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J. G. Herbein

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ENCLOSURE:

Consists of requested info. concerning the potential for Lamellar tearing of steam generator & reactor coolant pump support materials...w/att drawings.......

(6P)

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January 31, 1978
GQL 0148

Mr. R. W. Reid, Chief
Operating Reactors Branch No. 4
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Sir:

Three Mile Island Nuclear Station Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289

In response to your letter of September 14, 1977, and our letter of November 18, 1977 (GQL 1529), we have attached the requested information concerning the potential for Lamellar tearing of steam generator and reactor coolant pump support materials.

The TMI steam generators are supported at the bottom by the support skirt which is welded to the lower head. This skirt is attached to a base plate which is supported by the concrete foundation. The support near the top of the generators was not provided by B&W and is not being considered in this evaluation.

In evaluating the potential for Lamellar tearing, the skirt is not a primary concern as it is not subjected to major loading. This is because of the simplicity of its geometry and the minimizing of the large intersecting members. The weld areas of the skirt were preheated before welding; the finished assembly was post-heat treated and all welds were either magnetic particle and/or radiographically examined.

The TMI-1 reactor coolant pumps are completely supported by the reactor coolant piping which is supported from the reactor vessel and steam generators. Supplementary supports for the piping or reactor coolant pumps are not used. Therefore, no response is required.

We have not submitted our own evaluation of the fracture toughness of the steam generator and reactor coolant pump support materials per a telephone conversation of October 20, 1977 between R. Snaider of your office and D. G. Mitchell and E. G. Skuchas of Met-Ed.

Sincerely,

1493 154

J. G. Herbein
Vice President-Generation

Attachment

JGH:DGM:cjg

780380171

Question 1:

Provide engineering drawings of the steam generator and reactor coolant pump supports sufficient to show the geometry of all principal elements. Provide a listing of materials of construction.

Response:

Drawings enclosed:

a. Number 131130E - Assembly and Detail of Support Skirt
b. Number 131104E - List of Material/Steam Generator
c. Number 131112E - Shell and Tubesheet Attachment Assembly

Question 2:

Specify the detailed design loads used in the analysis and design of the supports. For each loading condition (normal, upset, emergency and faulted), provide the calculated maximum stress in each principal element of the support system and the corresponding allowable stresses.

Response:

	Normal & Upset		Emergency		Faulted	
Load (Fy Kip; Mr Kip in.)	1801	48,000	1911	96,000	2046	306,000
Max. Stress (Kip/in^2)	*Neg.		*Neg;	*Neg.	4.3	20.43
Allowable (Kip/in2)	.5(3Sm)	8r 33.9	.5(Sy) or 18		.5(1.2Sy) or 21.6	
			.5(1.5Sy) or 27		.5(1.8Sy) or 32.4	
					.5(1.8Sy) or 32.4	

*Indicates that the intersecting members are in a compressive state.

Question 3:

Describe how all heavy section intersection m[illegible]ber weldments were designed to minimize restraint and lamellar tearing. Spec[illegible]fy the actual section thicknesses in the structure and provide details of [illegible]pical joint designs. State the maximum design stress used for the through-thickness direction of plates and elements of rolled shapes.

Response:

There is no major loading on the steam generator skirt to show cause for concern about Lamellar tearing. See Item 1 for structure details. See No. 2 for design stress.

Question 4:

Specify the minimum operating temperature for the supports and describe the extent to which material temperatures have been measured and at various points on the supports during the operation of the plant.

Response:

No minimum operating temperature is specified for the steam generator supports. No temperatures have been measured on the supports of the steam generators during operation of the plant.

Question 5:

Specify all the materials used in the supports and the extent to which mill certificate data are available. Describe any supplemental requirements such as melting practice, toughness tests and through-thickness tests specified. Provide the results of all tests that may better define the properties of the materials used.

Response:

a. The materials are listed in the List of Materials Drawing.
b. Mark 96 has some mill certification information available which will be forwarded at a later date.
c. This support material was ordered in accordance with the requirements on the ASME Code, Section II.
d. No data from toughness or through wall tests is available.

Question 6:

Describe the welding procedures and any special welding process requirements that were specified to minimize residual stress, weld and heat affected zone cracking and lamellar tearing of the base metal.

Response:

a.

Weld Number	Process	Preheat (Min.)
61	Submerged Arc & Manual Metal Arc	200°F
64	Submerged Arc	200°F
65	Flux Core	200°F
70 & 71	Flux Core	200°F

b. These welds all received a full Section III stress relief.

Question 7:

Describe all inspections and non-destructive tests that were performed on the supports during their fabrication and installation, as well as any additional inspections that were performed during the life of the facility.

Response:

The Non-Destructive Examinations are shown on Drawing Number 131130E and described on Table I.

TABLE I

Response to NRC Question No. 7

Non-Destructive Examinations For Welds In Support Skirt Structure

Weld and Location	Drawing, View or Section	Non-Destructive Examination
No. 61, Support skirt (96) to transition ring (95)	131112, Detail D	Radiographic, Magnetic Particle
No. 64, Support skirt (96) assembly	131130, Detail J	Radiographic, Magnetic Particle
65, Support skirt (96) to base plate (97)	131130, Detail M	Magnetic Particle
70, Gusset plates (98) to support skirt (96)	131130, Detail K	Magnetic Particle
No. 71, Gusset plates (98) to base plate (97)	131130, Section FF	Magnetic Particle

1493 157

Please respond within 30 days of receipt of this letter, indicating your intent to proceed with an evaluation of the overall asymmetric loss of coolant accident (LOCA) loads as described herein. In addition, please submit to us, within 90 days, your detailed schedule for providing the required evaluation. Your schedule should be consistent with our desire to resolve this problem within two years and should clearly state your intent to demonstrate the safety of long term continued operation.

We are transmitting information copies of this letter to the Westinghouse, Combustion Engineering and Babcock & Wilcox Companies. If you have any questions or want any clarification on this matter, please call your NRC Project Manager.

Copies of this letter are being sent to all addressees on the current service lists for each docket.

Sincerely,

Victor Stello, Jr., Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

Enclosures:
1. Background and Current Status
2. Revised Request for Additional Information

cc w/enclosure:
See attached listing

1493 158

January 25, 1978

ENCLOSURE 1

BACKGROUND AND CURRENT STATUS OF THE NRC STAFF REVIEW OF ASYMMETRIC LOCA LOADS FOR PWR FACILITIES

On May 7, 1975, the NRC was informed by Virginia Electric & Power Company that an asymmetric loading on the reactor vessel supports resulting from a postulated reactor coolant pipe rupture at a specific location (e.g., the vessel nozzle) had not been considered by Westinghouse or Stone & Webster in the original design of the reactor vessel support system for North Anna, Units 1 and 2. It had been identified that in the event of a postulated instantaneous, double-ended offset LOCA at the vessel nozzle, asymmetric loading could result from forces induced on the reactor internals by transient differential pressure across the core barrel and by forces on the vessel due to transient differential pressures in the reactor cavity. With the advent of more sophisticated computer codes and the accompanying more detailed analytical models, it became apparent that such differential pressures, although of short duration, could place a significant load on the reactor vessel supports and on other components, thereby possibly affecting their integrity. Although this potential safety concern was first identified during the review of the North Anna facilities, it has generic implications for all PWRs.

Upon closer examination of this situation, it was determined that postulated breaks in a reactor coolant pipe at vessel nozzles were not the only area of concern but rather that other pipe breaks in the reactor coolant system could cause internal and external transient loads to act upon the reactor vessel and other components. For the postulated pipe break in the cold leg, asymmetric pressure changes could take place in the annulus between the core barrel and the vessel. Decompression could occur on the side of the vessel annulus nearest the pipe break before the pressure on the opposite side of the vessel changes. This momentary differential pressure across the core barrel could induce lateral loads both on the core barrel and on the reactor vessel. Vertical loads could also be applied to the core internals and to the vessel due to the vertical flow resistance through the core and asymmetric axial decompression of the vessel. Simultaneously, for vessel nozzle breaks, the annulus between the reactor and biological shield wall could become asymmetrically pressurized resulting in a differential pressure across the vessel causing additional horizontal and vertical external loads on the vessel. In addition, the vessel could be loaded by the effects of initial tension release and blowdown thrust at the pipe break. These loads could occur simultaneously. For a reactor vessel outlet break, the same type of loadings could occur, but the internal loads would be predominantly vertical due to more rapid decompression of the upper plenum.

Although the NRC staff's original emphasis and concern were focused primarily on the integrity of the reactor vessel support system with respect to postulated breaks inside the reactor cavity (i.e., at a nozzle), it has since become apparent that significant asymmetric forces can also be generated by postulated pipe breaks outside the cavity and that the scope of the problem is not limited to the vessel support system itself. For such outside-cavity postulated breaks, the aforementioned concerns, such as the integrity of fuel assemblies and other structures, need to be examined.

In June 1976, the NRC requested all operating PWR licensees to evaluate the adequacy of the reactor system components and their supports at their facilities with respect to these newly-identified loads.

In response to our request, most licensees with Westinghouse plants proposed an augmented inservice inspection program (ISI) of the reactor vessel safe-end-to-end pipe welds in lieu of providing an evaluation of postulated piping failures. Licensees with Combustion Engineering plants submitted a probability study (prepared by Science Applications, Inc.) in support of their conclusion that a break at a particular location (vessel nozzle) has such a low probability of occurrence that no further analysis is necessary. A similar study has been recently submitted by Science Applications, Inc. (SAI) for B&W plants.

When the Westinghouse and CE owners group reports were received in September 1976, the NRC formed a special review task group to evaluate these alternative proposals. In addition, EG&G Idaho, Inc., was contracted to perform an independent review of the SAI probability study submitted for the CE owners group.

This review effort resulted in a substantial number of questions which previously have been provided to representatives of each group. Based on the nature of these questions and other factors to be discussed later in this report, we cannot accept these reports in their present form as a resolution for the asymmetric LOCA load generic issue. Based on our review, we have concluded that a sufficient data base does not presently exist within the nuclear industry to provide satisfactory answers to these information needs. Several long-term experimental programs would be required to provide much of this information. Although the probability study recently submitted by SAI for certain B&W owners does respond to some of the informal questions raised during our review of the SAI report prepared by CE plants, the more fundamental questions remain. Therefore, this conclusion also applies to the SAI topical report for B&W plants (SAI-050-77-PA).

A second - and equally important - reason for not accepting probability/ISI approaches as a solution at this point concerns our need and industry's need to gain a better understanding of the problem. We consider it essential that an understanding of the important breaks and associated consequences be known before applying any remedy - be it pipe restraints, probability, ISI, or some combination of these measures. Only in this way will we have a basis on which to judge the importance of the remedy with respect to what it is designed to prevent.

Although we have many questions on each of these topical reports, this does not mean that we view the probabilistic/ISI approach as completely without merit. In fact, the results of a probabilistic evaluation serves as the basis for continued operation and licensing of nuclear plants during this interim period while additional evaluations can be performed by vendors and licensees.

We believe that the justification for continued plant operation has as its basic foundation the fact that the event in question, i.e., a hypothetical double-ended instantaneous rupture of the main coolant pipe at a particular location, has a very low probability of occurrence.

The disruptive failure probability of a reactor vessel itself has been estimated to lie between 10^{-6} and 10^{-7} per reactor year - so low that it is not considered as a design basis event. The rupture probability of pipes is estimated to be higher. WASH-1400 used a median value of 10^{-4} for LOCA initiating ruptures per plant-year for all pipes sizes 6" and greater (with a lower and upper bound of 10^{-5} and 10^{-3}, respectively). We believe that considering the large size of the pipes in question (up to 50" O.D. and 4-1/8" thick), the lower bound is more appropriate since these pipes are more like vessels in size. In addition, the quality control of this piping is the best available and somewhat better than that of the piping used in the WASH-1400 study.

These factors, coupled with the facts that (1) the break of primary concern must be very large, (2) it must occur at a specific location, (3) the break must occur essentially instantaneously, and (4) these welds are currently subject to inservice inspection by volumetric and surface techniques in accordance with ASME Code Section XI, lead us to conclude that the probability of a pipe break resulting in substantial transient loads on the vessel support system or other structures is acceptably small such that continued reactor operation and continued licensing of facilities for operation can continue while this matter is being resolved.

In support of the above, the staff has developed a short-term interim criterion to determine if an acceptable level of safety exists for operating PWRs under conditions of a postulated pipe break. This interim criterion is based on a simplified probabilistic model that incorporates elastic fracture mechanics techniques to estimate the probability of a pipe break. Critical flaw size and subcritical flaw growth rates were determined assuming the presence of a surface flaw located in a circumferential weld of a thick-walled pipe, Determination of the critical flaw size was based on an estimated fracture toughness value of K_{IC} at a minimum temperature of 200 F and a uniform tensile stress equal to the consideration of various operating conditions producing elastically calculated stresses ranging in value from 1 to 3 times the material minimum yield strength.

Then, using the calculated critical flaw size, the subcritical growth rate, and an estimated probability distribution of an undetected flaw in thick-walled pipe welds, the upper bound probability of pipe break was estimated to be 10^{-4}. This value is also supported by a recent publication by Dr. S. H. Bush* which states that actual failure statistics confirm rates of 10^{-4} to 10^{-6} per reactor-year in large pipes, with higher rates as the pipe size decreases. Considering these analyses, we conclude that our conservative estimate on a pipe break in the primary coolant system is in the range of 10^{-4} to 10^{-6}. This estimated pipe break probability is considered acceptably low to justify short-term operation of nuclear power plants.

In view of all previous discussions concerning this issue, the NRC staff has concluded that an evaluation must be undertaken to assess the design adequacy of the reactor vessel supports and other affected structures and systems to withstand asymmetric LOCA loads, including an assessment of the effects of asymmetric loads produced by various pipe breaks both inside and outside the reactor cavity. On performing these evaluations the staff will permit the grouping of plants, where adequate justification for such grouping exists, in order to limit the number of plants to be analyzed. Alternatively, the staff will permit the analyzing of a "prototypical" plant, which is sufficiently representative of a group of plants, to provide the necessary information. Both of these concepts have been discussed with the Westinghouse and CE Owners Groups, and we believe that such approaches could save a significant amount of time and effort in obtaining results on which to base any needed corrective measures. The NRC staff is prepared to meet with PWR licensees to discuss such approaches, and has already done so. For example, we met with the Westinghouse owners group on October 19, 1977 for the purpose of discussing a generic solution for breaks outside the reactor cavity. It is expected that a similar meeting will be held in the near

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*"Critical Factors in Blowdown Loads in the PWR Guillotine Nozzle Break (Volume 2 - the Asymmetric Load Problem)", dated June 6, 1977

future to address breaks located inside the cavity. This "phased" approach is acceptable to us, provided that it sheds light on and serves to expedite consideration of the more limiting inside-cavity breaks.

For your information, the NRC has a technical assistance contract with EG&G Idaho, Inc., to independently model representative Westinghouse, B&W, and CE plants for the purpose of assessing the loads on all major structures and components resulting from asymmetric LOCA loads. We believe that the results of this program which will include sensitivity studies, will provide significant confirmatory information related to this generic safety concern.

Although, as stated earlier, we believe that continued operation and licensing of facilities for the short-term is justified, we also believe that efforts to resolve this issue should proceed without delay, with the objective of both completing the necessary assessments and installing any necessary plant modifications within two years. In making this statement, we wish to make it clear that plant modifications, if indicated by licensee assessments, is the preferred approach. At the same time, we recognize that there may be cases wherein appropriate modifications may be judged to be unwarranted based on the consideration of overall risk. In such cases, and only in such cases, we will be prepared to give further consideration to alternate approaches, such as probability/ISI. We feel, however, that ISI techniques as they exist today could be considerably improved, and, to the extent that such improvements could have a direct bearing on this problem as well as an impact of nuclear safety in general, we would welcome their development.

1493 163

ENCLOSURE 2

REVISED REQUEST FOR ADDITIONAL INFORMATION

Recent analyses have shown that certain reactor system components and their supports may be subjected to previously underestimated asymmetric loads under the conditions that result from the postulation of ruptures of the reactor coolant piping at various locations. It is therefore necessary to reassess the capability of these reactor system components to assure that the calculated dynamic asymmetric loads resulting from these postulated pipe ruptures will be within the bounds necessary to provide high assurance that the reactor can be brought safely to a cold shutdown condition. For the purpose of this request for additional information the reactor system components that require reassessment shall include:

a. Reactor Pressure Vessel
b. Fuel Assemblies, Including Grid Structures
c. Control Rod Drives
d. ECCS Piping that is Attached to the Primary Coolant Piping
e. Primary Coolant Piping
f. Reactor Vessel, Steam Generator and Pump Supports
g. Reactor Internals
h. Biological Shield Wall and Neutron Shield Tank (where applicable)
i. Steam Generator Compartment Wall

The following information should be included in your reassessment of the effects of postulated asymmetric LOCA loads on the above-mentioned reactor system components and the reactor cavity structure.

1. Provide arrangement drawings of the reactor vessel, the steam generator and pump support systems to show the geometry of all principal elements and materials of construction.

2. If a plant-specific analysis will not be submitted for your plant, provide supporting information to demonstrate that the generic plant analysis under consideration adequately bounds the postulated accidents at your facility. Include a comparison of the geometric, structural, mechanical and thermal hydraulic similarities between your facility and the case analyzed. Discuss the effects of any differences.

3. Consider postulated breaks at the reactor vessel hot and cold leg nozzle safe ends, pump discharge nozzle and crossover leg that result in the most severe loading conditions for the above-mentioned

systems.* Provide an assessment of the effects of asymmetric pressure differentials on these systems/components in combination with all external loadings including asymmetric cavity pressurization for both the reactor vessel and steam generator which might result from the required postulate. This as essment should consider:

a. limited displacement break areas where applicable
b. consideration of fluid-structure interaction
c. use of actual time-dependent forcing function
d. reactor support stiffness
e. break opening times.

4. If the results of the assessment required by 3 above indicate loads leading to inelastic action in these systems or displacement exceeding previous design limits provide an evaluation of the following:

   a. Inelastic behavior (including strain hardening) of the material used in the system design and the effect on the load transmitted to the backup structures to which these systems are attached.

5. For all analysis performed, include the method of analysis, the structural and hydraulic computer codes employed, drawings of the models employed and comparisons of the calculated to allowable stresses and strains or deflections with a basis for the allowable values.

6. Provide an estimate of the total amount of permanent deformation sustained by the fuel spacer grids. Include a description of the impact testing that was performed in support of your estimate. Address the effects of operating temperatures, secondary impacts, and irradiated material properties (strength and ductility) on the amount of predicted deformation. Demonstrate that the fuel will remain coolable for all predicted geometries.

7. Demonstrate that active components will perform their safety function when subjected to the postulated loads resulting from a pipe break in the reactor coolant system.

8. Demonstrate functionability of any essential piping where service level B limits are exceeded.

In order to review the methods employed to compute the asymmetrical pressure differences across the core support barrel during subcooled portion of the blowdown analysis, the following information is requested:

*B&W and CE plant licensees should also consider breaks in the hot leg at the steam generator inlet.

1. A complete description of the hydraulic code(s) used including the development of the equations being solved, the assumptions and simplifications used to solve the equations, the limitations resulting from these assumptions and simplifications and the numerical methods used to solve the final set of equations. Provide comparisons with experimental data, covering a wide range of scales, to demonstrate the applicability of the code and of the modeling procedures of the subcooled blowdown portion of the transient. In addition, discuss application of the code to the multi-dimensional aspects of the reactor geometry.

   If an approved vendor code is used to obtain the asymmetric pressure difference across the core support barrel, state the name and version of the code used and the date of the NRC acceptance of the code.

2. If the assessment of the asymmetric pressure difference across the core support barrel is made without the use of a hydraulic blowdown code, present the methodology used to evaluate the asymmetric loads and provide justification that this assessment provides a conservative estimate of the effects of the postulated LOCA.

A compartment multi-node, space-time pressure response analysis is necessary to determine the external forces and moments on components. Analyses should be performed to determine the pressure transient resulting from postulated hot leg and cold leg reactor coolant system pipe ruptures within the reactor cavity and any pipe penetrations. If applicable, similar analyses should be performed for steam generator compartments that may be subject to pressurization where significant component support loads may result. This information can be provided to encompass a group of similarly designed plants (generic approach) or a purely plant specific (custom plant) evaluation can be developed. In either case, the proposed method of evaluation and principal assumptions to be used in the analysis should be provided for review in advance of the final load assessment.

For generic evaluations, perform a survey of the plants to be included and identify the principle parameters which may vary from plant to plant. For instance, this should include blowdown rate and geometrical variations in principle dimensions, volumes, vent areas, and vent locations. A typical or lead plant should be selected to perform sensitivity and envelope calculations. These analyses should include:

(1) nodal model development for the configuration representing the most restrictive geometry; i.e., requiring the greatest nodalization;

(2) the most restrictive configuration regarding vent areas and obstructions to flow should be analyzed; and,

(3) sensitivity to code data input should be evaluated; e.g., loss coefficients, inertia terms, vent areas, nodal volumes, and any other input data where there may be variations from plant to plant or uncertainty for the given plant.

These studies should be directed at evaluating the maximum lateral and vertical force and moment time functions, recognizing that models may be different for lateral as opposed to vertical load definitions.

The following is the type of information needed for both generic and custom plant evaluations. Although this request was primarily developed for reactor cavity analyses it may be applied to other component subcompartments by general application.

(1) Provide and justify the pipe break type, area, and location for each analysis. Specify whether the pipe break was postulated for the evaluation of the compartment structural design, component supports design, or both.

(2) For each compartment, provide a table of blowdown mass flow rate and energy release rate as a function of time for the break which results in the maximum structural load, and for the break which was used for the component supports evaluation.

(3) Provide a schematic drawing showing the compartment nodalization for the determination of maximum structural loads, and for the component supports evaluation. Provide sufficiently detailed plan and section drawings for several views, including principal dimensions, showing the arrangement of the compartment structure, major components, piping, and other major obstructions and vent areas to permit verification of the subcompartment nodalization and vent locations.

(4) Provide a tabulation of the nodal net-free volumes and interconnecting flow path areas. For each flow path, provide an L/A (ft^{-1}) ratio, where L is the average distance the fluid flows in that flow path and A is the effective cross sectional area. Provide and justify values of vent loss coefficients and/or friction factors used to calculate flow between nodal volumes. When a loss coefficient consists of more than one component, identify each component, its value and the flow area at which the loss coefficient applies.

(5) Describe the nodalization sensitivity study performed to determine the minimum number of volume nodes required to conservatively predict the maximum pressure load acting on the compartment structure. The nodalization sensitivity study should include consideration of spatial pressure variation; e.g., pressure variation circumferentially, axially and radially within the compartment. The nodal model development studies should show that a spatially convergent differential pressure distribution has been obtained for the selected evaluation model.

Describe the justify the nodalization sensitivity study performed for the major component supports evaluated, if different from the structural analysis model, where transient forces and moments acting on the components are of concern. Where component loads are of primary interest, show the effect of noding variations on the transient forces and moments. Use this information to justify the nodal model selected for use in the component supports evaluation.

If the pressurization of subvolumes located in regions away from the break location is of concern for plant safety, show that the selection of parameters which affect the calculations have been conservatively evaluated. This is particularly true for pressurization of the volume beneath the reactor vessel. In this case, a model which predicts the highest pressurization below the vessel should be selected for the evaluation.

NOTE: It has been our experience that for the reactor cavity, three regions should be considered (i.e., nodalized) when developing a total model. These are:

(1) the volume around or in the vicinity of the break location out to a radius approximated by the adjacent nozzles, and including portions of the penetration volume for some plants;

(2) the volume or region covering the upper reactor cavity, primarily the RPV nozzles other than the break nozzle; and

(3) the region encompassing the lower reactor cavity and other portions of the reactor cavity not included in Items (1) and (2).

(6) Discuss the manner in which movable obstructions to vent flow (such as insulation, ducting, plugs, and seals) were treated. Provide analytical and experimental justification that vent areas will not be partially or completely plugged by displaced objects. Discuss how insulation for piping and components was considered in determining volumes and vent areas.

(7) Graphically show the pressure (psia) and differential pressure (psi) response as functions of time for a representative number of nodes to indicate the spatial pressure response. Discuss the basis for establishing the differential pressure on structures and components.

(8) For the compartment structural design pressure evaluation, provide the peak calculated differential pressure and time of peak pressure for each node. Discuss whether the design differential pressure is uniformly applied to the compartment structure or whether it is spatially varied. If the design differential pressure varies depending on the proximity of the pipe break location, discuss how the vent areas and flow coefficients were determined to assure that regions removed from the break location are conservatively designed, particularly for the reactor cavity as discussed above.

(9) Provide the peak and transient loading on the major components used to establish the adequacy of the support design. This should include the load forcing functions (e.g., $f_x(t)$, $f_y(t)$, $f_z(t)$) and transient moments (e.g., $M_x(t)$, $M_y(t)$, $M_z(t)$) as resolved about a specific, identified coordinate system. The centerline of the break nozzle is recommended as the X coordinate and the center line of the vessel as the Z axis. Provide the projected area used to calculate these loads and identify the location of the area projections on plan and section drawings in the selected coordinate system. This information should be presented in such a manner that confirmatory evaluations of the loads and moments can be made.

1493 169

Metropolitan Edison Company

cc: G. F. Trowbridge, Esquire
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