APPENDIX 5.4A

DYNAMIC ANALYSIS FOR THE WATERFORD 3 REACTOR VESSEL SUPPORT LOADS UNDER LOCA CONDITIONS

→(DRN 03-2059, 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.

This appendix has been retained for the historical record because the Reactor Vessel (RV) support loads, as well as RV shell response motions used in RV internals evaluations, that were generated from this analysis, bound the respective RV responses generated by the branch line pipe break (BLPB) analysis for power uprate to 3716 MWt.

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5.4A.1 PURPOSE

This appendix presents the results of dynamic structural analyses of the WATERFORD 3 reactor vessel and supports subjected to loads induced by pipe breaks at the reactor vessel inlet and outlet nozzles and at the steam generator inlet nozzle. Two generations of dynamic analysis are presented; the original analysis using the original model and results of Reference 1 thermohydraulic methodology, the second analysis using an expanded model and results of Reference 2 methodology. Simultaneous effects of pipe break thrust and external and internal horizontal and vertical asymmetric pressure loads were applied to the reactor vessel and internals as a consequence of each postulated pipe rupture. A non-linear time history dynamic analysis of a three dimensional mathematical model of the Reactor Coolant System including details of the reactor internals, pressure vessel, supports and piping was performed for each postulated pipe break to demonstrate the adequacy of the reactor vessel supports.

5.4A.2 CONCLUSION

The calculated loads on reactor vessel supports do not exceed specified loads except for one load in the original analysis. For all supports, all calculated loads have been evaluated and found to be acceptable.

5.4A.3 BACKGROUND

The reactor internals, including the fuel and supporting structures, are suspended from the closure flange region of the reactor vessel and are surrounded by the cylindrical "core support barrel" (CSB) as shown on Figure 5.4A-1. The CSB and reactor vessel are essentially concentric cylinders throughout the length of the CSB.

Upon postulation of a break in a primary coolant pipe, as on Figure 5.4A-2, several rapidly occurring events cause internal and external transient loads to act upon the reactor vessel. For a reactor vessel inlet break, asymmetric pressure changes take place in the annulus between the CSB and the vessel. Decompression occurs on the side of the vessel nearest the pipe break before pressure on the opposite side changes. The momentary difference in pressure across the CSB induces lateral loads in opposite directions on the CSB and the reactor vessel as shown on Figure 5.4A-3. Vertical loads are also applied to the internals and to the vessel due to the vertical flow resistance through the core and asymmetric axial decompression of the vessel. Simultaneously, as fluid escapes through the break, the annulus between the reactor vessel and biological shield wall becomes asymmetrically pressurized. The difference in pressure across the reactor vessel causes additional horizontal and vertical external loads on the vessel as shown on Figure 5.4A-4. In addition, the vessel is loaded by the effects of initial tension release and blowdown thrust on the broken pipe (Figure 5.4A-5).

The loads occur simultaneously as shown on Figure 5.4A-6. For a reactor vessel outlet break, the same type of loadings as shown on Figure 5.4A-6 occur, but the internal loads are more predominately vertical due to smaller break size and more rapid decompression of the upper plenum. For each postulated break the time history maximum reactor vessel support reactions due to the complete set of horizontal and vertical loads are calculated.

For a steam generator (SG) inlet break, the same type of loadings on the reactor vessel (RV) as shown on Figure 5.4A-6 occur, but because of the position of the break, the initial tension release and blowdown thrust contain a vertical component of load that enhances rocking effects on the vessel. Because this break occurs in the SG subcompartment and not in the RV cavity, there are no differential pressure loads on the reactor vessel.

5.4A.4 MODELS

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Different models were developed for the two generations of dynamic analysis. In the original analysis, condensed structural models of the RCS and reactor internals were created from highly detailed representations of each component by maintaining response characteristics and interface response compatibility. A detailed model of the reactor coolant system is shown on Figure 5.4A-7 and a detailed model of the reactor internals is shown on Figure 5.4A-8. The internals model and the RCS model were condensed and coupled to form the final model used for the analysis of the vessel supports. (See Figures 5.4A-9 and 5.4A-10.) The condensed model of the internals is intended to represent the effects of the reactor internals on the reactor vessel support reactions and is not intended to be used to analyze the internals themselves. The adequacy of the reduced model of the internals for this purpose was verified by analysis of planar models of a coupled system having a detailed internals model and a simplified representation of the vessel and supports. Figures 5.4A-11 and 5.4A-12 show excellent agreement between the reactor vessel support reactions using the detailed and the reduced internals models. This model was used for the analysis of the 350 in.^{2} R.V. inlet nozzle and 100 in.^{2} R.V. outlet nozzle circumferential guillotines. The criteria used to define break and crack locations and configuration for high energy piping systems is discussed in FSAR Section 3.6.2.1.

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The details of the coupled model are summarized in Table 5.4A-2. The model is three dimensional and has 680 total static degrees of freedom and 61 mass degrees of freedom. The reactor vessel and all internal components are modeled as colinear elements and gaps are present at internal and support interfaces. Details of the reactor vessel support arrangements are shown on Figure 5.4A-13.

The models for the second set of dynamic asymmetric loads analyses contain a similarly detailed reactor vessel and condensed reactor internals model. However, they also contain detailed steam generator (SG), pump (RCP), RCS piping and RCS support modelling and are a combination of models shown by Figures 5.4A-7 and 5.4A-8. These expanded models were used for a comparative analysis of the 350 in.^{2} break (see Figure 5.4A-5) and for the analysis of the 600 in.^{2} S.G. inlet nozzle circumferential guillotine. For the 600 in.^{2} SG inlet nozzle break, an elasto-plastic multi-mass model of the severed hot leg with a gapped pipe restraint was added to the model. The model used for the 350 in.^{2} break has 1912 total static degrees of freedom and 108 mass degrees of freedom. The model used for the 600 in.^{2} break has 1961 total static degrees of freedom and 131 mass degrees of freedom.

The physical definition of the structure was given as input to the STRUDL[(3)] Computer Code which generated the condensed stiffness matrix used in the dynamic analysis. Hydrodynamic effects, including both virtual mass and annular effects, were accounted for in the coupling between the RV and CSB and between the CSB and the core shroud. The resulting mass matrix is non-diagonal because of the inclusion of the annular hydrodynamic effects (Reference 4).

All RCS support stiffnesses and pipe restraint stiffnesses used were plant specific.

5.4A.5 FORCING FUNCTIONS

The original model shown on Figures 5.4A-9 and 5.4-10 and described in Table 5.4A-2 was subjected to the loads that resulted from a postulated rupture at the inlet and outlet RV nozzles (350 in.^{2} and 100 in.^{2} breaks respectively). The location and size of breaks have been determined using the methods of Reference 5 and are summarized in Table 5.4A-3. For each postulated break a thermohydraulic analysis was performed according to the procedures and models developed in Reference 1. The resulting pressure and flow parameters were used to calculate three dimensional time history forcing functions acting on the reactor vessel and internals at the locations shown on Figures 5.4A-9 and 5.4A-10. Analyses to calculate the asymmetric subcompartment pressures in the reactor vessel cavity were performed by Ebasco using mass energy release data provided by CE. The resulting pressures were used to calculate the three dimensional time history forces applied to the exterior of the vessel. The forces were applied to the model at the locations shown on Figure 5.4A-10. Internal and external forces were applied simultaneously in three directions to the Vessel-Internals System. The dynamic analysis to determine the response to the system is performed using the DAGS and FORCE (6,7) codes to determine the maximum reactions at the vessel support to locations shown on Figure 5.4A-14.

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The expanded models, shown on Figures 5.4A-7 and 5.4A-10 and described in Tables 5.4A-4 and 5.4A-5, were subjected to the loads that resulted from a postulated rupture at each of the RV inlet and SG inlet nozzles (350 in.^{2} and 600 in.^{2} breaks, respectively). For each of these postulated breaks, a thermohydraulic analysis was performed according to the procedures and models developed in Reference 2. All dynamic analysis methodology used was the same as in the original analysis.

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5.4A.6 RESULTS

Results of maximum reactor vessel supports loads are given in Table 5.4A-1 for all postulated breaks.

It can be seen from Table 5.4A-1 that the 350 in.^{2} inlet break produces the controlling values for all support force components. The table also shows the comparison results between the original and revised analyses for the controlling 350 in.^{2} break. This comparison demonstrates that the original analysis provided conservative results. In addition, the table demonstrates that no updated support load exceeds any originally specified load. All calculated support loads in the table have been evaluated and found to be acceptable.

Figure 5.4A-15 shows vessel vertical support reactions both with and without the internal loads for the original 350 in.^{2} RV inlet nozzle guillotine analysis. The major effect of the internal loads on the vertical supports is to increase the maximum compressive reaction by a factor of three.

Figure 5.4A-16 shows horizontal support reaction with and without the internal loads for the same analysis. The major effect of the internal loads on the horizontal supports is to shift the phasing of the load response. The maximum horizontal support load with internal applied loads is not greater than the maximum horizontal support load without internal

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applied loads, but they occur at different times. The effect is due to the combination of a relatively low ratio of horizontal concrete stiffness to vertical concrete stiffness (under the support pads). This combination causes the pads to take vertical load in resisting the overturning effects of the horizontal forces on the vessel.

Inside the vessel, interface gaps are found to open and close many times. It can be seen from Figure 5.4A-15 that the total vertical effect of all loads on the system can cause the vessel to overcome its compressive dead weight and thermal growth loads on the support pad, traverse the vertical support gap, and engage the anchor bolts. The vessel does lift off, from as many as all four pads simultaneously, but most often it remains in contact, oscillating on one or more compressed pads. 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 oscillating about a deflected, closed gap position.

REFERENCES: SECTION 5.4A

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

2. CENPD-252-P-A, Topical Report, "Method for the Analysis of Blowdown Induced Forces in a Reactor Vessel," July 1979.

3. ICES STRUDL II. The Structural Design Language Engineers Users Manual, M.I.T. Press.,Cambridge, Massachusetts, 1968.

4. Fritz, R. J. and Kiss, E., The Vibration Response of a Cantilevered Cylinder Surrounded by an Annular Fluid KAPL-M-6539m Knolls Atomic Power Laboratory, Schenectady, New York.

5. CENPD-168A, Topical Report on Design Basis Pipe Breaks, June 1977.

6. Lien, J. S., R. P. Kassawara, H. B. Smith, Dynamic Analysis of Piecewise Liner Structures, ASCE Specialty Conference, New Orleans, Louisiana, 1975.

7. Kassawara, R. P., and Peck, D. A., Dynamic Analysis of Structural Systems Excited at Multiple Support Locations, SCE Specialty Conference, Chicago, Illinois, 1973.

TABLE 5.4A-1

RESULTS OF LOCA ANALYSIS OF REACTOR VESSEL SUPPORTS

Maximum LOCA Absolute Reactor Vessel Support Loads

Accident	Force Component (See Figure 5.4A-14) H	V (pad)	V (bolts)
100 in^2 RV Outlet Nozzle Guillotine(1)	2721	2001	514
350 in^2 RV Inlet Nozzle Guillotine(1)	3759	5160	2106
350 in^2 RV Inlet Nozzle Guillotine(2)	3625	3225	1403
600 in^2 SG Inlet Nozzle Guillotine(2)	1566	2255	145
Design Maximum LOCA Loads	5426	5200	4530

(1) Using condensed RCS model and internal loads from Reference 1 methodology

(2) Using full RCS model and internal loads from Reference 2 methodology

TABLE 5.4A-2

REACTOR COOLANT SYSTEM MODEL

Component	Mass Point Number	Degrees of Freedom	Hydrodynamic Coupling	Active Directions	
	9918	XYZ	9918-18		
	9913		9999-16		
Reactor			9909-9	XZ	
			9905-5		
Vessel	9909		9914-14		
	9905	XZ			
	9914				
	9999				
	18		Gap Locations	Degrees	Initial
	9	XYZ		of Freedom	Condition
Core	13		9918-18		
Support			9918-38	Y	Closed with
Barrel	5		18-38		Compressive
	14	XZ	28-29		Preload
	16		20-2		
Lower	2	XYZ	9914-14	XZ	Open, Gap ±
Support			9999-16	X	
Structure					
	20				
	24	XYZ			
Fuel	28				
Assembly	22	XZ			
	26				
Core	6	XYZ			
Shroud	10	XZ			
Upper	38	XYZ			
Guide	32				
Structure	29				

TABLE 5.4A-3

PIPE BREAK SUMMARY

Pipe	Location	Break Type Description	Flow Area (Sq. in..)
	R.V. terminal end	circumferential break	100
hot leg	S.G. terminal end	circumferential break	600
	R.V. terminal end	circumferential break	350
discharge leg	Pump terminal end	circumferential break	480
	Pump terminal end	circumferential break	430
suction leg	Pump elbow	slot $\pm$ 90° from elbow crotch	532
	S.G. elbow	slot $\pm$ 90° from elbow crotch	532
	S.G. terminal end	circumferential break	592

Note: Flow areas listed are total area available for flow from both sides of a given break location.

TABLE 5.4A-4

EXPANDED REACTOR COOLANT SYSTEM MODEL

350 IN.^{2} RV INLET NOZZLE BREAK

In addition to those mass degrees of freedom shown in Table 5.4A-2, the following were included for this break:

Component	Mass Point Number	Degrees of Freedom
R.C. Pumps (in intact legs)*	1103 2103 5103	XYZ
	1101 2101 5101	XZ
Intact R.C. Piping*	1760 2760 5760	XYZ
	1580 2580 5580	XYZ
	800 3800	XYZ
SG	409 3409	XZ
	404 3404	XYZ

* For break postulated at the 2B RV inlet nozzle.

TABLE 5.4A-5

EXPANDED REACTOR COOLANT SYSTEM MODEL

600 IN.2 S.G. INLET NOZZEL BREAK

In addition to those mass degrees of freedom shown in Tables 5.4A-2 and 5.4A-4, the following were included for this break:

Component	Mass Point Number	Degrees of Freedom
Reconnected	4103	XYZ
2B R.C. Pump	4101	XZ
and Piping	4760	XYZ
	4580	XYZ
Severed	3851	
Hot Leg	3850	XYZ
	38501	
	3550	XY
	3751	