the maximum linear elastic response of the line, support/restraint system by the generalized response analysis method. This method with consideration of the higher modes, is chosen instead of true time history for cost saving purposes even though it is known to produce conservative bounding answers.

The line analyzed by non-linear analysis in true time history has also been analyzed by the generalized response method so that direct comparison is possible between the two analyses.

4.2 BIOLOGICAL SHIELD AND SECONDARY SHIELD WALLS

The description of the biological or primary shield wall is provided in FSAR Subsection 3.8.3.1.1. The description of the secondary shield wall is provided in FSAR Subsection 3.8.3.1.2. FSAR Subsection 3.8.3.4.1.2 describes the models and computer programs used to design both of these walls.

5.0 RESULTS OF ASSESSMENT

Results of the assessment of the Waterford 3 RCS System and its support to withstand the loads resulting from the postulated design basis loads of Section 3.1 of this report are given in the following subsections.

5.1 REACTOR VESSEL AND STEAM GENERATOR

5.1.1 REACTOR VESSEL

The acceptance criteria for the reactor vessel are given in FSAR Subsection 3.9.1.4.1.1. Analyses reported in FSAR Appendix 5.4A indicated that when the LOCA resulting stresses are combined with normal operating and seismic stresses, the vessel meets the acceptance criteria.

5.1.2 STEAM GENERATOR AND REACTOR COOLANT PUMPS

The steam generator has also been demonstrated to meet the acceptance criteria under the design basis loads resulting from the breaks postulated in Section 3.1 of this report. The pumps have not been specifically analyzed except as part of the reactor coolant piping analysis. Refer to Subsection 5.1.7 of this report.

Maximum reactor vessel displacements have been imposed upon a flexibility analyses model such as described in Figure 5.4A-7 of FSAR Appendix 5.4A in order to account for the load on the steam generator supports as a result of vessel motion.

5.1.3 REACTOR VESSEL SUPPORTS

The analysis reported in FSAR Appendix 5.4A has demonstrated the adequacy of the reactor vessel supports to withstand the asymmetric loads resulting from the 350 in.^{2} cold leg vessel inlet, 100 in.^{2} hot leg vessel outlet, and 600 in.^{2} hot leg steam generator inlet guillotine breaks. The response of the vessel supports for other breaks needed no evaluation since it is clear that those breaks would result in lower loads. The absence of external asymmetric loads for those breaks, postulated to occur outside the reactor vessel cavity, is what produces the lower overall loads since (a) thrust loads remain the same or are less for cold leg and hot leg breaks respectively outside the reactor cavity as they are for the cold leg and hot leg breaks inside the cavity, and (b) internal asymmetric loads are not significantly affected by the size of the break area alone but rather depend on the combination of the break area and the break opening time. Sensitivity studies performed for instance, for cold leg breaks, have shown that a full area guillotine break (1414 in.^{2}) requiring approximately 30 msec to fully open, a 576 in.^{2} guillotine break requiring 18 msec to open, and a 188 in.^{2} guillotine break requiring 8 msec to open, result in internal asymmetric loads differing only by a few percent. Refer to Table 5.8 for the results of such sensitivity studies. Figure 5.1 shows the behavior of the break area developed as a function of time following an assumed instantaneously occurring severance of the cold leg pipe at the vessel inlet nozzle. This figure shows that to each break area there corresponds an opening time.

Table 3.1 reports the opening time required to achieve the circumferential breaks postulated in the RCS of Waterford 3 which are also tabulated in FSAR Table 6.2-1.

FSAR Table 3.8-36 indicates the margins between the allowable stress or load on the reactor vessel support and the actual stress or load.

FSAR Table 3.8-22 specifies the load combinations which were employed in the assessment.

5.1.4 BIOLOGICAL (PRIMARY) SHIELD WALL

The adequacy of the biological shield is insured by design since the wall has been designed to withstand loads in excess of those transmitted by the reactor vessel supports as computed in FSAR Appendix 5.4A in combination with the asymmetric pressure loads across the wall as described in FSAR Subsection 6.2.1.2, and other normal operating and seismic loads.

A comparison between the calculated design loads for this wall and the ultimate capacity of the wall is shown in FSAR Table 3.8-32/ FSAR Table 3.8-21 lists the loading combinations employed in this assessment of the secondary shield wall.

5.1.5 ASSESSMENT OF STEAM GENERATOR AND PUMP COMPARTMENT WALL (SECONDARY)

The adequacy of the compartment wall surrounding the reactor coolant pumps and steam generator is also assured by design. Refer to FSAR Subsections 3.8.3.3.1 and 3.8.3.3.2 for the structural design and Subsection 6.2.1.2 for the subcompartment analyses providing the asymmetric loads across the wall for the postulated breaks (see FSAR Table 6.2-1 for the breaks postulated in the compartments). FSAR Table 3.8-32 indicates the margins of safety for this wall.

5.1.6 ASSESSMENT OF STEAM GENERATOR, REACTOR COOLANT PUMP SUPPORTS

The asymmetric pressure histories reported for the breaks postulated in FSAR Subsection 6.2.1.2 have been used to determine the asymmetric loads across the steam generators and reactor coolant pumps. These components have been designed to accept these loads as stated in Subsection 5.1.2 of this report. The adequacy of the steam generator and reactor coolant pump supports is also assured by design. Refer to FSAR Subsections 3.6.2.3 and FSAR Appendix 3.6A for the design of those supports and restraints. FSAR Subsection 3.8.3.4.1.2, Items b3), b5), and b6) describe the basis and models used for the design. FSAR Tables 3.8-33, 3.8-34, 3.8-35, and 3.8-31, illustrate the margins between allowable and maximum calculated stresses for the pump supports, steam generator upper and lower supports and reactor coolant pipe stops respectively.

5.1.7 ASSESSMENT OF PRIMARY COOLANT PIPING

The adequacy of the reactor coolant piping under LOCA conditions had been verified when piping rupture restraints limiting the break areas were backfitted into the original design (refer to FSAR Subsection 3.6.2.3 and FSAR Appendix 3.6A). The Waterford 3 design falls within the bounds of the primary system design of CENPD 168 (Reference 3), hence the RCS piping is adequate to withstand the LOCA loads by reference to CENPD 168. Moreover, the Waterford 3 design being very similar to that of St. Lucie 1 (Reference 7), and both having seismic loads which are an order of magnitude less than the allowable loads, and the St. Lucie 1 RCS piping having been demonstrated (Reference 7) to be adequate by plant specific analysis for combined LOCA and seismic loads, then it is judged that the Waterford 3 RCS piping should also be adequate since its expected stresses are lower than those of St. Lucie 1 for the following reasons: (a) design basis break areas in Waterford 3 are smaller than break areas of St. Lucie 1 at corresponding locations, hence asymmetric effects are smaller, and (b) conservatively estimated vessel motions resulting from asymmetric loads due to pipe breaks within the cavity in Waterford 3 are significantly smaller than the corresponding vessel motions in St. Lucie 1 (refer to Figures 5.2, 5.3 and 5.4). These two results stem from the significant difference in the forces across the vessel generated by the pressures in the reactor cavities. Figure 5.5 shows the horizontal and vertical forces acting across the vessel for various plants with reactor cavities which are not necessarily very similar. The force computed for the Waterford 3 break area of 350 in^2 in the cold leg from pressures reported in FSAR Subsection 6.2.1.2 is seen to be considerably less than that for the generic plant full break, and to essentially fall in line with the trend of lateral force vs. break area shown in Figure 5.5.

Table 5.1 compares the seismic moments at selected points in the RCS main piping for St. Lucie 1 and Waterford 3.

5.1.8 ASSESSMENT OF CEDMS

Control rod insertion for break areas exceeding 0.5 square feet is not required due to NSSS design. Hence, it is only necessary to demonstrate that the CEDMs retain their integrity.

The CEDMs in Waterford 3 are identical to those of the CE generic plant (Reference 1) and St. Lucie 1 (Reference 7), with the exception that in the latter two plants, all nozzles are of equal length; whereas Waterford 3 CEDM nozzles have unequal length resulting in all CEDMs tops being at the same elevation. This difference causes the CEDMs differing in length from those analyzed for the CE generic plant and St. Lucie 1 (referred to hereinafter as the short nozzle CEDMs) to respond to lower frequencies on a mode by mode comparison than those previously analyzed. Table 5.2 lists the frequencies of the first seven modes for both the shortest and longest Waterford 3 CEDMs.

In Waterford 3 the conservatively estimated amplitudes of the Reactor Vessel head motions (which provide the excitation to the CEDMs) are at most one-half of those of the generic CE plant for the cold leg break and the amplitudes of the motions for the hot leg break are another 30 percent lower. This is evident from Table 5.4A-1 of FSAR Appendix 5.4A wherein peak forces on the supports from the 100 in^2 outlet nozzle guillotine break are 72 percent of those resulting from the 350 in^2 guillotine break at the inlet nozzle. Since, as stated in FSAR Subsection 5.4A.6 of that Appendix, the horizontal effect of all loads on the system causes the vessel to traverse the horizontal support gaps, impact the supports and remain in contact with the support, oscillating about a deflected, close gap position, it follows that the horizontal displacements from cold and hot leg breaks are simply proportional to the loads after the initial deflection. Because of the orientation of the supports with respect to the break, the initial deflection to close the gaps for a hot leg break is the same as for a cold leg break. Subsequent displacements for the hot leg break, however will be about 30 percent smaller since the computed loads from FSAR Appendix 5.4A are about 30 percent smaller.

For the cold leg break, the dominant frequency of the driving force is between 9-12 Hz for Waterford 3. This is sufficiently close to possible modal resonances for any length CEDMs to be concerned with amplification of motion. For the generic plant and for St. Lucie 1, the dominant frequency of the driving motion is about 17 Hz. On a purely elastic basis, long-term excitation of their CEDMs (all short) could be expected to result in two-fold amplification of motion as measured at the center of the mass. Results of an elasto-plastic analysis for the St. Lucie 1 CEDM nozzle (Reference 7) revealed instead an amplification of approximately 25 percent. This is attributed to plasticity in the nozzle as well as short time application of the input motions. A similar behavior can be expected for Waterford 3, except that a higher amplification would be predicted on an elastic basis (three and 1/2 times that of St. Lucie 1 for any length nozzle). The expected amplification from an elasto-plastic analysis would thus be of the order of 85 percent. Since the amplitudes of the driving displacements in Waterford 3 are half of those of St. Lucie 1, the maximum moment that can be expected for Waterford 3 is less than that computed for St. Lucie 1 (which was 173,000 in-lbs) and is expected to be approximately 130,000 in-lbs.*

The ultimate moment that the Waterford 3 nozzle can carry is approximately 356,m000 in-lbs (same as St. Lucie 1) as shown in Figure 5.6 and has been computed using the approach of Gerber (Reference 2) with minimum ASME property values for the nozzle material. To assure that the pressure boundary is not violated, the criterion used is from the ASME Code, Appendix F, that the integrity of the pressure boundary is assured if the applied loads do not exceed 70 percent of the plastic instability load. Since the maximum LOCA moment for cold leg breaks in Waterford 3 is well below this 70 percent criterion (249,000 in-lbs), it can be stated that the Waterford 3 CEDMs are adequate for cold leg breaks. The maximum seismic moment that the nozzle sees is 104,000 in-lbs by previous actual plant seismic analysis. The absolute addition of the LOCA and seismic moments would thus be less than 70 percent criterion.

For hot leg breaks, the dominant frequency of the driving force is about 16-17 Hz, hence it is not near any resonance with Waterford 3 CEDMs and further it would excite higher modes than the cold leg break. Since the amplitude of the driving motions for the hot leg break are even less than those for the cold leg break, the adequacy of the CEDMs for this break is assured.

5.1.9 RCS ATTACHED PIPING ASSESSMENT

For the ECCS lines, an assessment of the adequacy of the lines has been initially made by comparing the routing of the Waterford 3 lines with that of the ECCS line which had been analyzed in St. Lucie 1. That analysis had shown that some plasticity might occur in the vicinity of the first elbow after the nozzle. For the RCS attached line which is shown in FSAR Figures 3.6A-25 and 3.6A-26 for instance (one of the six lines attached to the RCS for Waterford), the first elbow is that one directly below the intersection of line 1RC42-2RL2 and line 1RC14-45RL2. That initial analysis had been predicated on the assumptions that one can conservatively estimate the bending moment at that location by use of the simple equation where a is the displacement at the nozzle, M is the maximum bending moment, L is the length (the length of the pipe section between the nozzle and the elbow), E is Young's Modulus, and I is the moment of inertia. Referring to the same example of FSAR Figures 3.6A-25 and 3.6A-26, the nozzle displacements are applied at the intersection of 1RC42-2RL2 and the centerline of 1RC14-45RL2, the length L is measured from that point downward to the midpoint of the first elbow, which is then assumed to be fixed and restrained against rotations by the rest of the piping system via line 1RC14-45RL2. Inherent in this simplistic approach was the modeling of a dynamic event (i.e., time dependent displacement or accelerations applied at the nozzle resulting in inertia forces which induce primary stresses) by a static cantilever subject to a displacement δ.

$$M = 3\,\sigma\,EI / L^2$$

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*based on Dynamic Amplification Factor = $\dfrac{1}{1-\left(\dfrac{f}{f_n}\right)^2}$ where f is the driving frequency and f_n the modal frequency.

The loads resulting from the simplistic model application therefore were taken as primary loads even though as static anchor displacement they could be construed to be secondary loads. The reason for originally employing the simplistic model were two-fold: (a) the exorbitant cost of performing potentially elasto-plastic nonlinear analyses for each of the six lines attached to the RCS piping and (b) the reasonably good agreement between the moments computed in the line by this method vs. the sophisticated analytical method employed in St. Lucie 1.

For instance, the maximum bending stress computed by the above formula for St. Lucie 1 is 22,880 psi which compares very well with the 19, 280 psi computed by time history nonlinear elasto-plastic analysis of that line. That analysis had also shown that everywhere else in the line, stresses in combination with seismic stresses were within allowable limits for St. Lucie 1.

In Waterford 3, the maximum bending moments had originally been computed on the basis that nozzle displacements would be half or less than the St. Lucie 1 nozzle displacements (since vessel displacements are one-half or less). The maximum bending moments were computed for two RCS attached lines; one a 12 inch and one a 14 inch nominal line, are 3.2×10^6 in-lbs and 4.1×10^6 in-lbs, respectively and they occurred a the first elbow near the nozzle at the RCS pipe.

Because of the first elbow plasticity, as computed in this manner, consideration of simultaneous seismic loads had been given as follows: the maximum bending moment due to LOCA and that due to seismic had been added absolutely and their sum has been compared to 70 percent of the ultimate capacity of the elbow to carry moment without collapse. The calculated moments are compared with 70 percent of the maximum bending moment carrying capacity of those elbows computed by the methodology of Gerber (Reference 2), which are
5.34×10^6 in-lbs and 5.81×10^6 in-lbs, respectively for the 12 inch and the 14 inch pipes scaled by the B_2 Index in the ASME Code, where $B_2 = 1.9026$ for the 12 inch line and
$B_2 = 2.04609$ for the 14 inch line.

The simpler method of computing the capability of the elbow to sustain an applied moment had been compared to a finite element analysis method employed in Reference 1 and was found to given excellent agreement. For instance, the simple method predicted maximum collapse moment for the elbow of an ECCS line in St. Lucie 1 to be 5.94×10^6 in-lbs whereas the finite element method of Reference 1 for a virtually identical elbow computed it to be
5.5×10^6 in-lbs.

For the two lines examined, the total bending moment (LOCA and seismic) was less in one instance and slightly higher in the other instance than 70 percent of the collapse moment. Hence, it could be concluded that the ECCS lines would retain their integrity and function during a simultaneous LOCA and seismic event. This conclusion is predicated on the similar conclusion derived in the CE analyses of the lines attached to the RCS for the generic plant and for specific plants as well (Reference 1). In this analysis CE has computed that even at 70 percent of the ultimate moments, plastic strains are of the order of 2 percent. At that strain level, CE concludes that functionability is not impaired. Refer to Subsections 4.5.3 and 4.9.3 of Reference 1 for a description of the methods employed in CE's analyses, and for the results. Attachment 1 provides additional details on the comparative analyses used in this evaluation.

Subsequent questions raised during the draft SER review meeting (between MEB Staff, MEB Consultants, LP&L's Staff and LP&L's Consultants) in New York on October 1, 1980, as to whether the seismic supports of the RCS attached lines would be able to withstand the reaction forces resulting from LOCA-induced motions, caused departure from the simple analysis which had been employed to determine the bending moment in the pipe. An elasto-plastic analysis has been performed on the basis of bounding Waterford 3 specific vessel displacements derived from FSAR Appendix 5.4A, which are assumed to adequately represent the ECCS nozzle displacements. The analysis concludes that the analyzed line and its supports/restraints are capable of withstanding loads resulting from the combined postulated LOCA and seismic event. This analysis has been carried out for one of the RCS attached lines only, because of the large cost involved.

The analysis utilizes the PLAST Code which is a direct integration piping dynamic analysis program which allows the consideration of geometric (i.e., gaps) as well as material (i.e., plasticity) nonlinearities in the supports/restraints and pipe itself. A description of this code is provided in FSAR Appendix 3.6B. The input motions at the nozzle utilized in the full dynamic analysis of the Waterford line are shown in Figure 5.7 where X and Z represent the time histories of the nozzle deflections in the two orthogonal horizontal directions and Y is the time history of the vertical nozzle deflection. Displacement time histories have been utilized instead of acceleration time histories since the former were available and the PLAST code can accept either as input.

The output of the program provides information regarding the maximum reaction forces due to the inertial LOCA motion alone at the various restraints, and a also the maximum moments in the different points of the piping system. The maxima are obtained by searching through the entire time histories of the forces and moments calculated by the program.

The piping model utilized for the dynamic analysis is shown on Figure 4.1. the maximum moment (resultant) in the Waterford 3 piping system is 7.8×10^5 in-lbs occurs at the first elbow (closest to the nozzle to the primary piping).

The maximum reaction forces at various restraints is reported in Table 5.3.

The maximum bending moment in the piping calculated by the actual dynamic analysis are considerably below 70 percent of the ultimate moment carrying capability. Strains are elastic, and so that the piping experiences extremely small deformations and retains its cross-sectional area, thus ensuring functionality of the pipe.

Table 5.3 lists the reaction loads computed for Waterford 3 for LOCA motions, together with a description of the type of support/restraints employed for the restraints. Seismic loads and thermal loads are also listed. The overall loads computed by summing thermal and the LOCA and seismic loads combined in SRSS fashion is compared to the capacity of the support/restraint to failure.

From Table 5.3 it can be concluded that the supports/restraints are capable of withstanding loads resulting from a combination of postulated LOCA and seismic events.

This analysis conducted for the 1RC14-45RL2 line also provides confirmation of the conservatism of the assessment of the first elbow bending moments by use of the simple formula previously used. For this line the bending moment computed by elasto-plastic detailed analysis is 7.8×10^5 in-lbs which can be compared with a calculated 4.1×10^6 in-lbs determined by the simple formula.

The remainder of the lines attached to the RCS have been analyzed using a linear elastic analysis described in Subsection 4.1.7 of this report.

The method employed is not a true time history method. Rather it is a generalized response method utilizing modal analysis. In this method the maxima of the modal amplitudes are summed at each point in the line, regardless of the time at which they occur and of their sign by the SRSS method or absolute method depending on the spacing of the modes. The stresses in the line and reactions at the support restraints are then computed for the three orthogonal excitations by the SRSS method. This method is guaranteed to provide a conservative bound on the value of the stresses in the lines and the loads in the supports/restraints, but is used since it can be run reasonably economically in the computer, and hence permits ready conservative assessment of all of the lines attached to the RCS.

As a measure of the conservatism of the method, line 1RC14-45RL2, which has been analyzed by the detailed non-linear analysis utilizing PLAST, has also been analyzed by the generalized response method. The results of the two analyses are shown in Table 5.4 for the restraint support reactions and Table 5.5 for the moments at the first elbow (region of highest stress).

Table 5.4 lists two outputs for the generalized response analysis. The first column assumes that the snubbers in the line are not present to simulate the behavior predicted by the non-linear analysis wherein the gaps in the snubbers were shown not to close. The last column includes the effect of the snubbers, which can be modeled only as linear springs.

Except in the immediate vicinity of the first restraint/snubber the two results are not too significantly different. As expected, the linear generalized response method severely overpredicts both the stresses in the line and the reactions at the supports. Table 5.6 shows the results obtained for the restraints/supports of the remaining lines by the linear generalized response and also repeats those results for the 1RC14-45RL2 line. Only three more lines are shown since the other two are virtually identical to two of those shown. Table 5.7 compares the stresses in the lines at the high stress points. Results for the remaining lines are very comparable to those obtained for the 1RC14-45RL2 line. FSAR Figures 3.6A-8 through 3.6A-10 and 3.6A-14 through 3.6A-18 show portions of the isometrics for the lines which have been analyzed and indicate the position of the supports/restraints. This means that detailed non-linear time history analysis of the other lines can be expected to provide results similar to those obtained for the 1RC14-45RL2 line shown in Table 5.3.

The above, coupled with the fact that the vessel motions used are a conservative bound of those expected, confirms that the lines attached to the RCS and their supports/restraints can adequately withstand the postulated LOCA.>

5.1.10 ASSESSMENT OF REACTOR INTERNALS

Motions of the core barrel, the reactor vessel, as well as the relative motion between core barrel and vessel and hydraulic loads, are the important parameters in assessing the adequacy of the reactor internals. The relative motion between core barrel and vessel represents the new element in the assessment since it was not not considered in the analyses reported in FSAR Subsection 3.9.2 which had utilized very conservative hydrodynamic asymmetric loads, but had held the reactor vessel fixed. Previously the asymmetric loads were not part of the design loads.

A comparison of the vessel motions (Figures 5.2 and 5.3) resulting from the most significant of the postulated breaks: namely, the break at the vessel inlet nozzle, obtained for Waterford 3 to those computed for similar plants (i.e., the generic CE plant (Calvert Cliffs, see Reference 1) and the St. Lucie 1 plant (Reference 7) shows that the Waterford 3 vessel deflections are significantly smaller. This is due in large part to the fact that the break area in Waterford 3 is significantly smaller (i.e., limited by design) than the corresponding areas in the other two plants. Another significant difference besides the smaller magnitude of the vessel deflections in Waterford 3 is that the period of oscillation is longer. Corresponding accelerations are thus much lower in Waterford 3 than the generic plant or St. Lucie 1. The vessel deflections of Waterford 3 are further compared with those calculated for a plant subjected to a cold leg nozzle inlet break area of 188 in^2. The comparison shows similar displacement time histories in magnitude and frequency for the Waterford 3 and the plant with 188 in^2 break area. The larger displacement in the plant with the smaller break area resulted from the larger gaps between the reactor vessel nozzle pads and the supports. The break area at the reactor vessel outlet nozzle for Waterford 3 (100 in^2) is slightly smaller than the corresponding break for the generic CE plant (135 in^2). It can be expected therefore that at worst the motions of the reactor vessel resulting from that hot leg break will be similar in both amplitude and periodicity for both plants but that the Waterford 3 amplitudes will be somewhat lower. For the generic CE plant (Reference 1) vessel motions resulting from the hot leg break at the reactor vessel have been computed to be less than 1/3 of those resulting from the cold leg break at the vessel. Since corresponding cold leg break vessel motions for Waterford 3 are essentially 1/2 of the generic plant motions, the cold leg break inside the cavity in Waterford 3 will produce vessel motions which are approximately 30 percent larger than the motions produced by the hot leg break. This is confirmed by the arguments presented in Subsection 5.1.8 wherein a 30 percent reduction of the hot leg induced motions vs. the cold leg break induced motion is justified by comparing the computed peak loads on the supports.

As indicated above, the vessel motions for Waterford 3 are at most one half of the corresponding motions of the generic plant (see Figures 5.2 and 5.3) and have about the same frequency content initially but subsequently they exhibit a longer period. The internal asymmetric hydraulic forces across the barrel, are essentially unaffected by the break size and can thus be considered nearly the same in both plants. Sensitivity studies have shown that internal asymmetric hydraulic loads only decrease about one percent when the break is reduced from full area opening in 30 msec. to an area of 188 in^2 opening in 8 msec. Table 5.8 summarizes these results. The relative motion between the vessel and the core barrel for Waterford 3 will be slightly smaller than that for the generic plant on the basis of slightly smaller internal hydraulic loads and the absolute values of core barrel and in-phase vessel motions for Waterford 3 being about half that for the generic plant. The in-phase nature of the barrel and vessel motions is confirmed by Figure 5.8 applicable to the plant with a break area of 188 in^2.

It can therefore be expected that reaction forces on the Waterford 3 internals will be less than the corresponding forces for a cold leg break in the CE generic plant; whereas, they will be nearly the same for hot leg breaks since the break areas in both plant are approximately identical (100 square inches inside the cavity for Waterford 3 and 135 square inches for the generic plant). The hot leg break outside the cavity is larger (600 in^2), however, it gives rise to no cavity pressure. Hence, a significant load, which would contribute to the motion of the vessel if the break had been inside the cavity, is absent.

Since the internals had been analyzed for a full hot leg break with a fixed vessel, and since the vessel motions in the absence of cavity pressure are expected to be 10 percent less than corresponding motions when cavity pressures are present, the determining hot leg break would be the hot leg break within the cavity. In the generic plant analysis it has been found that the cold leg break in the cavity results in more severe loading on the internals.

Table 5.9 compares the internals for Waterford 3 with those of the generic plant. Table 5.10 provides additional comparisons. As can be seen, the Waterford 3 internals, with one exception unimportant for this evaluation, are capable of withstanding higher loads than the generic plant design. The additional margin has been computed by considering the relative capacity of each element to accommodate bending (both meridional and circumferential) hoop stresses, local and global shear, and buckling; thus assuming the minimum capacity irrespective of the loading condition.

A comparison of the Waterford 3 internals with the generic plant internals indicates that generally where relatively low margins to design values had been computed for the generic plant, the Waterford 3 internals have at least 11 percent capacity, with much larger margins in most of the internals components. These added margins, coupled with the fact that loads are expected to be lower as a result of the limited break areas, assure the acceptability of Waterford 3 internals under the combined LOCA and seismic loads.

The fuel alignment plate is the only component whose capacity in Waterford 3 is less than that for the generic plant. Stresses in that plate are not critical for the generic plant, it is expected that similar conclusions can also be drawn for Waterford 3.

5.1.11 ASSESSMENT OF FUEL ASSEMBLIES

As indicated in Table 1.1, at the draft SER review meeting with the MEB Staff/Consultants on October 1, 1980, and at the meetings on January 22, 1981 in Bethesda with the MEB and CSB Staff, the adequacy of Waterford 3 fuel assemblies will be provided in LP&L's response to FSAR Question 231.2, as Question 231.2 specifically addresses the ability of fuel assemblies to withstand combined seismic and LOCA mechanical loads. However, as indicated at the October 1, 1980 meeting, and subsequently in the NRC minutes of the January 22, 1981 meetings, the Waterford 3 fuel assembly evaluation for LOCA will be limited to the cold leg vessel inlet break, since all prior analyses of fuel assemblies (generic plant, St. Lucie 1) have indicated that the cold leg break at the vessel inlet nozzle produces maximum LOCA loads on the assemblies. Attachment 2 provides the basis for arriving at the above conclusion.

6.0 CONCLUSIONS

Based on the results of the evaluation contained in this report, it is concluded that all the reactor system components/structures cited in Question 110.1 (except fuel assemblies which will be addressed under FSAR Question 231.2), can withstand the combination of loads due to a postulated, but highly unlikely, design basis LOCA plus seismic event and that Waterford 3 can operate without undue risk to the health and safety of the public. An evaluation using more sophisticated techniques than contained in this report would only quantify the safety margins that already exist based on the evaluation contained herein. Hence, from a value-impact standpoint it is believed that further analysis using more sophisticated techniques is not warranted.

At the conclusion of the draft SER Review Meeting and a subsequent meeting in Bethesda, the MEB Staff provided oral acceptance of our proposal to assess the adequacy of fuel assemblies subjected to LOCA loads resulting from the controlling break; i.e., a postulated break at the cold leg vessel inlet nozzle. The basis for selecting this break location is contained in the report. The Waterford 3 fuel assembly assessment will also be made on this basis.

At the same time the analyses of the internal hydraulics, the vessel motions, etc. required to provide the basic information needed to establish the fuel alignment plate and core support plant motions for this "worst" break, will provide information regarding the asymmetric loads across the internals, the actual vessel motion, etc. Thus the fuel analysis will verify the validity of the technical bases used in this report.

7.0 REFERENCES

1) Calvert Cliffs Nuclear Power Plant Units No. 1 and 2, Docket Nos. 50-317 and 50-318, Reactor Coolant System Asymmetric Loads Evaluation Program - Final Report.

2) Gerber, T.L., "Plastic Deformation of Piping due to Pipe Whip Loading," ASME Paper 74-NE-1, 1974.

3) Combustion Engineering Topical Report, CENPD-168, "Design Basis Pipe Breaks," July 1975.

4) CENPD-42, "Topical Report on "Dynamic Analysis of Reactor Vessel Internals", December 1973.

5) CESSAR, Combustion Engineering Standard Safety Analysis Report, Combustion Engineering, Incorporated, NRC Docket No. STN-50-470.

6) Rettig, W.H. et al, "RELAP-3 - A Computer Program for Reactor Blowdown Analysis," Idaho Nuclear Corporation, IN-1321, June 1970.

7) Florida Power & Light, Docket No. 50-335, "Reactor Coolant System Asymmetric LOCA Load Evaluation," July 1980.

TABLE 1.1 (1 OF 2)

ASSESSMENT OF STRUCTURES/COMPONENTS OF QUESTION 110.1

Component/Structure	Assessment Status	Evaluation Basis	References	Comments
Reactor Pressure Vessel	Complete	Plant Specific Analyses	FSAR App 5.4A	
Steam Generators	Complete	Design	CE Design Reports	
Reactor Coolant Pumps	Complete	Design	CE Design Reports	
Reactor Vessel Supports	Complete	Plant Specific Analyses	FSAR App 5.4A	
Steam Generator Supports	Complete	Design	FSAR Sect 3.6.2.3 and App. 3.6A	
Reactor Coolant Pump Supports	Complete	Design	FSAR Sect 3.6.2.3 and App. 3.6A	
Biological Shield Wall	Complete	Design	FSAR Sect 3.8.3.3.1, 3.8.3.3.2 and 6.2.1.2	
Steam Generator, R C Pump Compartment Wall	Complete	Design	FSAR Sect 3.8.3.3.1, 3.8.3.3.2 and 6.2.1.2	
RCS Main Piping	Complete	Plant Specific Analyses and and Reference to other Plant Analyses	FSAR Sect 3.6.2.3 App. 3.6A (Reference 1)	Waterford had been analyzed for all postulated breaks w/o consideration of asymmetric loads and for a 350 in^2 cold leg and 100 in^2 hot leg guillotine break at the reactor vessel nozzle for asymmetric loads also (see text).
RCS Attached Piping (ECCS, etc.)	Complete	Detailed non-linear analysis of one line and detailed linear analysis of remaining lines	Reference 1 and Reference 7 (See Text)	Generic Plant Analyses were for both cold and hot leg guillotine breaks

TABLE 1.1 (2 OF 2)

ASSESSMENT OF STRUCTURES/COMPONENTS OF QUESTION 110.1

Component/Structure	Assessment Status	Evaluation Basis	References	Comments
RCS Attached Piping Supports and Restraints	Complete	Detailed non-linear analysis of one line and detailed linear analysis of remaining lines	(See Text)	
CEDMs	Complete	Comparison to previously analyzed CEDM's	Reference 1 and Reference 7 (See Text)	Generic Plant Analyses were for both cold and hot leg guillotine breaks
Reactor Internals	Complete	Comparison to analyses of similar plants	Reference 1 (See Text)	Generic Plant Analyses were for both cold and hot leg guillotine breaks
Fuel	Will be addressed in response to Q 231.2		(See Text)	Prior analyses done for similar CE plants (with 14 x 14 fuel however) have indicated that the cold leg vessel inlet break is the determining break for fuel assessment. Comparative analyses indicate that there should be no problem with coolability of the core.

TABLE 3.1

DESIGN BASIS BREAK OPENING TIMES, SIZES AND LOCATIONS
FOR RCS POSTULATED BREAKS

Reactor Cavity	Minimum Opening Time (from CENPD-168, Reference 3)
100 in^2 hot leg guillotine break	20 msec.
350 in^2 discharge leg guillotine	6 msec.
Steam Generator Compartment	
600 in^2 hot leg guillotine	14 msec.
430 in^2 suction leg guillotine	11 msec.
592 in^2 suction leg guillotine	17 msec.
480 in^2 discharge leg guillotine	28 msec.

TABLE 5.1

COMPARISON OF SEISMIC MOMENTS (IN-KIPS)
SSE FOR ST. LUCIE 1 AND WATERFORD 3 AT
POINTS IN THE RCS MAIN INTACT LOOP

	ST. LUCIE 1	WATERFORD 3
RCP Discharge Nozzle	5910	6899
RCP Suction Nozzle	7256	6932
RV Inlet Nozzle	5272	7038
RV Outlet Nozzle	2535	8400

TABLE 5.2

MODAL FREQUENCIES (HZ) OF WATERFORD 3 CEDMs

Mode No.	Shortest	Longest
1	3.4	2,7
2	5.1	4.9
3	11.5	10.9
4	13.5	11.8
5	13.8	13.6
6	23.8	15.0
7	43.3	37.1

TABLE 5.3

WATERFORD 3 ECCS LINE REACTION FORCES ON SUPPORTS/RESTRAINTS (KIPS)

Support/Restraint	LOCA	SSE	Thermal (Normal)	Total*	Capacity
(Refer to Figure 6 for location of Supports/Restraints)					
RCSR-92 (Snubber)	0	19.2	0	19.2	91.0
RCRR-94 (Strut)	11.0/-4.7	17.8	.929	21.8	26.0
RCRR-292 (Strut)	2.1/-1.5	16.4	9.382	25.9	29.0
RCRR-100 (Strut)	1.3/-1.1	23.1	6.008	29.1	35.3
RCRR-183 (Strut)	2.0/-1.9	5.0	.212	5.6	27.0
RCRR-184 (Snubber)	0	31.5	0	31.5	91.0
RCSR-186 (Snubber)	0	27.3	0	27.3	92.0
RCRR-293	1.1/-0.9	12.2	3.292	15.5	27.0

*Total = Thermal + $(LOCA^2 + SSE^2)^{1/2}$

TABLE 5.4

COMPARISON BETWEEN RESULTS
OR ANALYSIS FOR 1RC14-45RL2 OBTAINED BY NONLINEAR TIME HISTORY
AND LINEAR ELASTIC GENERALIZED RESPONSE ANALYSIS

PROGRAM

PLAST 2267 (MAX.) — PIPESTRESS 2010 50 MODES 7 LEFTOUT FORCE* (GENERALIZED RESPONSE METHOD)

NODE POINT — NONLINEAR ELEASTIC SNUBBERS ENGAGED — SNUBBERS DELETED

NODE POINT	Fx (lb)	Fx (lb) / Fy(lb)	Fy (lb)	Fz (lb)	Fz(lb)	fX (lb)	fY (lb)	Fz (lb)	
7					11013			28134	
				35015					
800		2110.3				11191			
	10528								
10			4293.6			7846	16464		
	36080								
100		1252.4							
	10135								
13			1958.4				17933		
	39118								
1605			1005.8				6025		
	6197								
20			226				2367		
	2641								
235			476.8				2028		
	5504								
2905			415.4				1884		
	2638								
321			279				2248		
	2807								
2909			272.3				3195		
	4667								
2812					264			1052	
				1845					
2817					264.5			429	
				522					
2917		315.6				1518			
2794									
2818			170.9				1081		
	317								
2820		35.2	59.8		109.1	54	328	150	
67	70			112					
34/354		1292.3	204.9		1150.5	7660	2374	6411	
3831	3067			4393					
8 (v.s.)			122.3				481		
458									
6 (snub)									39072
130 (snub)									
16 (snub)									2217
			4759						
2350 (snub)									
2421				1398					
2805 (snub)									
1031				1786					
2819 (snub)									
1211									
2009 (snub)									
				3901					

NOTE: *Modes up to 96 Hz and Leftout Force to account for other modes.

TABLE 5.5

MAXIMUM BENDING MOMENTS IN 1RC14-45RL2 COMPUTER BY NONLINEAR ELASTIC ANALYSIS AND GENERALIZED RESPONSE ELASTIC ANALYSIS

Generalized Response

Line (Calc #)	Node Point	Name	Fx	Force (lb) Fy	Fz	Moment (Ft-lb) Mx	My	Mz	Moment	Stress (psi) Pressure	Total
1RC14-45RL2	3		7711	56450	76979	167915	24098	10294	22759	11326	34085
	1	Elbow									
	4		7711	56450	76979	189515	18965	15425	25589	11326	36916
K = 1 x 108 lb/in At Point #1	5		5990	48721	31999	69643	6647	15425	9594	11326	20920
(Snub. Deleted)	1	Elbow									
	5000		5990	48721	31999	142803	12367	23752	19457	11326	30783

Non-Linear Analysis

Line (Calc #)	Node Point	Name	Moment (Ft-lb) Mx	My	Mz
1RC14-45RL2	3		65067	703	-1686
	1	Elbow			
	4		-64383	-189	1301
	5		-47917	-34334	-3594
	1	Elbow			
	5000		-50125	3858	3347

TABLE 5.6 (1 of 4)

MAXIMUM REACTION FORCES AT SUPPORTS/RESTRAINTS
OF RCS ATTACHED PIPING

1RC14-45RL2

	POINT	------------FORCES IN POUNDS------------		
	NO.	Fx	Fy	Fz
RESTRAINT	7	0.	0.	28931.
VAR SUPPORT	8	0.	486.	0.
RESTRAINT	800	10170.	0.	0.
RESTRAINT	10	0.	17596.	0.
RESTRAINT	100	14911.	0.	0.
RESTRAINT	13	0.	19062.	0.
RESTRAINT	1605	0.	4981	0.
RESTRAINT	20	0.	2046	0.
RESTRAINT	235	0.	5554	0.
RESTRAINT	2905	0.	4537.	0.
RESTRAINT	321	0.	2566	0.
RESTRAINT	2909	0.	4537	0.
RESTRAINT	2812	0.	0.	954.
RESTRAINT	2817	0.	0.	461.
RESTRAINT	2917	1531.	0.	0.
RESTRAINT	2818	0.	167.	0.
RESTRAINT	2820	34.	0.	0.
RESTRAINT	2820	0.	44.	0.
RESTRAINT	2820	0.	0.	93.
ROT.RESTRAINT	2820			
ROT.RESTRAINT	2820			
ROT.RESTRAINT	2820			
SNUBBER	6	10695.	0.	0.
SNUBBER	130	0.	0.	22192.
SNUBBER	16	5175.	0.	11107.
SNUBBER	2350	4944.	0.	2856.
SNUBBER	2805	2143.	0.	3710.
SNUBBER	2809	1204.	0.	0.
SNUBBER	2009	0.	0.	1941.

TABLE 5.6 (2 of 4)

MAXIMUM REACTION FORCES AT SUPPORTS/RESTRAINTS
OF RCS ATTACHED PIPING

1RC12-40RL2B

	POINT NO.	FORCES IN POUNDS Fx	Fy	Fz
ANCHOR	4095	990.	2516.	880.
ANCHOR	430	442.	605.	330.
RESTRAINT	2601	1604.	0.	2378.
RESTRAINT	2600	0.	9599.	0.
RESTRAINT	320	0.	7978.	0.
RESTRAINT	3800	0.	8727.	234.
RESTRAINT	3801	8526.	0.	0.
SNUBBER	4000	0.	17464.	0.
RESTRAINT	4001	4482.	0.	0.
RESTRAINT	4101	0.	0.	37917.
VAR SUPPORT	4300	0.	179.	0.
RESTRAINT	610	0.	8454.	0.
RESTRAINT	6301	6157.	0.	4153.
RESTRAINT	6700	9571.	0.	2875.
RESTRAINT	7000	786.	0.	4169.
RESTRAINT	5900	1988.	0.	0.
RESTRAINT	770	0.	12825.	0.
RESTRAINT	3841	0.	0.	2973.
RESTRAINT	3945	1094.	0.	3180.
RESTRAINT	3961	0.	2074.	0.
RESTRAINT	397	858.	0.	2141.
RESTRAINT	4010	628.	0.	293.
RESTRAINT	4020	0.	2200.	0.
RESTRAINT	4045	827.	0.	331.
RESTRAINT	4046	0.	2026.	0.
VAR SUPPORT	414	0.	6.	0.
RESTRAINT	415	120.	0.	300.
SNUBBER	4150	0.	629.	0.
RESTRAINT	4165	0.	677.	0.
RESTRAINT	4180	840.	0.	336.
RESTRAINT	4221	0.	620.	0.
RESTRAINT	4220	392.	0.	309.
RESTRAINT	4240	0.	319.	0.

TABLE 5.6 (3 of 4)

MAXIMUM REACTION FORCES AT SUPPORTS/RESTRAINTS
OF RCS ATTACHED PIPING

1RC12-38RL1B

	POINT NO.	------------FORCES IN POUNDS------------ Fx	Fy	Fz
ANCHOR	5143	1383.	2230.	1213.
ANCHOR	5431	6551.	3765.	6466.
VAR SUPPORT	192	0.	22.	0.
RESTRAINT	194	15241.	0.	0.
SNUBBER	195	0.	8843.	0.
RESTRAINT	197	0.	0.	14542.
RESTRAINT	1995	0.	9114.	0.
RESTRAINT	200	8894.	0.	0.
RESTRAINT	2015	0.	9757.	0.
RESTRAINT	206	0.	16443.	0.
RESTRAINT	5025	2424.	0.	0.
RESTRAINT	5075.	0.	2619.	0.
RESTRAINT	5075	1069.	0.	2486.
RESTRAINT	5114	0.	2407.	0.
RESTRAINT	511	886.	0.	2067.
SNUBBER	516	22572.	0.	0.
RESTRAINT	532	0.	0.	34011.
SNUBBER	5171	12549.	0.	0.
RESTRAINT	517	0.	10675.	0.
RESTRAINT	5188	0.	21046.	0.
SNUBBER	519	0.	0.	43548.
RESTRAINT	522	0.	21847.	0.
RESTRAINT	5361	0.	27771.	0.

TABLE 5.6 (4 of 4)

MAXIMUM REACTION FORCES AT SUPPORTS/RESTRAINTS
OF RCS ATTACHED PIPING

1RC12-39RL2A

	POINT NO.	------------FORCES IN POUNDS------------ Fx	Fy	Fz
ANCHOR	7405	2609.	112.	2946.
ANCHOR	905.	334.	341.	137.
RESTRAINT	903	4095.	0.	1740.
SNUBBER	1306	0.	0.	6084.
SNUBBER	1501	0.	10073.	0.
RESTRAINT	1903	9466.	0.	0.
VAR SUPPORT	1901	0.	109.	0.
RESTRAINT	2103	0.	13407.	0.
SNUBBER	2101	0.	0.	18462.
RESTRAINT	2406	0.	19308.	0.
RESTRAINT	2601	23145.	0.	0.
RESTRAINT	5101.	0.	6846.	0.
RESTRAINT	5401	3215.	0.	402.
VAR SUPPORT	5903	0.	10.	0.
RESTRAINT	5901	0.	0.	5826.
SNUBBER	63	0.	2498.	0.
SNUBBER	6403	4091.	0.	0.
VAR SUPPORT	6401	0.	20.	0.
RESTRAINT	6705	0.	1386.	0.
RESTRAINT	7201	0.	1072.	0.
SNUBBER	7496	0.	0.	1381.
RESTRAINT	78	0.	1131.	0.
RESTRAINT	8005	810.	0.	0.
RESTRAINT	84	175.	0.	371.
RESTRAINT	8701	0.	284.	0.

NOTE: 1RC14-44RL1 is similar to 1RC14-45RL2
1RC12-40RC1A is similar to 1RC12-40RL2B

TABLE 5.7

MAXIMUM STRESSES IN RCS ATTACHED LINES RESULTING FROM LOCA MOTION AND INTERNAL PRESSURE

Line (Calc #)	Node Point	Name	Pressure Stress (psi)	Total Stress (psi)	Line (Calc #)	Node Point	Name	Pressure Stress (psi)	Total Stress (psi)
1RC12-40RL2B	46		11419	36024	1RC12-39RL2A	29		11419	34878
	1	Elbow				1	Elbow		
	47		11419	41784		30		11419	26756
	44		11419	31459					
	1	Elbow				3		22759	34085
	45		11419	26676	1RC14-45RL2	1	Elbow		
	48	Tangent	6885	30713					
						4		25589	36916
1RC12-38RL1B	5301	Tangent	6885	46418		5		9594	20920
					(Snub.Deleted)	1	Elbow		
	5301		11419	42718		5000		19457	30783
	1	Elbow							
	5302		11419	53527					
						3		19925	31251
	529		11419	26684	1RL14-45RL2	1	Elbow		
	1	Elbow				4		33448	44775
	530		11419	27249					
						5		13236	24563
	527		11419	26797	(Snub.Engaged)	1	Elbow		
	1	Elbow				1000		21921	33247
	528		11419	26249					

TABLE 5.8

A COMPARISON ON THE RESULTS OF THE COLD LEG BUILLOTINE
WITH DIFFERENT BREAK OPENING TIMES AND ASSUMPTIONS

Break Opening Time (milliseconds)	Break Area (in^2)	Maximum Pressure Differences Across the CSB (psi)
1	576	1120
18	576	372
8	188	374
30	1414	378

TABLE 5.9

MECHANICAL SYSTEMS AND COMPONENTS - COMPARISON OF STRUCTURAL DESIGN PARAMETERS

Parameters		Generic Plant	Waterford 3	Added Capacity of LP&L Internals (Percent)	
Structural					
	R_{mean}, in.	75 1/4	75 1/4	Hoop	20
				Bending+	40
Upper CSB	t, in.	2 1/2	3	Shear*	17
	L, in.	135 5/8	138 7/16	Buckling	72
	R_{mean}, in.	74 7/8	75 1/4	Hoop	42
				Bending+	66
Middle CSB	t, in.	1 3/4	2 1/2	Shear*	21
	L, in.	144 3/4	160 1/8	Buckling	133
	R_{mean}, in.	74 5/8	75 1/4	Hoop	32
				Bending+	75
Lower CSB	t, in.	2 1/4	3	Shear *	51
	L, in.	38	33 5/8	Buckling	133
Lower cylinder ID, in.		141	141	Hoop	No increase
Core cylinder OD, in.		145	145	Binding+	(No increase meridion) 29 Axially
				Shear*	11
Support cylinder L, in.		42	37 5/8	Buckling	No increase
Structure supported		CSB Lower Flange	CSB Lower Flange	Circumferential Bending	30
Fuel Alignment Plate					No change (not important)(
CSB OAL, in.		328 1/2	347		
Core shroud support		Core Support Plate			No increase
UGS				Hoop	27
	R_{mean}, in.	72 5/8	72 1/2	Bending+	56
Cylinder	t, in.	2	2 1/2	Shear*	24
	L, in.	24	24	Buckling	98
Beams	in.	24 x 1 1/2	24 x 1 1/2		No Change
Plate on Top of Beams	t, in.	4	3 1/2	Shear	-12.5 (This plate is not important
Shrouds		The two are the same but the support distance is shorter for Waterford 3		Bending	-3.0 since beams are weaker)
					+30

Legend: CSB - Core Support Barrell
UGS - Upper Guide Structure
\+ - Minimum of axial or meridional bending
* - Minimum of overall or local shear

TABLE 5.10

COMPARISON OF OPERATING CONDITIONS AND DIMENSIONS FOR THE WATERFORD 3* AND GENERIC PLANT AFFECTING INTERNAL ASYMMETRIC LOAD ANALYSIS

Operating Condition	Waterford 3	Generic Plant	St. Lucie 1
Normal Operating Presssure (psi)	2250	2250	2250
Normal Operating Temperature H.L. (F)	611	598	592
Normal Operating Temperature C.L. (F)	553	548	543
Total Flowrate (lb/hr)	148×10^6	130×10^6	130×10^6
Dimensions			
Hot Leg I.D. (in.)	42	42	42
Cold Leg I.D. (in.)	30	30	30
Reactor Vessel I.D. (in.)	172	172	172
RV Nozzle Support Interface Gap (in.)	.035	.125	.215

* Other dimensions as given in Table 5.9

ATTACHMENT 1 (Appendix 3.9E)

SAFETY INJECTION LINE STRESSES DUE TO LOCA + SEISMIC LOADING

WATERFORD 3

SAFETY INJECTION LINE STRESSES DUE TO MOTION OF REACTOR

The stresses developed in the Waterford SES 3 Safety Injection Lines are estimated by comparison with a previous analysis. Table 1-1 provides such a comparison.

TABLE 1-1

STRESSES IN ECCS LINES

Plant	St. Lucie 1	Waterford 3	
Line	1-B-1	1RC12-40RL2B	1RC14-45RL2
O.D.	12.750 inches		14.000 inches
I.D.	10.126 inches	10.270 inches	11.636 inches
Pressure	2235 psi	2235 psi	2235 psi
Pressure	10,860 psi	11,490 psi	13,240 psi\
Maximum Dynamic Displacement	2/3 inch	1/3 inch	1/3 inch
Length of First Appropriate Section	10.73 ft.	7.00 ft.	7.07 ft.
Bending (LOCA only)	19,280 psi(1) 22,800 psi(2)	26,830(2)	28,890 psi(1)
Equation 9(3)	≈45,000 psi	≈52,530 psi	≈72,350 psi

Since the maximum stresses calculated are at or above the 3 Sm limit of 48,000 psi, it is necessary to demonstrate the integrity/functionality of these lines. To do this, the maximum benidng moment which can be sustained by the pipe is computed and compared to bending moment actually developed in the lines. (The maximum sustainable bending moment is taken to be 70 percent of Gerber's (Reference 2) result; the actual bending moment is computed by the simplified mehtod as explained in Subsection 5.1.9 of Appendix 3.9E). Table 1-2 summarizes such a comparison.

(1) from PLAST detailed analysis

(2) Calculated from Simplified Method for Computation of Pipe Stress due to Specified Displacements $M = 3\sigma EI/L^2$

(3) $\sigma = B_1 \times \sigma_{Press} + B_2 \times \sigma_{Bend}$; B_1 = 1.000, B_2 = 1.817 (St. Lucie 1),
B_2 = 1.9026 (Waterford 3 - 12 inch pipe)
B_2 = 2.04609 (Waterford 3 - 14 inch pipe)

TABLE 1-2

MAXIMUM LOAD CARRYING CAPABILITY OF FIRST ELBOW IN ECCS

PIPE VS COMPUTED MOMENTS

Plant	St. Lucie 1	Waterford 3	
Line	1-B-1	1RC12-40RL2B	1RC14-45RL2
t/R_o	0.206	0.194	0.169
ϕ^*	0.309	0.292	0.253
ϕ^*/R_o	0.0484	0.0458	0.0362
$\left(\frac{{}^{40}o}{3+n}\ \frac{2}{3}\right)^n I_N$		93.3 x 10^3 $(psi)^4$	
$(R_o^{3+n} - R_i)^{3+n}$	165.9	158.8	189.9
$(\phi^*/R_o)^n$	0.695	0.691	0.671
M_{max}^+	5.94 x 10^6 in-lb	5.34 x 10^6 in-lb	5.81 x 10^6 in-lb
70% M_{max}	4.15 x 10^6 in-lb	3.74 x 10^6 in-lb	4.07 x 10^6 in-lb
$M_{Bending}$	2.8 x 10^6 in-lb	3.2 x 10^6 in-lb	4.1 x 10^6 in-lb
$M_{Seismic}$	6.,5 x 10^5 in-lb	3.98 x 10^5 in-lb	2.76 x 10^5 in-lb

+Computed by the method of Gerber (Reference 2) for straight pipes, and scaled down by the B_2 factor in equation 3.

ATTACHMENT 2 (APPENDIX 3.9E)

JUSTIFICATION OF COLD LEG INLET BREAK AS THE DETERMINING BREAK FOR FUEL

JUSTIFICATION OF COLD LEG INLET BREAK AS THE DETERMINING BREAK FOR FUEL

Analyses performed for the generic CE plant (Reference 1), as well as specific plants like Fort Calhoun, have indicated that the determining break insofar as fuel analysis is concerned is a cold leg guillotine break at the reactor vessel inlet nozzle. This conclusion is based on the fact that the response of the fuel for these plants has been analyzed for both a full area guillotine break at the reactor vessel inlet nozzle and a limited area (135 in^2) guillotine break at the reactor vessel outlet nozzle, and that the former break has been found to be limiting break.

Another result of these analysis has been that the beam-column effect, due to concurrent lateral and axial loading is also more pronounced for inlet break, but that this effect does not significantly increase the maximum fuel bundle stresses. (A dynamic beam-column analysis was performed for the plants of Reference 1 to determine any aditional bending stresses and stability of the fuel assembly due to concurrent lateral and axial loading, and that analysis had also shown that the beam-column effects are more sensitive to lateral bending moments than to axial forces.)

The lateral bending moments on the fuel assembly due to the input excitation of the core support plate, fuel alignment plate, and core shroud displacement time histories have been found to be significantly larger for the reactor vessel inlet breaks than for the reactor vessel outlet break.

These analyses have been performed for a 14 x 14 fuel and therefore detailed stress results have limited applicability to Waterford 3. However, some of the conclusions reached for these plants have a direct bearng on Waterford 3 fuel assemblies also:

a) The determining load condition on fuel is the lateral bending moments due to the excitation of the core support plate, fuel alignment plate and core shroud lateral displacement time histories. These were largest in the generic plant (Reference 1) for the inlet break; however, the inlet break in the plant is a full area break whereas in Waterford it is a limited area break (350 in^2). In both the generic plant and in Waterford, the outlet break area is limited. For the generic plant this break is 135 in^2 while in Waterford 3 that break area is 100 in^2.

   Figures 2-1, 2-2 and 2-3 compare the lateral motions of the generic plant vessel for cold and hot leg breaks. Significant displacements only occur in the x-direction (parallel to the hot leg direction) for hot leg break, so only that direction is shown.

   The motions of the vessel, together with the internal asymmetric hydraulic load determine the lateral motions of the core support plate, fuel alignment plate, andcore shroud. The vessel motions of the inlet break are clearly much larger than those for a hot leg break for the generic CE plant.

For Waterford 3, the vessel motions (lateral) are essentially half of those of the generic plant (see Figures 2-4 and 2-5 for an inlet break 350 in^2). The motions of the Waterford vessel resulting from a hot leg outlet break of 100 in^2 would be very similar to those of the generic plant (135 in^2). However, the amplitudes of these motions would still be approximately 30 to 40 percent less than the corresponding motions for the 350 in^2 cold leg inlet guillotine break.

In addition, the hot leg break would produce essentially no asymmetric internal hydraulic loads across the core barrel. Hence, it can be concluded that the inlet break is the determining break for establishing the largest lateral bending moments on the fuel bundle for Waterford also.

b) Axial loads can be higher for the hot leg breaks, however, the generic plant analysis has indicated that they are not as significant as lateral bending loads in determining fuel stability and bending stresses. This, coupled with the fact that the beam-column analyses performed for the generic plant indicated negligible additional effects on the fuel bundle stresses, over those predicted from the lateral loading analysis, is indicative of the cold leg break being the determining break for the fuel assessment.

c) For axial loads, the hot leg breaks would provide the dominant loads. The Waterford 3 plant has two postulated hot leg breaks, a 100 in^2 break at the vessel outlet and a 600 in^2 break at the S.G. inlet. (Lateral Vessel motions, resulting from either breaks, would be considerably smaller than those resulting from the 350 in^2 cold leg vessel inlet break. The 600 in^2 break would produce about 90 percent of the vessel motions produced by the 100 in^2 break.) These two breaks require different times to open. In any case, Waterford 3 has been analyzed for the pure axial loads resulting from a full area hot leg break (the so-called core bounce analysis), and the fuel was shown to be adequate for tha tbreak which produced axial loads that are larger than either the 100 in^2 or 600 in^2 hot leg breaks.

It is therefore concluded that the cold leg vessel inlet break is the determining break for demonstrating the adequacy of Waterford 3 fuel assemblies.