Appendix M

REPORT ON PIPE FAILURES OUTSIDE CONTAINMENT IN THE BROWNS FERRY NUCLEAR PLANT

M.1 INTRODUCTION

The Browns Ferry Nuclear Plant was reanalyzed for the consequences of postulated pipe failures in the main steam and feedwater lines located outside the containment structure. This reevaluation was performed in accordance with the information guide forwarded by AEC to TVA in a letter dated December 18, 1972. The objective of the reanalyses was to show that the plant had been designed so that the reactors could be shut down and maintained in a safe shutdown condition following a postulated rupture in the main steam and feedwater systems. The plant structures, systems, and components important to safety were examined to assure that they could accommodate such a postulated rupture without compromising the ability of the plant to be put into a safe shutdown condition assuming a concurrent and unrelated single active failure of protected equipment.

This appendix provides the documentation to support TVA's position that Browns Ferry complies with Criterion No. 4 of the AEC's General Design Criteria, listed in Appendix A of 10 CFR 50, as far as the main steam and feedwater systems are concerned. The information and analyses contained in Appendix M have been further supplemented by two separate reports. The title of each of the reports and the date of its submittal to the Atomic Energy Commission are given below.

"Concluding Report on the Effects of Postulated Pipe Failure Outside of Containment for Unit 1 of the Browns Ferry Nuclear Plant," DED-TM-PF1, November 2, 1973.

"Concluding Report on the Effects of Postulated Pipe Failure Outside of Containment for Units 2 and 3 of the Browns Ferry Nuclear Plant," DED-TM-PF2, March 29, 1974.

In addition to these supplemental reports, Reference 4 documents the Browns Ferry Unit 1 Cycle 7, Unit 2 cycle 5, and Unit 3 cycle 6 restart evaluations conducted to assure compliance with the original pipe rupture licensing commitments and to assure that the original pipe rupture design basis for the plant has not been invalidated by design changes subsequent to the original evaluation. Reference 7 contains the provisions of Generic Letter 87-11 (Reference 8) for the selection of required pipe break sizes and locations.

In conjunction with the 5% uprate in core thermal power, additional analyses were performed to evaluate the effects of high energy line breaks outside containment during operation at a core thermal power of 3458 MWt with a nominal steam dome

pressure of 1050 psia. Plant structures, systems, and components were evaluated to assure that the plant could be shutdown and maintained in a safe shutdown condition following the postulated pipe failures.

Furthermore, documentation has been previously supplied that describes how flooding from ruptures in low-pressure fluid systems is accommodated. This information provides additional confirmation and substantiation that the Browns Ferry Nuclear Plant is in compliance with Criterion No. 4 of Appendix A, 10 CFR 50, as well as Criteria Nos. 40 and 42 of the previous (1967) issue of AEC's General Design Criteria, which reflected similar requirements as Criterion No. 4. Refer to Chapter 10.16, Section 10.16.4.6., for a summarized discussion of the station's internal flooding event and coping strategy.

The steam vault, shown in Figure M.0-1, contains all major steam and feedwater piping components running between the drywell penetrations and the Turbine Building. This part of the vault in the Reactor Building is maintained at a slightly negative pressure during normal reactor operation via the duct connection to the normal ventilation system. The blowout panel, which seals the vault in the Reactor Building from the Turbine Building, is designed to be blown out when a pressure of 90 psf exists. (This feature assures the vault in the Reactor Building does not become uniformly pressurized in excess of the design value of 10 psi.)

M.2   PRESSURE ANALYSES

Most of any major blowdown into the steam vault (inside the Reactor Building or Turbine Building) is discharged via the Turbine Building to the atmosphere, whereas a small portion will be discharged to the Reactor Building via the ventilation duct and labyrinth passageway. The analyses account for the parallel paths and evaluate the pressure transient and consequential effects of the blowdown fluid for both paths.

The main steam valve vault pressure analysis was originally performed in 1967 and updated as described below at the time of initial startup. The analysis was updated again in 1988 prior to unit restart to include increased blowout panel and hollow metal door blowout pressures. The analysis was conducted using the MONSTER computer program. The discussion below has been supplemented to reflect the differences between the MONSTER analysis and the previous analysis. The analyses are consistent except where noted.

Break locations were considered for the piping arrangement shown in Figure M.0-1. Both circumferential and longitudinal breaks were considered; however, only the worst-case pressure transient was analyzed. That is, the double-ended break of a main steam line near the outboard isolation valve on the drywell penetration results in the maximum pressure the structural enclosure will experience. This maximum pressure included blowdown from a consequential break of the 4-inch RCIC steam line located in the steam vault as part of the worst-case analysis. The updated

MONSTER analysis does not include the effects of the four-inch RCIC steam line rupture since, in accordance with References 4 and 7, the rupture of the 26-inch main steam line does not result in the consequential RCIC break. The efflux passing through the steam vault was assumed to travel over the anchor wall and on through the vault in the Turbine Building, where it must make a right angle upward, turn, and pass into the room housing the stop valves, bypass valves, and control valves. From there it exits through some additional blowout panels and the opening at the front of the turbine pedestal into the Turbine Building. Then it passes from the Turbine Building to the atmosphere.

The efflux, passing through the ventilation duct and labyrinth passageway, discharges into the Reactor Building. The Reactor Building was treated as a large enclosed volume without leakage or condensation of the efflux in determining the consequential effects of flow into the Reactor Building.

The blowdown flow rates from the PSAR (D and A report) and FSAR were compared at the time of initial startup to assure that we had a common starting point. The original design calculations were made in January 1967, using flow rates from the PSAR to establish the 10-psi design value for the steam vault pressure loading. The idealized flow in Figure M.0-2 was used in both the initial startup evaluation and the MONSTER evaluation for a 5-second closure of the isolation valves and a time of 0.5 second to sense and initiate closure (total of 5.5 seconds to close valve). Chapter 14 of the FSAR gives the idealized flow for a 10-second isolation valve closure and 0.5 second to sense and initiate closure. The total mass losses for these times and the corresponding total mass losses given in the PSAR were as follows:

Time	FSAR	PSAR
5.5 seconds	61,200 pounds	58,000 pounds
10.5 seconds	185,000 pounds	185,000 pounds

The flow, averaged over the 2-phase region of Figure M.0-2, was approximately 14,400 lb/sec which matches the value quoted in the PSAR. Thus, we conclude that we had a common basis for our analysis of the maximum pressure transient in the steam vault.

The maximum pressure transient was calculated at initial startup for the 2-phase blowdown period shown in Figure M.0-2 with the following set of assumptions:

a.   A steady state flow model,
b.   An average flow over the 2-phase region of 14,400 lb/sec,
c.   A RCIC steam flow of 100 lb/sec, and
d.   A moisture entrainment of 50 percent.

The MONSTER analysis was performed with the following assumptions:

a. A transient steam/water/air flow model based on UFSAR Figure M.0-2,
b. Valve vault blowout panel differential pressure of 90 psf,
c. Valve vault door blowout differential pressure of 1.5 psi.

The calculations were made with the anchor wall, the blowout panel area, the ventilation duct entrance, and the passageway entrance as square-edged orifices. The right angle turn was treated as a 90-degree miter bend, and the flow into the room that houses the valves was treated as a sudden expansion. The results of applying this model gave the maximum pressures along the steam vault flow path as follows.

Location	Pressure
Reactor Building Vault	8.1 psig
Turbine Building Vault	2.6 psig
Turbine Valve Room	0.7 psig

The calculated maximum pressure was 4.9 psig (compared to 8.1 psig) in the 1967 evaluation. However, the original calculation did not include the anchor wall (added later) or moisture entrainment. The 1967 design value for pressure loading in the steam vault was taken as 10 psi to provide margin to take care of model improvements as well as physical layout changes. Some of the margin was used up in the initial startup calculation, but the design value of 10 psig for the steam vault pressure loading is still adequate. The MONSTER pressure transient analysis was performed which resulted in a maximum pressure of 9.98 psig. Thus, the original design basis of 10 psi for the pressure loading in the steam vault was a sound decision and will continue to be used as the design basis.

In conjunction with the 5% power uprate, additional analyses were performed to evaluate the effects of a double-ended main steam line break in the steam vault during operation at a core thermal power of 3458 MWt with a nominal reactor steam dome pressure of 1050 psia. The mass and energy release data was calculated using the LAMB computer code. The pressure response for the main steam valve vault was calculated using the GOTHIC computer code. The maximum steam vault pressure calculated using the GOTHIC computer code was less than 8 psig.

The initial startup calculational model predicted that a flow of 1000 lb/sec was discharged into the Reactor Building during the 2-phase blowdown period and 620 lb/sec during the steam blowdown period. The total mass released in the reactor zone was less than 4600 pounds, and it was assumed to mix only with the building air in the reactor zone of the ventilation system. By assuming no leakage or condensation on the walls, the maximum pressure rise will be less than 1.4 psi. This pressure rise was analogous to the pressure changes in the Reactor Building during

a tornado. The effect of tornadoes on the Reactor Building is discussed in paragraph 12.2.2.3.1 of the FSAR. The transient pressure calculation for a tornado reached a maximum of 1.44 psi, which is higher than the upper limit (1.4 psig) given above. The design value used for the Reactor Building was 1.75 psig. Thus, by analogy, the Reactor Building was designed to accept this steam release into the Reactor Building by virtue of the tornado design considerations.

Pressure transients in the steam vault were less severe for other break locations than that discussed above. Also, smaller breaks resulted in lower pressures. Since neither of these pressure transients was controlling, no further analyses were necessary.

A negative pressure transient could occur at the end of the blowdown, since the steam vault contained essentially a steam atmosphere that would begin condensing. Only a small negative pressure transient would be anticipated from condensation on the vault surfaces and water on the floor. Air would be drawn from the Reactor Building and the Turbine Building. Outside air would be drawn into the Reactor Building through the ventilation system and any relief panels that had been opened.

The Turbine Building volume, ventilation system, and inleakage were sufficient to prevent large negative pressures in the Turbine Building. (The Turbine Building was not designed as a low-leakage building; that is, it has huge unsealed doors as well as a large number of smaller unsealed doors.)

The magnitude of the slightly negative pressure in the steam vault was immaterial, since the structural enclosure had been designed with symmetrical reinforcing on each face of the walls and slabs. It was assumed that the vault could take a negative pressure loading that approaches the internal design pressure loading of 10 psi. Also, a slightly negative pressure would not be a problem for the Reactor Building, because it had been designed for an equivalent external pressure loading of 3.3 psi. However, the Turbine Building was not designed for negative pressures; but the volume and inleakage are such that large negative pressures cannot be developed. Thus, any negative pressure transient would be small and, therefore, of little consequence to the design of the Browns Ferry Nuclear Plant.

M.3 PIPING DESIGN PHILOSOPHY

M.3.1 Piping From Reactor Vessel Through the Anchor Outside the Drywell

Steam-feedwater piping components from the reactor vessel nozzles, through the isolation valves outside the drywell, are critical from the pressure-integrity point of view, since they serve as part of the reactor coolant pressure boundary. Therefore, these components are designed, fabricated, installed, tested, and inspected to quality requirements that are consistent with their importance to safety. These considerations and requirements were extended to piping beyond the isolation valves through the anchor. A summary of the considerations given to the various phases of designing, manufacturing, and installation of the steam-feedwater piping components is provided below.

Analysis

a. A detailed stress analysis was made by the General Electric MASS computer program of all significant temperature conditions. (Refer to Appendix C for analysis methods used subsequent to the original design and construction.)

b. The forces, moments, and stresses due to earthquake were calculated from a detailed dynamic analysis of the system.

c. The stress conditions for all credible emergency and faulted conditions were analyzed and compared with established limits.

d. All critical points through the system were evaluated completely to the stress limits of USAS B31.1.0, Power Piping Code.

Materials and Fabrication

a. All butt welds used in fabricating pipe and fittings were completely examined by radiographic and liquid penetrant methods.

b. The weld processes used were limited to those which provide the highest quality level achievable by the industry.

c. All materials were per ASTM specification, and the supplemental requirements on tension and bending tests were invoked.

d. Quality control programs and quality control records were established to ensure that all materials and fabrication were in accordance with specifications.

Erection and Tests

a. All field welds were completely examined by radiographic and liquid penetrant methods.

b. Erection procedures were carefully worked out in advance, and the installation of the system was performed in a manner to minimize erection stresses.

c. The weld processes used were limited to those which provide the highest quality level achievable by the industry.

M.3.2   Piping from the Anchor Through Turbine Inset or Feedwater Pump Discharge

Piping components running between the anchor outside the drywell and the turbine stop valves (steam lines) or feedwater pump discharge (feedwater lines) have been fabricated, erected, inspected, and tested to high quality-assurance requirements. Deadweight supports have been located to within limits established in USAS B3l.1.0, 1967 edition which assures the deadweight loading does not produce stresses in excess of 1500 psi. The wall thickness, including an additional thickness for corrosion allowance of 0.120 inch, assures the operating pressure stresses will not exceed the code allowable during the 40-year life of the plant. For the 60 year operating life, the aging effects will be managed using the Chemistry Control Program, Flow Accelerated Corrosion and One-Time Inspection Program described in Appendix O, Sections O.1.10, O.1.14, and O.1.26. Thermal flexibility has been provided for in the design, and stresses have been limited to the extent that the sum of the stresses resulting from the combined loading of deadweight, pressure, and thermal does not exceed the code allowable (USAS B31.1.0, 1967 edition). Piping components in this area were not seismically designed.

M.4   BREAKS POSTULATED AND LOADING EFFECTS CONSIDERED

Circumferential and longitudinal breaks were postulated to occur in accordance with guidelines provided in the general information request. Breaks were postulated in piping components in the two main regions of the steam vault as follows.

a. Between drywell penetration and the anchor.

    (1) Circumferential and longitudinal at the anchor, and

    (2) Critical crack size postulated to occur at locations considered to have potential for producing adverse effects on the surroundings.

b. From the anchor through the Turbine Building region of the steam vault to the steam chest header:

(1)   Circumferential and longitudinal breaks at the anchor and at large changes in flexibility, such as at elbows and tees.

Loading effects considered at all break locations for the circumferential and longitudinal breaks are as follows:

a.   Jet thrust acting at the point of break,

b.   Jet impingement loading on surrounding structural walls and components, and

c. Pipe whip impact loading when applicable.

M.5 PIPE RUPTURE LOADS

Theoretical techniques for evaluating dynamic effects such as thrust or blowdown loads, jet impingement loads, and jet expansion considerations are developed in References 1 through 3. It is considered beyond the scope of this report to redevelop those methods here. Where the methods had a direct application, they became the tools for evaluating the effects considered and have been referenced as such.

NOTE: See Section M.5.3.3 for current methodology regarding evaluation of jet impingement loads for source pipes containing steam or flashing subcooled liquids.

Piping components were assumed to sever instantaneously. At the instant the pipe ruptures, the force (F_i) acting perpendicular to the plane of break is given as:

$$F_i = P_o A_B$$

where P_o is the operating pressure in the line and A_B is the break area. This load is considered to be altered immediately after rupture occurs since, during the highly transient situation, the opening to atmospheric conditions causes a decompression wave to form (References 1 and 2). The wave moves along the axis of the pipe and away from the break, leaving at the break opening. However, the thrust load (F_T) that develops is given by the impulse function (Reference 3) as:

$$F_T = PA_B + \dot{M} \overline{V}$$

where P, $\dot{M}$, and $\overline{V}$ are the pressure, mass flow rate, and velocity of the fluid at the point of break.

Many simplifying assumptions must be made to evaluate the dynamic effects at the break. For example, the geometrical character of a split for a longitudinal break is

very uncertain. Therefore, a model such as a short nozzle type was selected rather than a sharp-edged orifice which would reduce the computed reaction thrust. Steady thrust, considering friction effects in piping ducts, is developed in References 1 and 2 for reactor vessel conditions similar to the Browns Ferry Nuclear Plant (see Figure 2, Reference 1). For piping systems that have friction parameters (fL/D) of less than approximately 7, the opportunity for error in determining the steady thrust load is considerably greater than for larger friction parameters. Since the steam-feedwater piping is compact in design, the friction parameter (fL/D) was within the range for which considerable uncertainty exists. Therefore, pressure losses due to friction were conservatively ignored in evaluating thrust loading effects at the break points assumed.

Flow limiting devices were considered in evaluating steady thrust effects at break locations in the steam piping components. Procedures for evaluating the thrust loads for isentropic flow of a compressible fluid are well established and have been experimentally verified (Reference 3). Jet thrust loading at the break was determined by considering steam as a perfect gas with a polytropic gas constant of 1.28 expanding through an ideal nozzle.

For circumferential breaks in the steam piping components, a double-ended break results since flow from each leg will result. On the reactor side of the break, the steady thrust load is directly obtainable knowing the break area (A_B) to throat (upstream restrictor) area (A_T) ratio. The remaining three lines were conservatively assumed to blow down through the broken leg opposite the reactor side of the break. The sum of the flow restrictor areas for the three lines was set equal to the throat area (A_T) in defining the area ratio (A_B/A_T) to establish the blowdown loading assumed to act on this piping leg.

For longitudinal breaks, blowdown flow may come from either the upstream or downstream, or both, directions at a break. For sufficiently small breaks, flow rate is limited by the break itself. If there are restrictions either upstream or downstream of the break, or both, the sum of the restriction areas is set equal to A_T. Note that maximum thrust is obtained when $A_B/A_T = 1.0$. For area ratios less than 1.0, a ratio of 1.0 was used to establish the thrust load.

M.5.1 Critical Crack Size Loading

The critical crack size assumed to occur in piping components was taken to be one-half of the pipe diameter in length and one-half of the wall thickness in width. The thrust is defined at the break by the impulse function (Reference 3) as:

$$F = PA_c + \dot{M} \overline{V}.$$

For a compressible fluid such as steam, and assuming choked flow at the cracked opening, thrust will be maximized. Therefore, the thrust load at the point of break may be expressed as:

$$F = PA_c \ (1 + kM^2)$$

where:

P = Pressure at the break opening

A_c = Crack area

k = 1.28

M = Mach number = 1.

Therefore, the thrust loading assumed to act at the break point for a steam crack is:

$$F_g = 2.28PA_c$$

The maximum operating pressure was used for evaluating thrust loads for the critical crack size.

For incompressible fluids such as feedwater, the thrust loading assumed at the point of break is given as:

$$F_L = 2 \ PA_c$$

where P is the operating pressure.

M.5.2   <u>Jet Expansion Considerations</u>

The jet was assumed to expand symmetrically at an angle of 20 degrees and was considered to be fully expanded or asymptotic at a distance equal to five pipe diameters. These assumptions are conservatively based upon the jet expansion considerations developed in References 1 and 2. An example of an application of the assumptions for jet expansion is shown in Figure M.0-3.

M.5.3   Jet Impingement Loads

M.5.3.1   Flow Around Circular Targets

The reaction loading on a circular target is given as:

$$F_j = C_D P_j A_i$$

where C_D is the drag coefficient, P_j is the pressure of the jet, and A_i is the area of the target exposed to the jet. The jet pressure (P_j) used in the evaluation is obtained from the following relationship:

$$P_j = \frac{F_t}{A_j}$$

where F_T is the thrust force and A_j is the area of the expanded jet at the location of concern. For breaks in steam-feedwater components, the local Reynolds numbers along an expanding jet range from 3×10^7 to approximately 9×10^7 assuming the velocity of jet remains constant. Published experimentally determined drag coefficients for Reynolds numbers in this range for cylinders in crossflow range from 0.30 to 0.36. Therefore, 0.36 was used for impingement loading evaluation on circular objects.

M.5.3.2 Jet Impingement on a Flat or Concave Surface

The jet impingement loading assumed to act on a flat or concave surface was assumed to be equal to the thrust load acting at the point of break. The area over which the load was assumed to act was determined using the jet expansion considerations discussed above.

M.5.3.3 Jet Impingement Loading -- Steam/Flashing Liquid Sources

NUREG/CR-2913, "Two-Phase Jet Loads," (Reference 6), evaluated jet pressures for axisymmetric target geometries for loadings associated with jet sources of steam/flashing subcooled liquids. Based on findings in this report, it has generally been concluded that jet impingement loading from steam or flashing subcooled liquid sources is not significant for targets whose separation distance from the break source is greater than ten times the inside diameter of the source pipe. Therefore, for evaluating pipe rupture sources containing steam or flashing subcooled liquids, the following conclusion shall be utilized:

When the jet consists of steam or flashing subcooled liquid, unprotected equipment/components located at a distance greater than ten diameters[*] from the break shall be assumed to be undamaged without further analysis provided that the environmental qualification of the target is not exceeded.

M.5.4   Break Size

Break size for circumferential breaks was taken to be the flow area for the pipe. Longitudinal breaks were assumed to have a flow area equal to that of the pipe and a split length equal to twice the pipe diameter. The longitudinal break orientation was assumed to be aligned with the longitudinal axis of the pipe and may occur at any location around the circumference of the pipe.

The critical crack size, defined per the AEC general information guide for considering breaks, was assumed to have any orientation.

M.6   PIPE BREAK ASSUMPTIONS, ANALYSIS, AND BREAK LOCATIONS

General assumptions that were applicable to the evaluation of dynamic effects of a postulated rupture have previously been summarized in Chapter 14 of the FSAR. Additional assumptions that were applied are as follows.

a.   Blowdown loads acting perpendicular to the plane of break were assumed to cause unrestrained motion of the piping component about the nearest restraint point, provided a plastic hinge moment was capable of developing.

b.   Piping components impacted by another piping component were assumed to remain functional if they were of equal or greater schedule and of equal or greater diameter. The reverse condition was assumed to cause a loss-of-function of the impacted component.

M.6.1   Pipe Rupture Analysis

Figures M.0-4 and M.0-5 show locations for which design basis pipe breaks were considered in steam and feedwater-piping components. The types of breaks (circumferential or longitudinal) and the analyses made are discussed individually on a case basis. Critical cracks were assumed to occur at random locations between the drywell penetration and the anchor for both systems. Their location (not shown in the figures) was postulated to occur where the most adverse reactions could

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[*] For this application, "diameter" is defined to be the inside diameter of the broken pipe for a circumferential/longitudinal rupture and the equivalent circular diameter for a critical crack.

result. For all high energy piping greater than 1 inch nominal diameter, critical cracks were assumed to occur everywhere except in high energy piping that was seismically analyzed in accordance with USAS B31.1 (1967) where the primary plus secondary stress intensity is below $0.4(S_h+S_A)$ as defined in Section 6.3.2.3 of Reference 4.

M.6.2 Case I - Circumferential Break Postulated at Point 1

A break postulated in a steam line constitutes a double-ended break. Jet thrust loads were assumed to act along the axis of each broken element. The reactor blows down directly through the leg on the reactor side of the break and by a reverse of the flow from the steam chest from the three remaining steam lines.

Jet thrust reaction on the reactor side of the break for an A_B/A_T ratio of 2.78 is 0.54 $P_o A_B$. This loading is transmitted to the anchor via tie rods (see Figure M.0-1) connecting the flued head at the drywell penetration with the anchor. The tie rods may permit a very slight opening to form in the event of a circumferential break. Steam effluent from this type of break is assumed to jet radially outward from the crack opening separating the legs of the broken component. The resulting impingement loading will be no greater per unit length than will be shown for the critical crack size, which for the steam lines is 15.0-kips per foot. Since a 15.0-kip load offers no serious loading effects in general, circumferential breaks at point 1 in the steam line are not expected to jeopardize the ability to shut down and maintain the plant in a safe condition.

For the feedwater components, the break postulated at point 1 constitutes a single-ended break. A break in the feedwater line is not considered to be as serious as the steam line, since the maximum load cannot exceed $P_o A_B$ and rapidly decays immediately after the break occurs.

M.6.3 Case II - Circumferential Break Postulated at Points 2 and 3

The circumferential break can essentially be postulated at any point along the horizontal axis between the anchor at one end and the elbow at the other. The net effect is essentially the same for either location. Axial thrust loads for the double-ended break, identical to those in case I, will be imparted to each broken element. The anchor is designed to resist the loading associated with the leg connected to it. A plastic hinge would be expected to occur at the steam chest-pipe connection at point 5, and the leg would be expected to move away from the anchor until the north wall of the steam vault in the Turbine Building is struck. While an analysis for a more severe break considered in case VI, below, will show that the wall can be expected to contain the whipping pipe, consequential failure of the wall would not cause a failure of the Turbine Building structure that could jeopardize the reactor or Control Building structure or necessary equipment needed to mitigate the effects of the postulated break.

In general, the comments above are equally applicable to the feedwater piping. Since a single-ended break results, the net loading effect will be lower than that assumed for the steam components.

M.6.4   Case III - Circumferential Break Postulated at Points 4 and 5

A circumferential break assumed to occur either at the nozzle joining the pipe to the steam chest at point 5 or the pipe to the elbow at point 4 results in a similar situation. The effect of the jet thrust load will be to cause a plastic hinge to occur at the anchor (point 2). The pipe will strike the floor of the Turbine Building. Since the floor is founded on soil and fill, no damage to the Turbine Building structure is expected that would potentially jeopardize the reactor or Control Buildings or equipment necessary to effect a safe shutdown of the reactor.

M.6.5   Case IV - Longitudinal Break Postulated at Point 1

A longitudinal break on the valve side of the anchor provides a more serious interaction with the surroundings than the circumferential break. The rupture opening was assumed to be twice the pipe diameter in length (48 inches) for both the steam and feedwater piping components and a flow area equal to the pipe flow area.

Jet thrust loads resulting from breaks in the steam lines were based on flow from both directions to the break with an area ratio $A_B/A_T = 1.0$. The loading that results is:

$$F_T = 1.26\ P_oA_B$$

where P_o is the maximum operating pressure (1000 psi) for the design power rating and A_B is the break flow area. Steam-feedwater piping in this area is 24.0-inch OD, Schedule 80, ASTM A 106, Grade B piping. The steam line thrust load resulting from this evaluation is 460 kips.

The feedwater-line-break thrust load was defined as P_oA_B, where P_o is the maximum operating pressure (1250 psi). The feedwater lines do not have upstream restrictions, and the pressure is assumed to decay rapidly to the saturation pressure for the temperature of the water. The maximum thrust at the instant the break occurs is 456.5 kips. Since both the steam and feedwater lines are located in close proximity throughout the steam vault, the steam line break thrust was considered as the limiting load for evaluating dynamic effects.

The RCIC piping connecting the discharge pump to feedwater pipe B outside the drywell is routed between steam lines C and D. A longitudinal break was postulated to occur at the location around the circumference of either steam lines C or D, and

the resulting impingement loads were calculated using the model shown in Figure M.0-6.

For a break in steam line D oriented such that blowdown impinges on the RCIC line, the area of the jet, A_j, at the plane where impingement occurs is 1635 square inches while the intercepted area of the RCIC line is 260 square inches. The resulting impingement load is:

$$F_1 = 0.36 \ F_T \ \frac{A_i}{A_j} = 26.4 \text{ kips}$$

Torsional and bending stresses at the nozzle of the tee connecting the RCIC water line to the feedwater line for the loading exceeded the code allowable stresses for the material. Consequently, the RCIC line will be postulated to fail in the event of a longitudinal break of either steam line B, C, or D.

A direct impingement is not possible from steam line A because of the shielding effects from steam lines B and C. The reduced pressure effects in the jet profile due to greater expansion in the distance traveled from line A will produce less severe loading on the RCIC line. Therefore, the RCIC line will not be damaged to the extent that its capability to function will be jeopardized from breaks in steam line A.

A similar analysis was performed for the RCIC steam supply line that is located between steam lines B and C in this area of the steam vault. The 4-inch steam supply line will not resist the loading from the jet impingement; therefore, it is assumed that it could fail as a consequential effect of longitudinal breaks in either steam line A, B, C, or D in this location.

Jet impingement loading evaluations on adjacent main steam and feedwater piping in the steam vault resulted in pipe stresses within the code allowable stress. These results are consistent with the current industry approach that piping will not be damaged from pipe rupture loads originating in adjacent piping of equal nominal size and schedule.

M.6.6 Case V - Longitudinal Breaks Postulated at and Between Points 2 and 3

Longitudinal breaks postulated in piping components in this region of the steam vault will impart jet loads on surrounding piping components. Also, plastic hinges may be developed which will permit the ruptured component to strike surrounding components and/or structures.

The break considered as the most severe from the standpoint of assuring structural integrity is considered in case VI. There are no mechanical or electrical systems in this region that must be relied upon to mitigate the effects of the break. Since these breaks are postulated to occur downstream of the anchor separating the isolation valves from the failure locations, the integrity and the operability of the valves can be maintained.

M.6.7 Case VI - Longitudinal Break Postulated at the Elbow in Main Steam Lines (Turbine Building)

This break was assumed to occur along the longitudinal axis of the elbow such that the resulting thrust loading would tend to drive the pipe against the steam vault wall. The purpose of this postulated break was to evaluate loading in the wall caused by the pipe impact and to determine the design margin for failure of the wall. A sketch of the steam piping component considered, along with pertinent analytical detail, is shown in Figure M.0-7.

At the instant the break occurs, a thrust load (F_T) of 460 kips was assumed to act along the opening. The magnitude of the thrust load was such that elastic resistance in the piping components at points A and C that counteracted the thrust load was overcome. Therefore, plastic moments were shown to develop at those hinge points.

When the pipe impacts against the wall, three separate loads will be considered to act simultaneously during the time the pipe is brought to rest. They are:

a.   Uniform pressure in the steam vault of 2.6 psi,

b.   Jet thrust load assumed to continue to be applied for a finite time after impact, and

c.   Loading associated with the deceleration of the whipping pipe.

The pipe was assumed to be brought to rest, provided the structure could be shown to be capable of withstanding the loading assumed. The analysis of the wall supporting these loads will demonstrate the assumptions are justified.

A plastic hinge was assumed to develop in the piping component when the fully plastic bending moment is exceeded. The bending moment for a plastic hinge to be developed in piping components is given as follows:

$$M_p = \frac{4}{3} \sigma_m (R_0{}^3 - R_i{}^3)$$

where R_o and R_i are the outer and inner radii and σ_m is defined for a carbon steel material to be the minimum yield stress for the operating environment considered. The material used in the fabrication of the piping is assumed to behave elastically until the fully plastic bending moment is exceeded, after which the material is assumed to behave in a perfectly plastic manner.

Since the horizontal and vertical piping lengths are long (56.26 and 36.28 feet) compared to the deflection required to cause the pipe to strike the wall (2 feet 9 inches), it is conservative to assume the two legs A-B and B-C are connected by an idealized joint that permits only bending to occur. The torsional component may be ignored and pure bending will be assumed to exist in each leg. The total load to produce a plastic hinge can be assumed to be made up of two components, one being the force required to cause a plastic hinge at A and the other to cause a hinge at C. The sum of these forces, being less than the jet thrust loading at the break, will demonstrate that a plastic hinge will be developed at the hinge points (A and C).

$$F_A = \frac{M_P}{L_{(A-B)}} = \frac{4}{3}(\sigma y)\frac{(R_0{}^3 - R_1{}^3)}{L_{(A-B)}}$$

For A 106, Grade B, at 545F,

σ_y = Yield stress = 27,000 psi

R_o = 12.0 in., R_i = 10.78 in., $L_{(A-B)}$ = 56.26 ft
FA = 25,328 lb.

Similarly, for the force required to cause a plastic hinge at C,

$$F_c = \frac{M_p}{L_{(B-C)}} = 39,277 lb.$$

$F_A + F_C$ = 64.6 kips < 460.0 kips.

Consequently, a plastic hinge will be formed at points A and C.

The jet thrust loading at the elbow was assumed to accelerate the pipe until the wall was struck, at which time the pipe was assumed to deform while decelerating, until brought to rest with no rebound from the wall. The loading at the wall during deceleration was considered from an energy point of view by equating the kinetic energy at the instant of contact with the energy required to deform the pipe in bringing it to rest.

The wall is constructed of reinforced concrete that is 4-feet 6-inches thick. The pipe was considered to be structurally weaker than the wall and was assumed to locally deform to the extent that the kinetic energy of the whipping pipe could be absorbed by the deformation in the pipe. The pipe-crushing load per linear foot of contact with the wall, during the decelerating stages, was calculated to be 33.2 kips. It will be demonstrated in the wall analyses below that the pipe-crushing load is low compared with the load-carrying capability of the reinforced concrete wall. Therefore, it is reasonable to assume that pipe deformation can be considered as a means of dissipating the kinetic energy of the whipping pipe.

The loading assumed to act simultaneously on the steam vault wall during the deceleration stages of impact, and for a sufficiently long period that is greater than the natural period of the structure, is:

a.   A uniform pressure of 2.6 psi,

b.   A uniform load of 33.2 kips/feet applied along the length of the wall calculated for each leg of the piping sections, and

c.   A concentrated jet thrust load (460 kips) assumed to be distributed along the assumed longitudinal break in the elbow.

M.6.8   Case VII - Critical Crack Loading

The critical crack was applied at those locations considered to be capable of producing the most adverse effects. The resulting jet was assumed to expand in the same manner as for the large breaks. The magnitude of the load was determined using the expressions developed in Subsection M.5.1. A summary of the loads for the steam and the feedwater systems is given below.

System	Maximum Operating Pressure psi	Break Area, in^2	Thrust Load kips
Main steam	1000	6.57	14.98
Feedwater	1250	6.57	16.43

The layout of the piping is such that no significant loading effects from the critical crack will be imparted to mechanical components. The most adverse location for the crack was considered to be in steam line A in the vicinity of the free-standing block wall in the Reactor Building part of the steam vault. A steel plate, sized to resist the jet impingement loading, has been attached to the wall to protect the isolation capabilities of the wall and to assure excessive steam is not dumped into the Reactor Building.

M.6.9   Additional High Energy Line Break Analysis

1.   Reactor Building

In addition to the main steam lines and main feedwater lines, the following high energy lines are located in the Reactor Building:

(1)   High Pressure Coolant Injection (HPCI) steam supply to the pump turbine (design pressure and temperature of 1146 psig and 562°F, respectively - saturated steam)

(2)   Reactor Core Isolation Cooling (RCIC) steam supply to the pump turbine (design pressure and temperature of 1146 psig and 562°F, respectively - saturated steam)

(3)   Reactor Water Cleanup (RWCU) System (maximum design pressure and temperature of 1300 psig and 564°F, respectively - subcooled liquid).

The above listed systems were evaluated for the purposes of 10 CFR 50.49 environmental qualification of electrical equipment for a spectrum of break locations and sizes as described below.

Break locations were chosen based on pipe stress and routing (Reference 7) within the Reactor Building such that all required double-ended breaks, longitudinal breaks, and critical cracks were analyzed.

The mass and energy (M&E) releases for these breaks were generated using the RELAP5 computer code. The steam supply lines isolate automatically on

high flow following a double-ended break or large longitudinal break. The steam supply line critical cracks and all RWCU breaks are detected by temperature switches located in the vicinity of the high energy piping which assures rapid detection and isolation. In addition, M&E releases for intermediate sized breaks were calculated for the steam supply lines. An intermediate sized break (which is a subset of the longitudinal breaks) is defined as the largest break not detectable by the high flow sensors. The intermediate sized breaks rely on the temperature switches for break detection. Isolation times for each break include break detection time and valve stroke time. The signal process time is negligible when compared with the detection and valve stroke times. The sensors used are redundant, Class 1E, and electrically trained.

A multi-node model of the Reactor Building was developed for input into the MONSTER computer code. The Reactor Building was divided such that each room was a separate compartment. The general floor areas (i.e., elevations 519.0', 565.0', 593.0', 621.0', and 639.0') were further subdivided into quadrants to better represent the flow paths around the centrally located containment. The model includes flow paths (e.g., doorways, stairwells, blowout panels, and ductwork) between various compartments, and concrete and metal heat sinks. The Uchida heat transfer coefficient was used on all heat structures with a revaporization fraction of .08. A deentrainment rate (rate at which liquid water in the atmosphere region of each compartment is removed and deposited in the pool region) was set so that all liquid is removed each time step.

The M&E releases and Reactor Building model were input to the MONSTER computer code to determine the environmental response of the Reactor Building to the high energy line breaks. The bounding breaks for individual systems within each break compartment were determined based on the total M&E releases. Temperature, pressure, and relative humidity peak values and bounding profiles were generated for each room within the Reactor Building. In conjunction with the 5% power uprate, a multi-node model of the Reactor Building was developed for input into the GOTHIC computer code. The M&E releases and Reactor Building model were input to the GOTHIC computer code to determine the environmental response of the Reactor Building to the high energy line breaks at uprated conditions. Temperature, pressure, and relative humidity bounding profiles were generated for each room within the Reactor Building. The high energy line break analyses results are reflected in the environmental data drawings (1-47E225-series - Unit 1, 2-47E225-series - Unit 2, and 3-47E225-series - Unit 3).

2.   Radiation Environment

The radiation environments inside the drywell and in the Reactor Building after a design basis LOCA were calculated consistent with the requirements of IE Bulletin 79-01B and NUREG-0588.

Initial airborne sources in the drywell were calculated assuming an instantaneous release of 50 percent of the core inventory of iodine and 100 percent of the core noble gases. Transfer of iodine from the drywell free volume to the water in the torus was conservatively calculated as a function of time until the airborne concentration was reduced by a factor of 200 (considered to be at equilibrium). Sources in the water in the torus were calculated assuming an instantaneous release of 1 percent of the core inventory of the solid fission products and 50 percent of the core inventory of iodine. Airborne activity in the Reactor Building was calculated based on a design basis leak rate from the primary containment and design flow of the SGTS.

Source terms were calculated at various times after an accident allowing for decay and dose rates were calculated with a point-kernel-with-buildup computer code. Radiation exposures in the Reactor Building due to recirculation of the torus water through the RHR and containment spray systems were also calculated. These dose rates were then integrated over the duration of the accident.

3. Main Steam Valve Vaults

The high energy lines in the valve vaults are the main steam lines and the main feedwater lines, RWCU return line, and the RCIC steam supply line. Breaks in the main steam line are controlling from an environmental standpoint due to the large line size and the high energy associated with the steam. Conditions of the main steam are 550°F, 1050 psia, quality - 1.

A double-ended rupture of the main steam was evaluated. Mass and energy releases are as described in Section M.2 for the 5.5 second closure. Break flow was terminated by isolation of the main steam lines based on signals from safety-related sensors.

4. Operation Environmental Conditions

   A listing of environmental service conditions is tabulated on the environmental data drawings (1-47E225-series - Unit 1, 2-47E225-series - Unit 2, and 3-47E225-series - Unit 3). The service conditions considered were pressure, temperature, humidity, flooding, and radiation. Normal and abnormal space ambient temperatures for nonaccident conditions were obtained from information used in the initial design phase of the plant in conjunction with data accumulated at the plant site in various spaces, for all units, under extreme outside temperature conditions (100°F outside atmosphere). Pressures and temperatures for accident conditions were obtained from transient curves and analysis which studied the effects of a LOCA/HELB on reactor zone spaces. Environmental service and conditions were considered only in the reactor zone, refueling zone, and primary containment. The control bay and electrical board room were not considered, since their atmospheres did not interface with the reactor zone environment. The environmental table of service conditions was developed for various plant conditions including the following: normal average day, abnormal conditions (outside temperature 96°-100°F maximum river water temperature) LOCA/HELB inside containment, HELB outside primary containment, and tornado (sudden pressure drop by 3 pounds per square inch).

M.7 <u>STRUCTURAL ANALYSIS</u>

The structural investigation showed that failure of the walls and slabs enclosing the vault will occur via shear if failure loading is applied, with the possible exception of the west- and east-side walls in the Turbine Building. The potential shear failures are of the diagonal tension type, as distinguished from peripheral shears closely surrounding load concentrations, or shears between a compressive type support and a "critical section" for shear located "d" distance away. All of these members were constructed of concrete for which 3000-psi compressive strength at 28 days was specified on the drawings and in the design. At age 1 year, a 75-percent increase above 28-day strength is indicated by TVA tests on fly-ash concrete. This additional strength was not generally applied in these calculations. On cross sections with shear reinforcement

$$(2 \sqrt{f_c} \quad = \quad 2 \sqrt{3000}),$$

110 psi is considered to be the point of failure in diagonal tension.

Each loading case consisted of the 460-k jet load applied to an area which varies with distance from pipe break location plus uniform pressure loading outside the jet

area. Concrete dead load was included. A dynamic load factor of 2.0 was applied to all loads and the calculations were conducted to determine the particular value of the uniform load which resulted in a failure condition.

All walls and slabs involve flexure in two or more directions. Determination of flexural failures was by yield line analysis, which rests largely on the work of Hognestad (Reference 5). Except for the east and west walls in the Turbine Building, all flexural failure mechanisms are shown to be weaker in shear than in bending. Shear failure calculations were repeated for shear distributions without consideration of shear redistribution due to the development of all yield lines.

In the Reactor Building, with the 2.0 dynamic load factor, all members will carry the 460-k jet load, plus at least 15-psi steam pressure, without exceeding the failure load.

In the Turbine Building, no member will fail under the application of the stated loads, with a 2.0 dynamic load factor, with the possible exception of the closure panel in the north wall of the chamber. This panel, with a 2.0 dynamic load factor, will carry the 460-k jet load at midpanel, plus 1.1-psi pressure. For eccentric application of the jet load, it becomes necessary to rely on the extra strength provided by the age of the concrete. Generally, any increase in additional strength greater than the 28-day strength is an added safety factor.

M.8   EFFECTS ON SAFETY-RELATED COMPONENTS AND STRUCTURES

When originally submitted, the information in Appendix M was valid. TVA's responses to IEB 79-01B, "Environmental Qualification of Class 1E Equipment," dated October 31, 1980, and supplements thereto, and the revised Order for Modification of Licenses dated September 19, 1980, provide updated information. This later information is reflected in Subsection 6.9 of Appendix M.

The safety-related components inside the Units 1, 2, and 3 Reactor Buildings were examined for the environmental effects of the steam issuing from a postulated failure in a main steam line or feedwater line in the steam vault. Of most concern were electrical components, such as pump motors, valve motors, electrical distribution boards, instrumentation power cables, and control cables. These components had to be evaluated for operability in a high-humidity environment resulting from the steam that entered the Reactor Building. The reactor building room temperatures increase for a short duration and the relative humidity approaches 100 percent.

All of these electrical components were examined and found to be essentially unaffected by the temperature and humidity. A type test of the motors has been conducted in a steam environment, and they have been shown to work during and after the test. All of the safety-related motor operated isolation valves use Limitorque operators. Those with type H insulation have been type-tested for

service inside the drywell at >300°F and a steam environment. Those with type B insulation have also been tested. All of the power and control cabling would survive these environmental conditions. The MOV board components have been evaluated for humidity or temperature. For the required operating time(s), all of the essential functions would be available for the operator to establish normal shutdown cooling after the reactor is depressurized. Thus, components of most concern were judged to be acceptable for service in the environment created when steam entered the Reactor Building from the steam vault or an alternate method of coping with failures was available. Therefore, the reactor could be put into a safe condition and maintained indefinitely.

The drywell, torus, and all equipment they contain would not be adversely affected by the pressure, temperature, and humidity in the Reactor Building. The maximum calculated pressure (<1.4 psi in Section M.2) would be less than the external design pressure (2.0 psi) for the drywell and torus. However, the drywell and torus vacuum breakers would open and admit sufficient reactor zone mixture to equalize the pressure. All of the safety-related components inside the drywell have been designed to function in such an environment and tested for the service conditions. The temperature would be much less than the primary containment design temperature of 281°F. The humidity would not cause any short-term damage. Therefore, the primary containment and equipment inside would continue to perform without any degradation to their normal function.

Another safety-related component that could be affected is the SGTS. The temperature and humidity would be below its acceptable range (<140°F and 100-percent relative humidity). The performance of the SGTS would be degraded with the blowout panel removed from the steam vault. That is, the amount of negative pressure the SGTS could establish in the Reactor Buildings would be reduced. However, the steam line-break offsite radiological dose calculations reported in FSAR Chapter 14 do not depend on the functioning of SGTS. Thus, the reduction in effectiveness of the SGTS would not result in any higher offsite doses and, therefore, was acceptable for a high-pressure pipe failure in the steam vault.

Other safety-related components, such as the diesel generators, RHRSW pumps, EECW pumps, and selected essential air-conditioning equipment, are located where they cannot be influenced by either the dynamic effects or the environmental effects of a high-pressure pipe failure in the steam vault. Essential air conditioning equipment located where it could be influenced by a high energy line break has been environmentally qualified to perform its design function under the resulting environmental conditions. Therefore, they would be available to mitigate the consequences of a postulated pipe failure in the main steam and feedwater systems outside the containment.

None of the safety-related structures, such as the intake pumping station, Diesel Generator Buildings, Control Building, and the two remaining reactor zones, would be jeopardized by the failure of a steam line or feedwater line in the steam vault. There are no direct passageways that would admit any of the steam to the pumping station or the diesel buildings. The Control Building and the other reactor zones are connected together via personnel access locks. Additionally, the reactor zones are connected to the refuel floor through large open equipment hatches. The high temperature and high humidity in one reactor zone resulting from a postulated steam vault pipe failure could possibly spread, but not very likely, through these access locks and hatches. Any inleakage of these environmental effects into the Control Building would be quickly dissipated in the air-conditioning systems in the Control Building. The perturbation on the temperature and humidity would probably not be noticed by the operators or the equipment in the Control Building. Any crossflow to the other reactor zones would be of much less concern than the environmental effects discussed above for the Reactor Building, with the postulated failure in its steam vault.

Any environmentally-induced temperature and humidity effects that would migrate to the common refueling floor would be felt in all three units.

In addition to the evaluations performed for environmental effects of the main steam line and main feedwater line breaks, the safety-related components inside the Reactor Building were examined for the environments resulting from high energy line breaks discussed in section 6.9. These evaluations are documented in the BFN Equipment Qualification Data Packages (EQDPs).

M.9 ADDITIONAL WORK

As a result of the evaluation of both the dynamic and environmental effects of a postulated failure in the main steam and feedwater systems inside the steam vault, a decision was made to increase the stiffness of the steel plate covering the stacked block wall (see Subsection M.6.8). The stiffness was added to assure that the plate does not fail and result in additional steam release into the Reactor Building during a postulated steam vault pipe failure. The stiffened plate was designed for a 10-psi uniform pressure and a 17-kip concentrated load at the worst location. A working stress analysis was used, and the maximum stress was limited to 90 percent of the yield stress. In addition, an ultimate analysis was performed with at least a 15-psi uniform pressure, a 17-kip concentrated load, and a dynamic factor of 2 applied to both loads. This additional analysis was performed to demonstrate the amount of margin available in the strength of the stiffened plate and to provide assurance that the stacked block wall does not fail. The stiffened plate was installed on Unit 1 at the first refueling outage. Stiffened plates for Units 2 and 3 were installed before those units exceeded 1 percent power.

M.10 SUMMARY AND CONCLUSIONS

This report provides partial documentation for the dynamic effects and environmental conditions resulting from a postulated failure in the main steam and feedwater systems outside the containment. The reanalysis of the consequences of the postulated event are complete, and this report supports TVA's position that the Browns Ferry Nuclear Plant complies with Criterion No. 4 of the AEC General Design Criteria as listed in Appendix A, 10 CFR 50. That is to say, all of the postulated break locations, dynamic loads, environmental conditions, structural analyses, and safety-related component evaluations reported herein would not prevent Browns Ferry from being shut down and maintained in a safe shutdown condition for the accident considered.

The qualification of safety-related electrical equipment is further addressed in TVA's responses to IEB-79-01B and supplements thereto, and the revised Order for Modification of Licenses, dated September 19, 1980.

M.11 <u>REFERENCES</u>

1. General Electric Company, System Criteria and Applications for Protection Against the Dynamic Effects of Pipe Whip. Nuclear Energy Division, Atomic Power Equipment Department, Document No. 22A2625, Revision 1, February 14, 1972.

2. Moody, F. J., Prediction of Blowdown Thrust and Jet Forces, ASME Publication No. 69-HT-31.

3. Shapiro, Ascher H., The Dynamics and Thermodynamics of Compressible Fluid Flow, Volume 1, New York: The Ronald Press Company 1953.

4. "Pipe Rupture Evaluation Program For Inside and Outside Primary Containment for the Browns Ferry Nuclear Plant Units 1, 2, and 3," CEB 88-06-C.

5. Hognestad, E., Yield Line Theory For the Ultimate Flexural Strength of Reinforced Concrete Slabs, J. ACI, Vol. 24, No. 7, pp. 637-656, March 1953.

6. G. G. Weigand, S. L. Thompson, D. Tomasko; "TWO-PHASE JET LOADS," NUREG/CR-2913, SAND82-1935 R4, January 1983.

7. BFN Design Criteria BFN-50-C-7105, "Pipe Rupture, Internal Missiles, Internal Flooding, Seismic Qualification and Vibration Qualification of Piping."

8. NRC Generic Letter 87-11, "Relaxation of Arbitrary Intermediate Pipe Rupture Requirements," June 19, 1987.