POWER AUTHORITY OF THE STATE OF NEW YORK

10 COLUMBUS CIRCLE NEW YORK, N. Y. 10019
(212) 397-6200

May 16, 1980
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Director of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Operating Reactors

Subject: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
Additional Information on WCAP-9117

Dear Sir:

This letter responds to Mr. Schwencer's April 16, 1980 letter requesting additional information to continue the review of WCAP-9117, "Analysis of Reactor Coolant System for Postulated Loss of Coolant Accident: Indian Point 3 Nuclear Power Plant," which was submitted on June 15, 1977 by Consolidated Edison.

The Commission's April 16, 1980 questions and the Authority's responses to each are provided in Attachment 1.

Very truly yours,

George M. Wilverding
for Paul J. Early
Vice President and
Assistant Chief Engineer-Projects

cc: Mr. T. Rebelowski, Resident Inspector
U. S. Nuclear Regulatory Commission
P. O. Box 38
Buchanan, New York 10511

Mr. Doug Gaynor
Con Edison
4 Irving Place
New York, New York 10003

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ATTACHMENT 1

A) With regard to the reactor cavity pressurization analysis discuss the following items:

1. Provide supporting evidence for the conservation or accuracy of the insulation (both adjacent to the vessel and on the RCS piping) assumptions. Was the insulation thickness added to the reactor vessel radius for calculation of areas used in the calculation of forces on the reactor vessel?

Response:

Because of high pressures near the break the insulation around the break, specifically on the broken loop nozzle and piping, was assumed to totally crush. All other insulation was presumed intact. There is normally a 1/2" gap between the reactor vessel and the insulation. No credit was taken for the reactor vessel insulation moving against the reactor vessel post LOCA. The surface area of the insulation, which is slightly higher than the projected area, was used to convert pressures into forces on the reactor vessel.

If the projected area of the reactor vessel was used instead of the surface area of the insulation approximately 15% margin could be removed from the analysis.

2. Verify the accuracy of the assumed flow path direction in the TMD model.

Response:

The flow direction for flow paths near the break (i.e., out of the inspection port, out of the pipe sleeve, and towards the reactor vessel) is known. The flow direction for paths inside the cavity near the break is also known. K factors were calculated based on the assumed flow directions. Directional changes in flow away from the break could possibly occur. The effect of these changes on the overall results is insignificant because of the relative unimportance of paths far removed from the break and because flow is often sonic. (Flow is essentially independent of the K factor in the sonic flow regime.) This phenomenon was investigated on applications submitted after the transmittal of WCAP-9117.

3. Discuss the rationale for not using the location of the neutron detectors as control volume boundaries. Were the flow losses past the detectors included in the presented cavity model?

Response:

The excore instrumentation cutouts were conservatively neglected on this application. The IP3 model presumed no flow into or out of the excore volumes. The reactor cavity liner was assumed to have a uniform radius measured from the reactor vessel to the face of the excore volume closest to the reactor vessel. Work performed subsequent to the submittal of WCAP-9117 showed the loads could be reduced by approximately 10% if venting into and out of the excore volumes was considered.

4. Discuss the effects of the inspection port plugs as missiles. Clarify how the additional vent area (after plug movement) is accounted for. Verify that there will be no other shielding or structural components, e.g., cover plate, shielding to inhibit plug movement after the plug is accelerated.

Response:

The design of the replacement inspection port plugs is not complete as yet. However, the shielding material which is presently proposed to be utilized is Reactor Experiments, Inc. type 236 Boro-Silicone. This is a very resilient material which will minimize any possible damage in the event of missile generation. In addition, the Boro-Silicone is planned to be installed as 2" cube blocks such that in the event of a pipe break the Boro-Silicone blocks will be forced out of the plug area.

During the design of the replacement inspection port plugs, the Authority will consider the effects of the plug material as missiles.

The R in equation 3-2 of WCAP-9117 accounts for any drag resistances that may be present. The right hand side of equation 3-2, however, is essentially controlled by the ΔP term. The velocity profile of the plug was assumed to be flat, which is conservative. No flow through the ports is permitted until the plugs are slightly above the top of the concrete. This is also conservative as flow will initiate the same instant the plug starts to accelerate. The blow-out equation probably induces approximately 10% margin in the analysis.

The Authority will ensure that there will be no other shielding or structural components which will inhibit the movement of the Boro-Silicone blocks. However, hinged doors are presently proposed to be installed over the plugs to protect personnel walking in the area during shutdown. The hinged door will be designed such that it does not become a secondary missile or restrict plug movement.

5. Discuss how uncertainties in the TMD input data, due to imprecise dimensions, recent design changes, and modeling capability, are accounted for.

Response:

It is common practice to use the best information available. If for some reason complete data or drawings are not available to sufficiently define some input, it is Westinghouse policy to apply margin consistent with the induced uncertainties. All recent design changes were included in the cavity modelling for this application.

6. Provide justification for the assumption that all break flow enters node 1 and that none of the blowdown immediately enters by direct jetting into a volume adjacent to the reactor vessel.

Response:

Since the axial and radial separation at the broken loop safe end nozzle weld is limited by the primary shield wall restraints to less than the wall thickness of the pipe, no jetting into the cavity or into the steam generator compartments is possible. The blowdown emanates from the break in a 360^{o} fan jet. With this profile any jetting would impinge on the inspection plug or thru the inspection port. If credit for this is taken the loads would be reduced by at least 10%.

7. Discuss how loads resulting from differential pressures acting across the vessel piping and nozzles were accounted for in the reactor vessel load development. If those loads were not considered then provide justification for their elimination.

Response:

The pressure force acting on the nozzles in the reactor cavity was included in the evaluation by multiplying the appropriate differential pressure and the proper area. In this tight cavity arrangement most of the nozzle and all of the Reactor Coolant System piping is outside the cavity and would have little effect on the vessel loadings.

8. Provide a discussion of the methodology used to calculatte the mass and energy release data which is input to the reactor cavity pressure analysis. Include references to any applicable topical reports.

Response:

The methodology used by the Satan V computer program in calculating the mass and energy release rates input to the reactor cavity analysis is described in WCAP-8264-P-A, June 1975 (Proprietary) WCAP-8312-A, June 1975 (non-proprietary) .

B) MULTIFLEX

1. How was the limited displacement break simulated using MULTIFLEX? Specifically, the limitation noted on Page 13 of the NRC Safety Evaluation Report on WCAP-8708 should be addressed.

Response:

The simple slot break model (orifice break model, designated as 0-type in MULTIFLEX program) was used to simulate the limited-displacement area, guillotine-type rupture at the safe end of the RPV inlet nozzle. This break model has been reviewed by the NRC and has been shown to be acceptable.

2. What lower plenum radial distance was utilized in the MULTIFLEX representation of the low plenum region of the Indian Point 3 model? Specifically, the comment, Item 2, on Page 12 of the NRC SER on WCAP 8708 should be addressed.

3. What sonic velocity was used in the MULTIFLEX of the Indian Point 3 model? Specifically, the comment, Item 1, on Page 11 of the NRC SER on WCAP 8708 should be addressed.

Response (2&3):

In the Indian Point No. 3 MULTIFLEX hydraulic model, the equivalent piping network representation of the lower plenum region does not correspond to the NRC modified lower plenum model described on page 12 of the Safety Evaluation Report. Also, the sonic velocity data utilized in the Indian Point No. 3 MULTIFLEX analysis corresponds to the original, uncorrected wave speed.

Parametric studies performed during the NRC review of the MULTIFLEX code demonstrated, however, that the inclusion of a more representative radial transport distance in the lower plenum model and the utilization of updated sonic velocity data results in a maximum of 5% increase in peak total horizontal hydraulic force (THF) on the vessel wall.

In order to further demonstrate the small effect these two modifications have on the hydraulic force, we have performed additional sensitivity studies which consider a limited-displacement area, guillotine-type break at the RPV inlet nozzle. The results of these parametric studies are shown in Figures 1 through 3, which are the THF time-history plots for the three different cases. The peak THF's and the corresponding times they are attained in the transient are given below:

Figure	THF (x 10^6 lb_f)	Time (Msec)
1	6.786	24.9
2	6.689	22.6
3	7.216	22.5

Comparing Figures 1 and 2 shows that the utilization of the corrected sonic velocity data produced a small reduction (∿1.4%) in the maximum THF. A comparison of Figures 1 and 3 demonstrates that the inclusion of a more representative lower plenum model and updated sonic velocity data results in a small increase (∿6.3%) in the peak THF. Notice, also, that in all three cases the major forcing function frequency is unchanged.

The results of these additional sensitivity studies support the findings disclosed during the NRC review of the MULTIFLEX Code, i.e., they further demonstrate that consideration of the above two MULTIFLEX modifications has an insignificant effect on the magnitude and frequency of the hydraulic forcing function. In addition, since the internals forces are one of three different inputs required for the evaluation of the vessel support loads and fuel grid impact loads, this means that these two MULTIFLEX modifications would produce an even smaller effect on the resultant mechanical loads.

4. With respect to Item 3, on Page 12 of the NRC SER on WCAP 8708, justify the use of the 5 lump mass model in Indian Point Unit 3.

Response:

As was the case for the above two modifications, during the NRC review of the MULTIFLEX program a parametric study was performed demonstrating that the predicted results (magnitude and frequency of the forcing functions, core barrel displacements) from the original five mass points calculation are in very close agreement with the results of a ten mass points computation. The excellent agreement between the two different hydraulic forcing functions can be fully appreciated from Figure 4, where both force time-histories have been plotted on the same graph. There is less than 4.0% difference in the maximum THF, while the forcing function frequencies are essentially identical.

Thus, the MULTIFLEX five mass points beam representation of the core support barrel can be considered satisfactory, when compared to the predicted results of a beam model incorporating twice as many mass points.

5. What is the total overall effect on the structural response (internal components) if the Indian Point 3 model is different from the NRC SER on WCAP-8708 with respect to the Items listed above (2), (3) and (4). The response to this request should include the effect of Items 2, 3 and 4 on the amplitude and frequency of the core plates. Please be quantitative.

Response:

The effect of Items (2), (3), and (4) on the total overall structural response (internals components) has not been specifically evaluated for the Indian Point No. 3 plant. Studies with parametric variations for 2 loop, 3 loop and 4 loop plants have demonstrated consistency in the changes in internals forces. That is, percentage changes in internals forces for a parametric variation in hydraulic modeling are not dependent upon the reactor configuration. Therefore, the trends observed in WCAP-8708 can be used to predict results for the Indian Point No. 3 plant. The core plate motions for the Indian Point No. 3 plant would not significantly change (less than 10%) with the inclusion of the suggested modifications. The allowable peak grid load has incresed since WCAP-9117 by 12%, based upon recent grid crush tests.

C) 1. Was fuel analysis performed according to WCAP-8326?

Response:

The reactor core model used in the analysis was essentially the same as the model presented in WCAP-8236 which was reviewed and approved by the NRC. A bilinear representation of the grid impact (stiffness) element was incorporated into the basic model in order to assess the grid information.

2. What is the increase in peak clad temperature calculated assuming theoretical deformation on the spacer grid? What load is required to achieve theoretical maximum deformation?

Response:

The peak clad temperature assuming the theoretical maximum deformation of the spacer grid will be less than 20°F higher than that reported in the latest Indian Point Unit 3 FSAR analysis. (Calculated for the Diablo Canyon Units.) Since the grid loads predicted analytically exceeded the grid strength, the displacements obtained from the bilinear grid elements were used to estimate the grid permanent deformation. The total estimated inelastic deformation for the largest grid load was .046 inches - the deformation necessary to collapse one cell to the theoretical limit of 22% blockage is .064 inches. The peak clad temperature calculations, reported in WCAP-9117, were based on the assumption that two adjacent rows were fully (22%) collapsed; consequently, a significant margin in grid coolable geometry is still maintained.

3. Provide justification that the effect of crushing the grids for fuel bundles in the core of Indian Point 3 will not increase the peak cladding temperature of the hot rod by more than 20°F during a loss-of-coolant accident.

Response:

In WCAP-9117, reference was made to several calculations that were performed by Westinghouse to assess the impact of partially deformed grids on calculated peak clad temperature during a postulated loss-of-coolant accident. In these calculations, the maximum possible grid deformation, resulting in a 22% reduction in flow area, was assumed. The study focused on the reflood portion of a large break LOCA, per agreement with NRC staff and resulted in less than 20°F increase in peak clad temperature relative to the case without grid deformation. In all cases referred to by WCAP-9117, the peak clad temperature occurred some 100-seconds into the reflood transient at a non-burst elevation.

Since those calculations were performed, the NRC staff has recommended a minor modification to the way the clad heatup calculation, due to deformed grids, is performed. However, a quick review of the latest Indian Point Unit #3 ECCS analysis reveals that no impact is expected if fuel grids were to deform during a LOCA. This is because the peak clad temperature occurs at the burst node for Indian Point Unit #3. The sensitivity studies cited above were for non-burst node limited plants. At the burst node, the clad temperature turns around just 10 seconds after bottom of the core recovery compared to 100 seconds for the non-burst node. This is true because burst occurs before bottom of the core recovery and strains the clad significantly away from the pellet such that only small fluid heat transfer coefficients are required to turn clad temperature around. Since the impact of grid deformation is to slightly reduce the reflood heat transfer coefficients, it is expected to have only a small impact on the burst node, which only needs low heat transfer coefficients. Also, for all break sizes, the worst non-burst node peak clad temperature for Indian Point Unit #3 is 85°F less than the burst node.

Thus, it is certain that deformed grids will have less than a 20°F impact on the calculated peak clad temperature for Indian Point Unit #3.

D) In the steam generator and pump supports, was the possibility of plastic collapse addressed for the members that yielded? If not what is the basis for neglecting this effect?

Response:

The effect of the analysis support member yeilding was included by modelling these highly loaded members as plastic spars, pipes, and wide flange members in the combined RCL/support model.

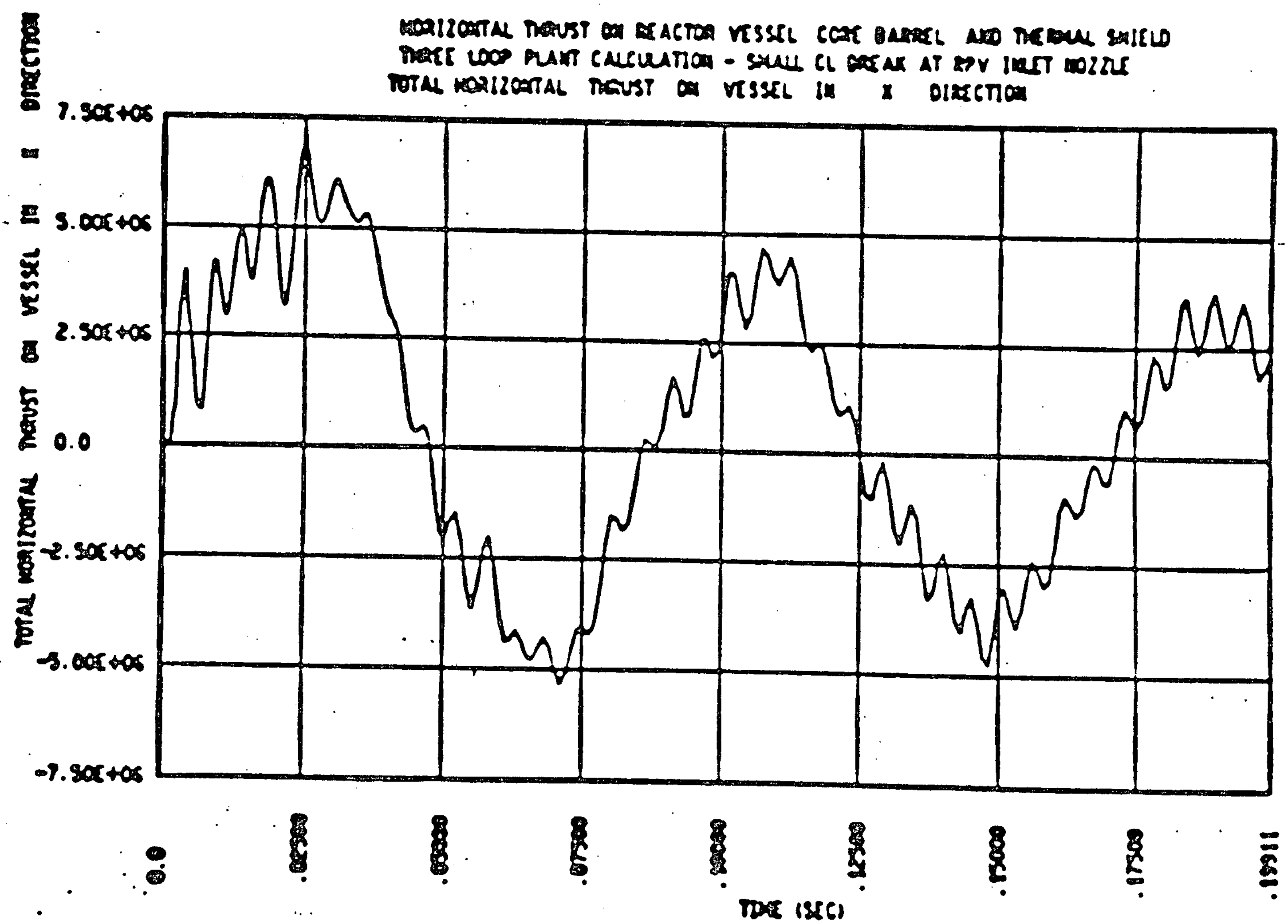
E) What acceptable criteria was employed for the shield wall concrete evaluation?

Response:

For the concrete evaluation, yield strength was used as the criteria for evaluation of rebar. Shear allowables for concrete were determined by ACI-318.

FIGURE 1

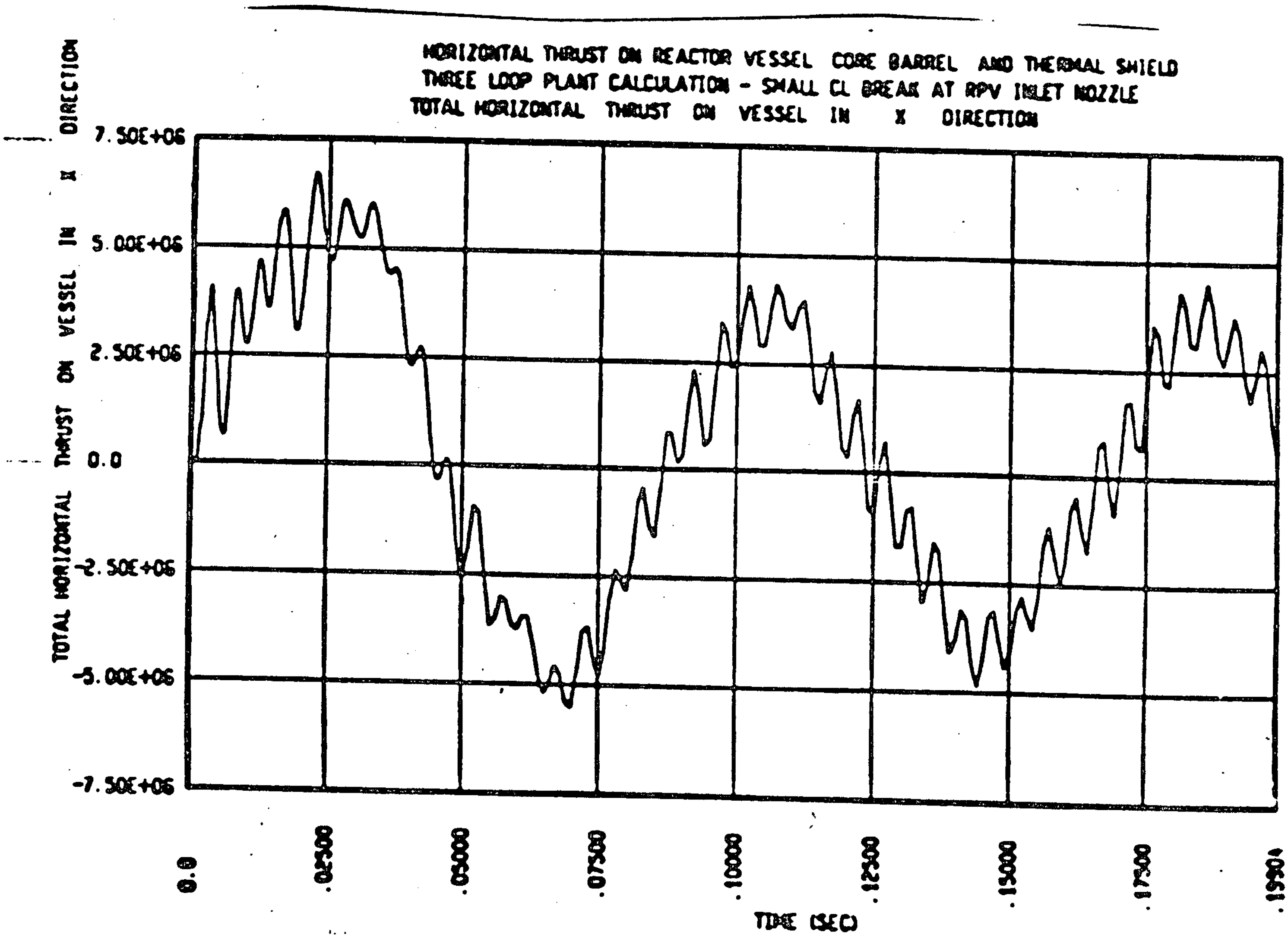
TOTAL HORIZONTAL FORCE ON VESSEL WALL



- 144 in^2 RPV inlet nozzle BK
- MULTIFLEX / 5 mass pts. / 1 msec BOT
- Original sonic velocity data
- Original lower plenum model

FIGURE 2

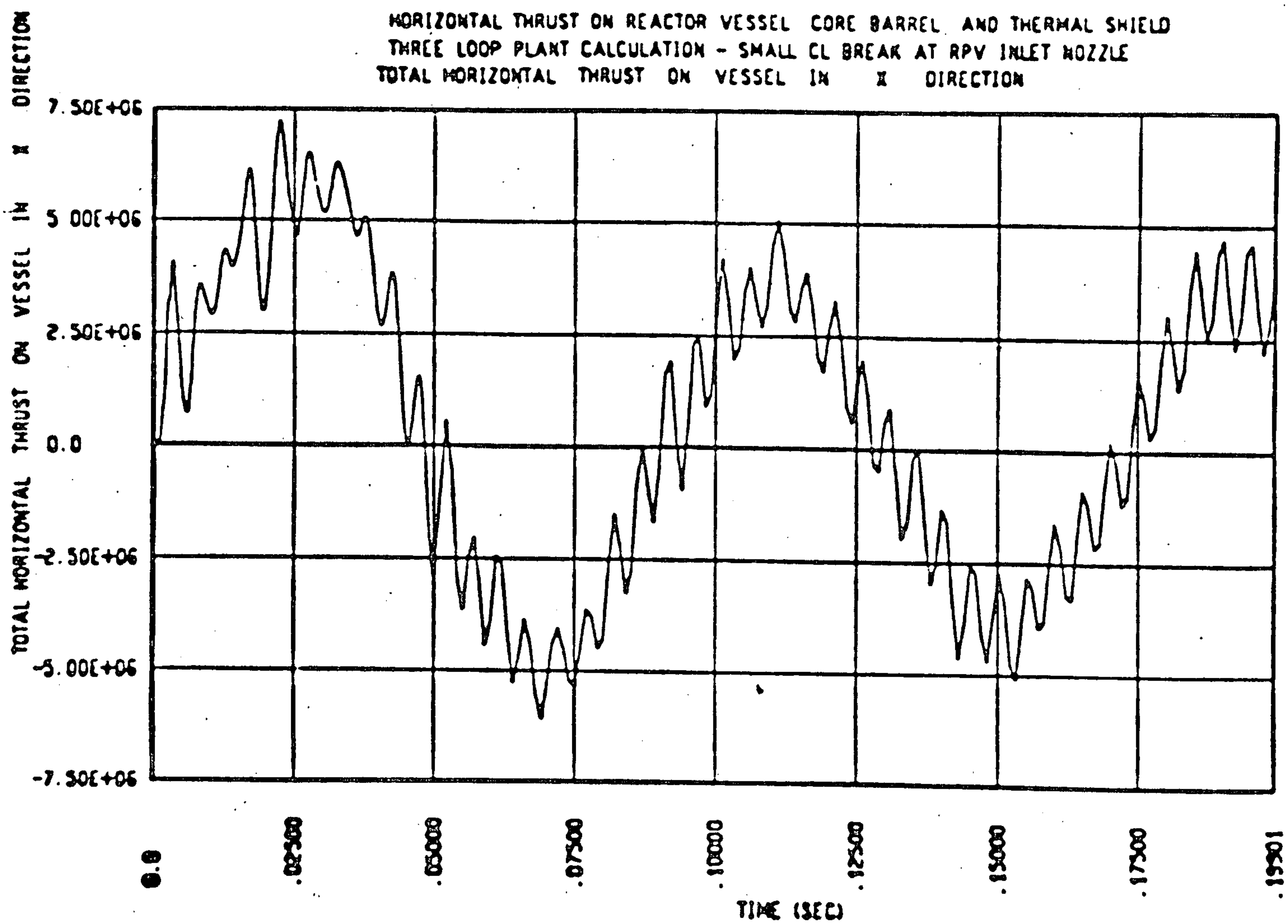
TOTAL HORIZONTAL FORCE ON VESSEL WALL



- 144 in^2 RPV inlet nozzle BK
- MULTIFLEX / 5 mass pts. / 1 msec BOT
- Updated sonic velocity data
- Original lower plenum model

FIGURE 3

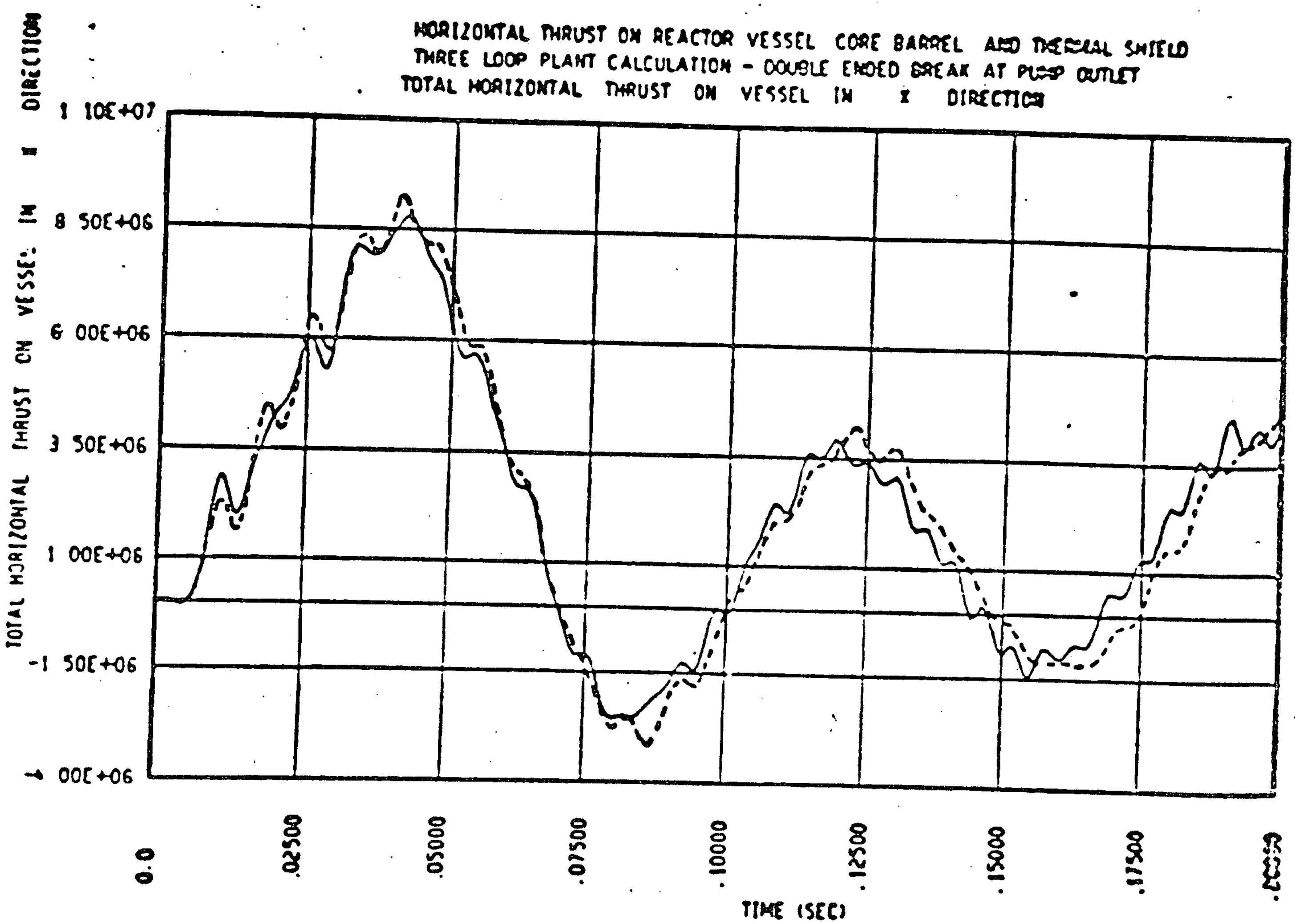
TOTAL HORIZONTAL FORCE ON VESSEL WALL



- 144 in^2 RPV inlet nozzle BK
- MULTIFLEX / 5 mass pts. / 1 msec BOT
- Updated sonic velocity data
- Modified lower plenum model

FIGURE 4

COMPARISON OF TOTAL HORIZONTAL FORCE ON VESSEL WALL
FOR 5 MASS POINTS AND 10 MASS POINTS BEAM MODEL



5 Mass Points Calculation

10 Mass Points Calculation