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OCT 30 1972

G. Lainas, Chief, Containment Systems Branch, Directorate of Licensing
THRU: A. Schwencer, Chief, Pressurized Water Reactors Branch No. 4, Licensing

Original Signed by
Albert Schwencer

CONTAINMENT PEAK PRESSURE AND INTERNAL PRESSURE DIFFERENTIAL

In regard to Mr. R. C. DeYoung's technical assistance request (TAR) to Mr. J. M. Hendrie dated October 20, 1972, we understand that you will need additional information similar to that requested for the Kewaunee review to resolve the problems discussed in the TAR for Oconee and Three Mile Island Unit 1. In an effort to obtain this information, requests for additional information concerning the containment peak pressure problem have been transmitted to the applicants for Oconee and Three Mile Island Unit 1. Responses to these requests are expected by November 20, 1972 and will be forwarded to you upon receipt. For the second problem, containment internal pressure differentials, information needed for Oconee is presently available since this information was supplied during the original review. A copy of the appropriate FSAR pages is enclosed for your review. In regard to Three Mile Island Unit 1, a request for additional information relative to this problem is in preparation and will be transmitted to the applicant. Response to this request is also expected by November 20, 1972.

As was previously indicated in the October 25, 1972 notice we are holding a meeting with B&W on November 3, 1972 to discuss several ECCS and containment topics. In particular, B&W wishes clarification of our requests concerning the containment peak pressure problem. This would also be an excellent opportunity to discuss the containment internal pressure differentials problem relative to the enclosed Oconee FSAR material if your initial review of this material can be completed by this time.

Original Signed by
Donald K. Davis

POOR ORIGINAL

Donald K. Davis
Pressurized Water Reactors Branch No. 4
Directorate of Licensing

AO 50-269
for encl.

Enclosure:
FSAR pages

cc: R. C. DeYoung
R. Tedesco
A. Schwencer
H. Schierling
J. Peltier

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A breathing rate of 3.47×10^{-4} m^3/s is assumed for the 2-hour exposure. For the 30-day exposure, a breathing rate of 2.32×10^{-4} m^3/s is assumed.

The total integrated thyroid doses resulting from this LOCA fission product release are 6.0 Rem for the 2-hour exposure at the 1 mile exclusion distance, and 5.5 Rem for the 30-day exposure at the 6 mile low population distance. The corresponding whole body doses are 0.014 Rem and 0.018 Rem. These doses are shown in Table 14-17.

The effect on the dose from an LOCA release was also investigated in the event a LOCA occurred while the reactor building was being purged. Assuming that all the fission products had been initially released to the reactor building, which was at a constant pressure of 59 psig, the flow from the purge vents would resemble flow through a long duct. With an isolation valve closing time of 5 seconds and with no credit taken for flow restrictions due to the closing valve, additional dose equivalent curies of I 131 will be released. This release corresponds to a 0.05 Rem additional roid dose at the site boundary.

14.2.2.3.8 Pressure Buildup In Primary And Secondary Due To Loss-Of-Coolant Accident

Following the design basis accident pipe rupture within either compartment (reactor cavity, steam generator), high-enthalpy water flows out of both ends of the pipe, flashing partly to steam. As the pressure builds up within the compartment, the steam-air-water mixture will flow through openings in the compartment into the main containment. The pressure built up in the compartment is dependent on the number and shape of vent areas leading into the main containment, the volume of the compartment, and the blowdown rate from the broken pipe.

14.2.2.3.8.1 Pressure Buildup Within The Reactor Cavity

The cavity model, as analyzed, consists of a 5520 cubic foot compartment, that has a vent (always open) and concrete shield plugs which blowout during the accident. A value of six (6) square feet, representing the effective (always open) vent area of the cavity, was used. The total effective area of the vent
5. and the shield plugs (after shield plugs blow out) is 75 square feet. The effect of the increase in area with time, as the shield plugs blow, has been taken into account in the analysis by calculating the acceleration of the plugs due to the pressure buildup within the reactor cavity. The results of the analysis are presented in Figure 14-68. The indicated rapid pressure drop results from the shield plug blowout.

Largest Break Size Reactor Cavity Can Withstand

An analysis of the preliminary design of the reactor cavity was initially made in June 1967 to determine the largest break size that the cavity could withstand. The documentation established a break size of 8 square feet with the associated peak pressure of 195 psi across the cavity walls. Reanalysis of the final design determined that the cavity could withstand a somewhat higher pressure (205 psi) associated with a larger break size (8.5 square feet).

Largest Brea[illegible] ssible Within Reactor Cavity

Similarly, the ana[illegible]is of the largest break size possible that could occur within the react[illegible]vity was calculated to be 3.0 square feet which could produce a differ[illegible]l pressure across the cavity walls of 120 psi. This is accomplished by [illegible]aints on the primary lines which limit the physical separation of the en[illegible] the ruptured lines. The cavity, therefore, can safely withstand the pres[illegible]es resulting from the largest break possible in the coolant pipes within reactor cavity. A chart and tabulation of the reactor cavity pressures [illegible]ciated with various break sizes is shown on Figure 14-68.

14.2.2.3.8.2 [illegible]ssure Buildup Within The Steam Generator Compartments

There are two st[illegible]enerator compartments designated East and West. The idealized geomet[illegible]ed assumes two separate compartments. The East compartment has an "eff[illegible]e" volume of 60,400 cubic feet and an "effective" flow area of 1000 squ[illegible]eet. The West compartment has an "effective" volume of 61,700 cubic fee[illegible] an "effective" flow area of 1100 square feet. Only one blowdown was inv[illegible]ated for these calculations: the hot-leg rupture, since this case repre[illegible]the greatest energy release rate.

The pressure di[illegible]tial across the East and West generator compartment walls as a function [illegible] is presented in Figures 14-69 and 14-70. The values associated wit[illegible] differentials are 9.68 psi and 7.98 psi, respectively.

Maximum and De[illegible]ssures Within Steam Generator Compartments

The table bel[illegible] a summary of the maximum pressure and the design pressure in the st[illegible]rator compartments:

STEAM GENERATOR COMPARTMENTS

BREAK SIZE	[illegible]X. PRESS. East	West	DESIGN PRESS.	MAX./DESIGN PRESS. East	West
14.0 S. F.	[illegible].7 psi	8.0 psi	11.1 psi	87%	72%

14.2.2.3.8.3 Ass[illegible]ptions Were Incorporated In The Analysis

The steam and air throughout the compartment are in thermal equilibrium at all times.

During each time-step the blowdown mass expands isenthalpicly to the total compartment pressure. The water present at that time could form more steam only by (slow) evaporation. The water is assumed to undergo no further change of phase and drops to the floor of the compartment. The steam completes its isenthalpic expansion to the partial pressure of the steam already in the compartment and mixes homogeneously with the air.

During each time-step the mass of water added is temporarily stored. If the equilibrium calculation indicates atomsphere superheating, then a sufficient

$$K_{eff} = \Sigma_i K_i \qquad (3)$$

Orifice Coefficient Data

Available information on nominal orifice coefficients is summarized in Figure 14-73.

Head Loss Coefficient Data

The required experimental head loss coefficient data are represented in Figures 14-74 and 14-75.

Expansion Factor

The expansion factor used in the orifice flow relation is that applicable to a converging nozzle:

$$Y = \sqrt{\frac{1 - \left[\frac{A_2}{A_1}\right]^2}{\left[\frac{A_2}{A_1}\right]^2 \left[\frac{P_2}{P_1}\right]^{\frac{2}{k}}} \cdot \frac{\frac{k}{k-1} \left[\frac{P_2}{P_1}\right]^{\frac{2}{k}} \left[1 - \left[\frac{P_2}{P_1}\right]^{\frac{k-1}{k}}\right]}{1 - \frac{P_2}{P_1}}}$$

Where:

A_1 = upstream area

A_2 = orifice area

P_1 = upstream pressure

P_2 = downstream pressure

k = isentropic exponent

The isentropic exponent (k) for these calculations is based on the mixture of air and steam.

This calculation is conservative because for orifices which are not well rounded compressibility effects increase the expansion factor slightly.

quantity of this temporarily stored water is flashed into steam such that the atmosphere is just saturated. If there is not sufficient water in the temporary storage to saturate the atmosphere, then the atmosphere is allowed to superheat.

The flow out of the compartment into the main containment is calculated using the compressible flow equations for subsonic and choked flow. The orifice coefficient and expansion factors used are discussed in Section 14.2.2.3.8.4.

The flow is assumed to become sonic at the critical pressure ratio defined by:

$$\frac{P_2}{P_1} = \left[\frac{2}{k + 1}\right]^{\frac{k}{k - 1}}$$

Where:

P_2 = downstream pressure

P_1 = upstream pressure

k = isentropic exponent (see Section 14.2.2.3.8.4)

The volume of the compartment is measured to the minimum cross sectional area of each of the flow openings leading out of the compartment.

14.2.2.3.8.4 Calculation of Orifice Coefficient and Expansion Factor

The orifice coefficients (C) used in the orifice flow relation are sensitive to Reynolds number, orifice size, and orifice shape. Orifice coefficients typically become independent of Reynolds number at high Reynolds numbers. Reynolds numbers through orifices in the present area are greater than 10^6, so the Reynolds number effect does not apply here.

Sufficient experimental information on orifice coefficients for our geometry is not available. However, information is available on the head loss coefficient (K) defined by:

$$P = K \frac{V^2}{2} \qquad (1)$$

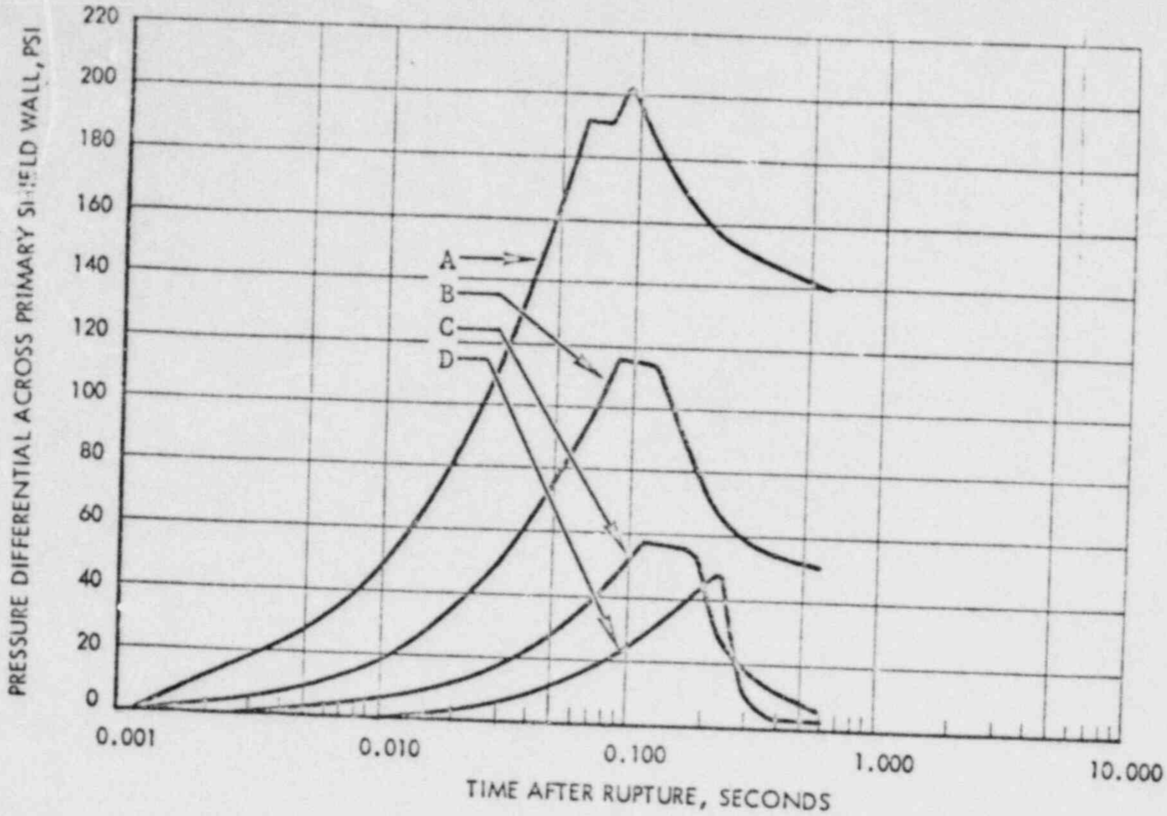
Where:

P = Pressure drop across orifice

V == Velocity through orifice

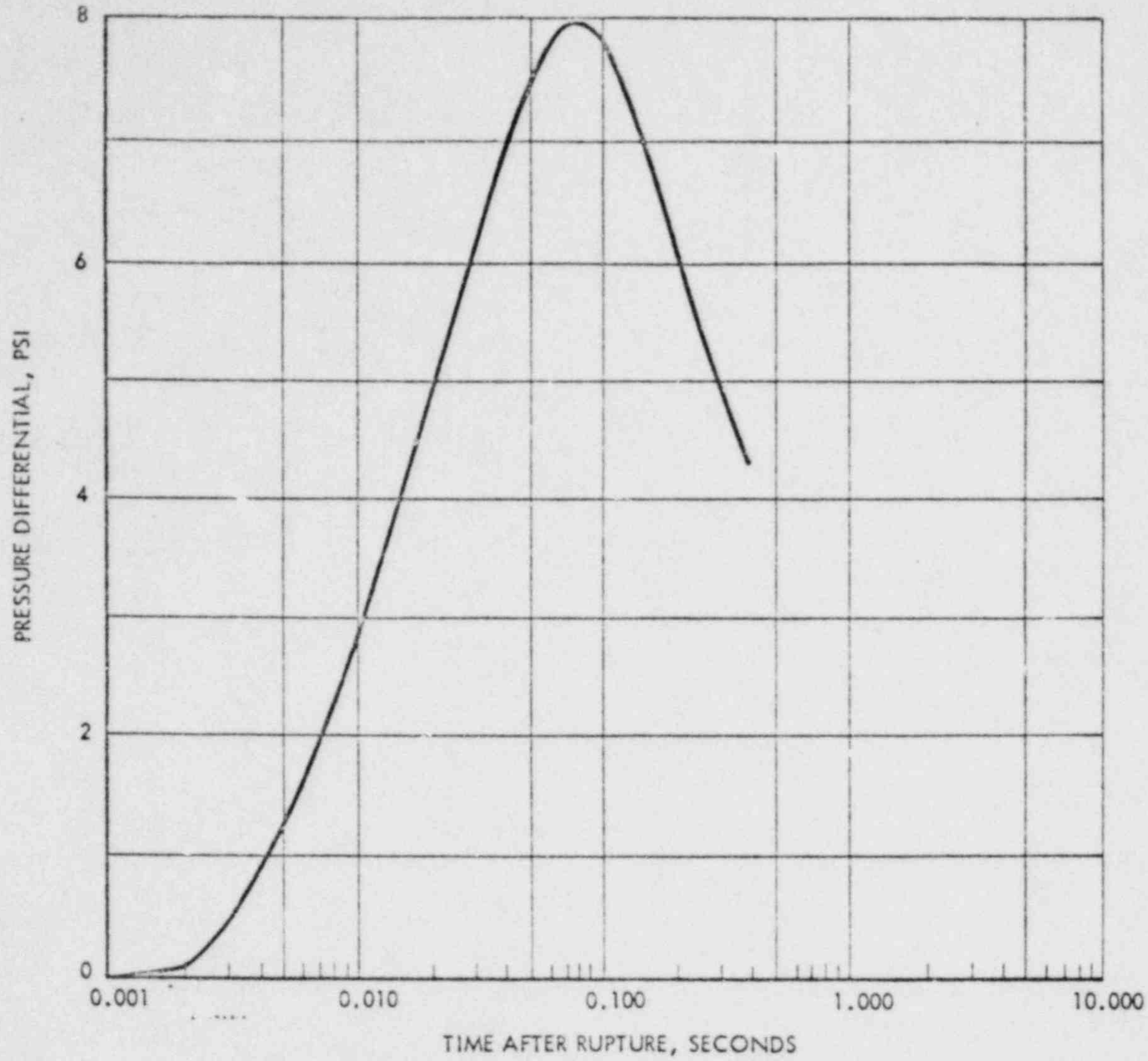
The relation between (C) and (K) is: $C = \frac{1}{\sqrt{K}}$ (2)

The advantage of the head loss coefficient is that the total head loss for a complicated flow system can be determined from an equivalent K obtained by adding K's for separate parts of the system (i.e. bends, expansions, contractions, etc.) as follows:



CURVE	BREAK SIZE*	DESCRIPTION
A	8.5 sq. ft.	Corresponds to maximum break size reactor cavity can withstand
B	3 sq. ft.	Corresponds to maximum hot leg break size possible within the reactor cavity
C	1 sq. ft.	1.0 square foot hot leg rupture
D	0.4 sq. ft.	0.4 square foot hot leg rupture
See Figures 14-71 and 14-72 for description of blowdown data used		

Figure 14-68 Pressure Transients for Range of Rupture Sizes Within Reactor Cavity.

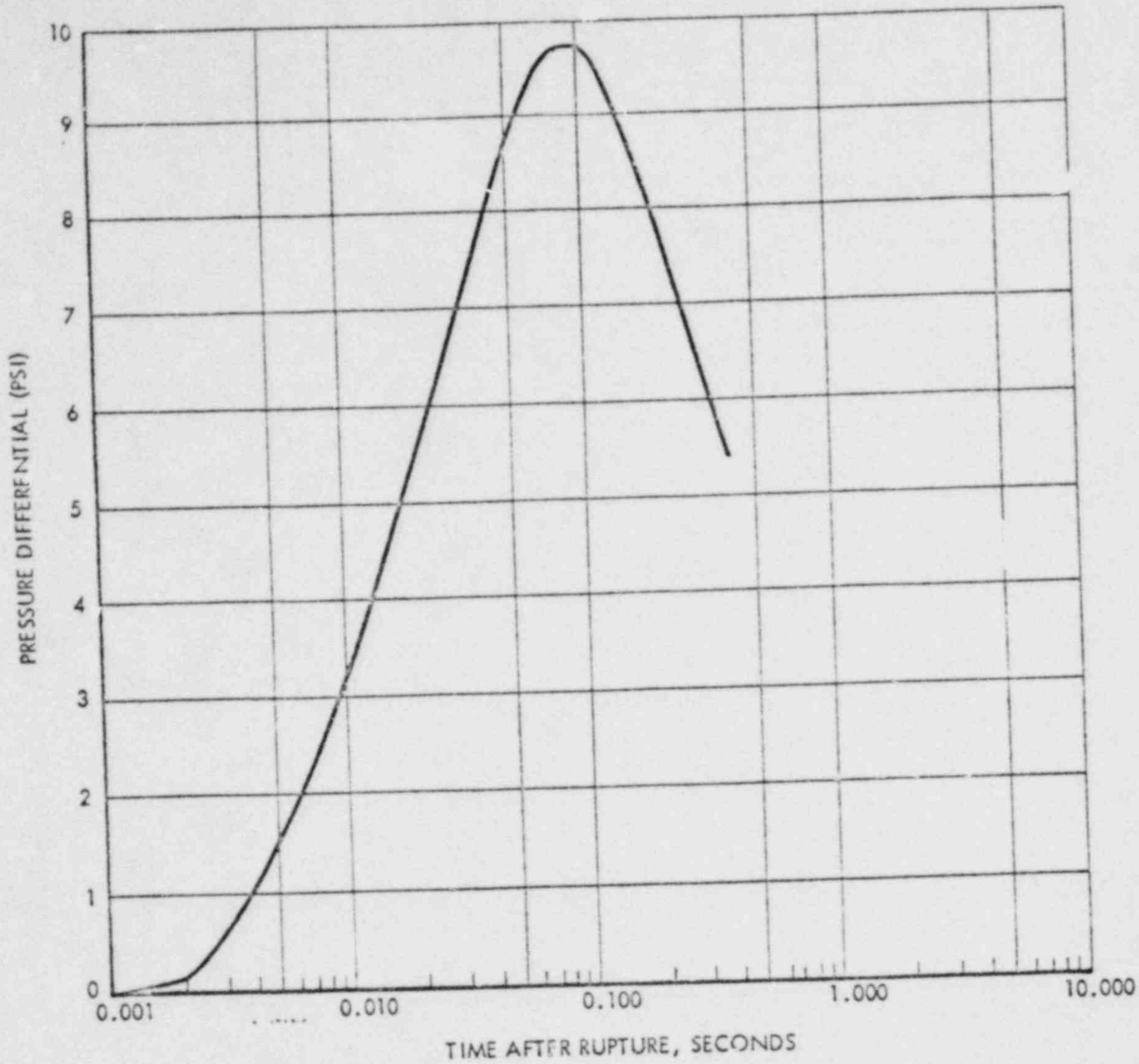


Pressure Differential Across West Compartment Walls Versus Time After Double-Ended Hot-Leg Rupture (Case 1-14 Square Feet).

OCONEE NUCLEAR STATION

Figure 14 - 69

(New) Rev. 5 [illegible]



Pressure Differential Across East Compartment Walls Versus Time After Double-Ended Hot-Leg Rupture (Case 1-14 Square Feet).

OCONEE NUCLEAR STATION

Figure 14 - 70

(New) Rev. 5 5/25/70

b. Section 5.1.2.5 requires a minimum of 0.15 percent bonded steel reinforcing (as stated above) for any location. At the base of the cylinder, the controlling design case requires 0.25 percent vertical reinforcing. As a result of pursuing the recommendation of the AEC Staff to further investigate current research on shear in concrete, several steps were taken:

1. The work of Dr. Alan H. Mattock was reviewed and he was retained as a consultant on the implementation of the current research being conducted under his direction. The criteria has been updated in accordance with his recommendations.

2. Concurrently with reviewing Dr. Mattock's work, the firm of T. Y. Lin, Kulka, Yang and Associates was consulted to review the detailed design of the cylinder to slab connection. It was their recommendation to use approximately 0.5 percent reinforcing rather than the 0.25 percent reinforcing indicated by the detailed design analysis for the vertical wall dowels. This increase would assure that there was sufficient flexural steel to place the section within the lower limits of Mattock's test data (approximately 0.3 percent) to prevent flexural cracking from adversely affecting the shear capability of the ection.

5.1.5 INTERIOR STRUCTURE

5.1.5.1 Design Bases

The Reactor Building interior structure (comprising all elements inside the Reactor Building shell) is a Seismic Class I structure and is designed on the following bases:

a. The stresses in any portion of the structure under the action of dead load, live load and design seismic load will be below the allowable
1. stresses given by either the ACI Building Code, ACI 318-1963 except as noted in 5.1.2.6, AISC Manual of Steel Construction, 6th Edition.

b. The stresses in any portion of the structure under the action of dead load, and thermal load will be below 133 percent of the allowable stresses given in (a).

c. The capability to safely shut down the plant will be maintained under the combined action of dead load, maximum seismic load, pressure and jet impingement load. The latter two loads are based on the rupture of one pipe in the primary loop. The deflections of structures and supports under these combined loads would be such that the functioning of engineered safeguards equipment would not be impaired. The yield load equations in Appendix 5A are adhered to except that local yielding is permitted for pipe, jet or missile barriers provided there is no general failure.

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5.1.5.2 Design Loads and Materials

The Reactor Building interior structure consists of (1) the reactor cavity, (2) two steam generator compartments, and (3) a refueling pool which is located between the steam generator compartments and above the reactor cavity.

The reactor cavity houses the reactor vessel and serves as a biological shield wall. The reactor cavity is also designed to contain core flooding water up to the level of the reactor nozzles.

The primary functions of the steam generator compartment walls are to serve as secondary shield walls and to resist the pressure and jet loads described below.

The foundations for all NSSS equipment including the reactor vessel, the steam generators, and the pressurizer are designed to remain within the elastic range during rupture of any pipe combined with the "maximum earthquake."

The design pressure differential across walls and slabs of enclosed compartments in the internal structure are as follows:

Reactor Cavity	- 208 psi
East Steam Generator Compartment	- 11.1 psi
West Steam Generator Compartment	- 11.1 psi

In addition to the peak pressure differentials, the steam generator compartment walls are designed for simultaneous action of a single jet impingement load and the safe shutdown earthquake. Design of structures was done using conventional structural analytical techniques.

Pipe whipping restraints are provided for the main steam, feedwater and other high-pressure piping in accordance with criteria in Section 5.4.

The materials used for the above structural elements are as follows:

Structural Steel	-ASTM A36
Concrete	$-f'_c$ = 4000 psi at 28 days.
	$-f'_c$ = 5000 psi at 28 days (for steam generator bases, reactor foundation, and primary shield wall).
Reinforcing Bars	-ASTM A615, Grade 40 for Bars #11 and under ASTM A615, Grade 60 for Bars larger than #11.

5.1.5.3 Missile Protection

High-pressure reactor coolant system equipment which could be the source of missiles is suitably screened by the concrete shield wall enclosing the reactor coolant loops and by special missile shields to block any passage of missiles to the Reactor Building walls. Potential missile sources are oriented so that the

REQUEST 12

Discuss the analysis which shows that primary pipe whip will not cause failure of the secondary system.

RESPONSE:

A detailed study of the primary loop was performed to determine potential pipe break locations which could possibly cause either fluid impingement or pipe impact forces on the secondary system. The results of this evaluation indicated the most credible break locations which could cause either of these effects are:

1. a guillotine break at the pump discharge in the cold leg piping;

2. a longitudinal split in the vertical pump suction segment of the cold leg piping; or,

3. a longitudinal split in the vertical segment of the hot leg piping.

All of the above breaks could potentially affect the generator because of their prosimity to it. The main steam lines, however, are shielded from the effects of pipe breaks by the generator.

The primary piping and steam generator were analyzed for each of the above breaks and supports provided to restrain the pipe from whipping into the generator. In addition, the stresses in the generator shell due to the fluid impingement forces were calculated and found to be within acceptable limits.

The restraints on the primary loop are shown in Figures 12-1 and 12-2. The coolant pump is restrained by steel supports from the primary shield wall. The hot leg piping is restrained by the concrete support at the primary cavity penetration, an intermediate steel support from the primary wall, and another steel support near the generator upper tube sheet. The vertical segment of the cold leg piping is restrained by a steel support midway along its length, which would spread any rupture load over a larger area of the generator shell.

To verify the location and size of the piping supports, the piping was analyzed for rupture loads occurring at the worst point along its length. The rupture thrust force was assumed equal to Px A, where P is the coolant pressure and A the flow cross-sectional area of the pipe. The thrust was applied as an equivalent static force using a dynamic load factor of 2.0. Assuming the force to be a point load acting at the midpoint of the span between supports, the piping stresses were calculated using beam models. The supports are located so as to prevent the formation of plastic hinges in the piping, which would lead to an unstable linkage-type structure and possible impacting against the generator.

To evaluate the effect of fluid jet impingement on the generator, an equivalent static pressure load on the shell was calculated. A break of 14 ft^2 for the hot leg or 8.5 ft^2 for the cold leg was assumed. The maximum initial mass velocity was computed using the methods outlined in the report "Maximum Two-Phase Vessel Blowdown From Pipes, APED-4827," by F. J. Moody. It was assumed that the fluid leaves the break in a direction normal to the pipe and that its velocity undergoes a 90° change in direction upon impinging on the OTSG. The resulting shell pressure loading was calculated to be 1300 psi.

A shell analysis was performed on the OTSG to determine the stress intensity due to the above loading. A B&W proprietary digital computer code, which considers two-dimensional shells with asymmetric loading, was utilized. The loading distribution and stress model are shown in Figures 12-3 and 12-4.

The maximum stress intensity was computed [illegible]e 38,6[illegible] psi. This is less than the allowable stress of 46,670 psi. Based on these results for the 36" ID pipe break, it was concluded that the OTSG shell could also withstand the reduced loading which would be generated by a 28" ID break.