APPENDIX 14A – DESIGN REVIEW FOR CONSIDERATION OF EFFECTS OF PIPING SYSTEM BREAKS OUTSIDE CONTAINMENT

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DESIGN REVIEW FOR CONSIDERATION OF EFFECTS OF PIPING SYSTEM BREAKS OUTSIDE CONTAINMENT

1.0 INTRODUCTION

A design review was performed in response to the United States Atomic Energy Commission (AEC) request, "General Information Required for Consideration of Effects of Piping System Break Outside Containment," of December 15, 1972, for Three Mile Island Nuclear Power Station, Unit-1. Also in response to the AEC letter of June 1, 1973, an additional design review was performed. The review covers all piping systems in the unit in accordance with the criteria presented in the AEC request. All of the possible effects of postulated pipe failures as outlined in the AEC request have been considered in the review, and it is concluded that the unit can be safely shut down. This report summarizes the results of the design review and outlines the methods of achieving safe shutdown.

Design modifications will be implemented to provide assurance that the Engineered Safeguard Systems and the Emergency Feedwater System will be operable after postulated high energy pipe breaks. Inservice inspection of certain specified postulated break locations will provide assurance that rupture will not occur at those respective points. Operation of the Emergency Feedwater System will permit a more rapid shutdown capability than is possible with Engineered Safeguard Systems alone. Thus, the primary means of effecting a cooldown after a postulated break outside containment would be the Emergency Feedwater System with high pressure-low pressure injection cooldown serving as a backup.

2.0 REVIEW CRITERIA

The sets of criteria used for the review are as follows: (1) considering the type of affects that can result from a pipe break and (2) criteria addressing the degree of system operability required following a break. The criteria used are as follows:

Pipe Break Criteria

Pipes Evaluated	Effects Considered
Fluid above 200 F and 275 psig at terminal ends and high stress locations on lines 4 inches or greater	Longitudinal and circumferential breaks including pipe whip, jet impingement, flooding, and environmental conditions
Fluid above 200 F and 275 psig at terminal ends and high stress locations on lines 1 to 4 inches	Circumferential breaks including pipe whip, jet impingement, flooding, and environmental conditions

Pipe Break Criteria

Pipes Evaluated	Effects Considered
Fluid above 200 F and/or 275 psig at the most adverse locations for all pipes	Crack breaks including jet impingement, flooding, and environmental conditions
Fluid below 200 F and 275 psig	None

System Operability Criteria

System	Operability Required
Systems required to bring the unit to safe shutdown following the break	No loss of required redundancy permitted
Reactor Protection System and the Engineered Safeguards System	Loss of redundancy but no loss of function permitted

Table 14A-1, "High Energy Lines Investigated for Consequences of Postulated Pipe Ruptures," indicates the various high energy lines investigated under the above criteria and presents appropriate data or information concerning them. Figures 14A-1, 14A-2, 14A-3, and 14A-4 supplement the location and elevation data given in Table 14A-1 for the various high energy lines investigated.

3.0 GENERAL DISCUSSION

3.1 DESIGN

A rupture of the high energy piping is considered highly unlikely due to the low seismic and operating stress levels. All these systems have been conservatively designed and all the systems except auxiliary steam to the Auxiliary Building have been analyzed in accordance with USAS B31.1.0, Code For Power Piping. The auxiliary steam system to the Auxiliary Building has been designed and analyzed as described in Section 5.4.4.2. This includes all portions of the auxiliary steam system located in the Control, Fuel Handling and Auxiliary Buildings. Results of these analyses show that the maximum stress levels from combined operating and seismic conditions are well below those limits designated as potential pipe rupture stress levels.

Piping systems are designed to USAS B31.1.0. In addition, portions of the auxiliary steam system piping are analyzed in accordance with the CDFM methodology as described in Section 5.4.4.2. Quality assurance was applied to USAS B31.1.0 requirements for the non nuclear piping and to USAS B31.7 requirements for nuclear piping. (Nuclear piping is defined as piping that normally contains radioactivity.) The analysis of the auxiliary steam piping is based on the configuration and conditions of the piping system at the time of the walkdown and evaluation. To ensure that the existing condition is maintained, the quality classification of the auxiliary steam piping system has been upgraded to “Regulatory Required with QA”.

On non-nuclear piping, welders qualified to ASME Section IX requirements were used. Piping system leakage testing was performed in accordance with piping code (USAS B31.1) requirements. All welds were visually inspected.

a. The main steam piping welds 4 inches and over were 100 percent radiographed from steam generators to the turbine generator.

b. The main feedwater piping welds 4 inches and over were 100 percent radiographed from pumps to steam generator.

c. The emergency feedwater piping welds were 100 percent radiographed from the steam generators up to the first isolation valve (which is in the Intermediate Building).

d. The steam supply (to the emergency feedwater pump turbine) piping welds were 100 percent radiographed.

All the NDT required by USAS B31.7 was applied to nuclear piping systems, i.e., decay heat, makeup and purification, sampling, and so forth.

3.2 QUALITY ASSURANCE

The design and construction phase Quality Assurance Program was a three-level program. The first level of the program was performed by the equipment manufacturer or site contractor, the second level by Met- Ed's main contractor (i.e., B&W--NPGD, GAI, or UE&C, as appropriate), and the third level by Met-Ed itself and/or its agent, MPR Associates.

3.3 SHUTDOWN CAPABILITY

The unit can be brought to a cold shutdown condition by utilization of either the emergency feedwater system and atmospheric dump valves, or HPI cooling and the Reactor Building emergency coolers.

The emergency feedwater system and the atmospheric dump valves are located in the Intermediate Building. The emergency core cooling systems are located in the Auxiliary Building area, separated from the Intermediate and Turbine Buildings by the Reactor and Fuel Handling Buildings.

The highest energy lines are located in the Intermediate and Turbine Buildings. For major breaks in these lines, unit shutdown will be accomplished through utilization of the Emergency Feedwater System and atmospheric dump valves or HPI Cooling and the Reactor Building Cooling Systems. For crack breaks, a normal unit shutdown will be achieved.

4.0 DISCUSSION OF BREAKS

The layout of equipment within the buildings and the physical arrangement of the buildings themselves provide protection for the shutdown equipment from high energy line breaks. See Figure 14A-5 and Table 14A-2 for relative building and equipment locations.

Locations of postulated breaks have been determined for each of the high energy piping systems that might endanger the Emergency Feedwater System if rupture should occur. The selection of breaks is based on the results of stress analyses previously performed on the as-built piping systems. This review considered effects of pressure, deadweight, and thermal expansion during normal operating, upset, test conditions, and the operating basis earthquake (OBE). The stress levels obtained by this review for main steam and feedwater were found to be lower than those of the AEC pipe rupture criteria. To provide a conservative criterion for selecting break locations, the two intermediate points of highest stress are postulated as break locations.

Design basis breaks in straight or curved pipes 4 inches in diameter or greater are assumed to be either longitudinal or circumferential with the break area equal to the flow area of the pipe. Design basis breaks at branch and longitudinal in the run with the break area equal to the flow area of the branch. The criteria used to select design basis break locations are as follows:

a. Postulated breaks at all terminal points (anchors or rigid attachment to equipment or anchor extensions).

b. Postulated breaks at all branch points.

c. Postulated intermediate breaks between terminal points whenever the primary stress (pressure, weight, OBE) plus secondary stress (thermal) exceeds 80 percent of S_h + S_A), or where secondary stress alone exceeds 80 percent of S_A.

d. As a minimum, two intermediate breaks between terminal points were selected at locations of highest stress.

The above pipe break location criteria were applied to the high energy piping systems, and the results are tabulated in Table 14A-3. This tabulation shows the pipe isometric drawing numbers associated with each of these systems, the potential break point identification number, and the primary and secondary stresses proportional to the 80 percent allowable limit. The break locations are shown on the isometrics on Figures 14A-6 through 14A-13 and in plan on Figures 14A-1 through 14A-4.

In addition, crack breaks were postulated at adverse locations and assumed to be one half the pipe diameter in length and one half the pipe wall thickness in width.

The specific thrust versus time curves used in designing the restraints defined in this supplement are shown on Figures 14A-14 and 14A-15.

Item 15 of the Atomic Energy Commission document titled "General Information Required for Consideration of the Effects of a Piping System Break Outside Containment," Reference 12, required that a discussion should be provided for the potential for flooding of safety related equipment in the event of failure of a feedwater line or any other line carrying high energy fluid.

4.1 CONTROL ROOM AND CONTROL BUILDING

The Control Building equipment, electrical power and control, chilled water system, and ductwork systems are contained within the structure of the Control Building. In this isolated location they would not experience adverse effects from any high energy pipe break. Access to the Control Building structure is either through the Turbine Building or the Fuel Handling Building. Outside air to the Control Building is ducted to the Control Room from a remote underground intake terminal and would not be adversely affected by a high-energy pipe break. The Turbine Building would experience momentary overpressure if the break occurred in this area, but this would be dissipated through numerous wall and roof openings. Steam leakage from the turbine hall to the Control Room or Control Building during this period is minimized as it is forced to travel through the west Turbine Building wall, through multiple doors in series, before entering the Control Building areas. Also, the doors have automatic closers. Any steam leakage into the corridor space outside the Control Room or the Control Room space will be condensed and dissipated by the ventilation systems, and no significant ambient changes would be anticipated in these areas.

Investigation indicates that there are no high-energy lines larger than 1 inch other than the auxiliary steam pipe in or near the Control Building, and thus, postulated pipe whip and steam jet impingement are not able to damage the Control Room. Due to the low operating pressure of the auxiliary steam system, rupture of this line is not a consideration.

High-energy sample lines (under 1 inch) are discussed in Section 4.3.

4.2 INTERMEDIATE AND TURBINE BUILDINGS

a. The Intermediate and Turbine Buildings contain all of the lines over 1 inch with internal fluid exceeding both 200^{o}F and 275 psig.

   The pressure and temperature response in the Intermediate Building following steam line breaks are based on plant specific mass and energy release rates as shown in Table 14A-8. The thermal-hydraulic results are summarized on Table 14A-7 and supporting Figures 14A-21 through 14A-26 which indicate the temperature and pressure time history profiles for each level or compartment. As shown, the Main Steam Line Break is the limiting and enveloping break in the Intermediate Building from a pressure/temperature viewpoint. This is due to the much larger mass of steam being released compared to the EFW pump turbine steam supply line break.

   The Reactor Building will maintain its containment integrity when subjected to the resultant external pressurization of a main steam or feedwater break within the Intermediate Building. The Reactor Building will not be subjected to main steam pipe whip because the 3 ft thick interior walls of the Intermediate Building effectively restrain the pipe.

b. Electrical equipment is required to function subsequent to a High Energy Break (HELB) inside or outside of containment and must meet environmental qualification requirements. The qualification conditions considered include post accident pressure and temperature conditions in Table 14A-7. Electrical equipment which is required to

function following a postulated HELB is located at the 295 ft elevation of the Intermediate Building.

c. Isolation valves in the Intermediate Building are on lines that might be open to the containment atmosphere during normal operation (purge valve) and have been reviewed. It was found that for any postulated high energy line break, the valve will not be damaged and will close upon receiving an electrical (deenergize) signal or loss of control air.

d. The systems that will be used to bring the plant to a safe shutdown after the postulated major break in the Intermediate Building are listed in Table 14A-2. A detailed accident review was made to resolve the effect of each postulated break defined by Section 4.0 on the operability of the Emergency Feedwater Train. The objective of this review was to establish those breaks that would ultimately prevent the operation of both steam generators or both Emergency Feedwater Trains and to determine the design modifications necessary to assure emergency feedwater operation.

   1) Breaks at the containment penetrations in the small compartments 2 and 5 (refer to Figure 14A-16) could produce pressures in excess of wall and/or slab capacities. Portions of the Emergency Feedwater System below elevation 322 ft 0 inch could be damaged by the resulting debris. To provide reasonable assurance that the postulated ruptures in those compartments will not occur, the associated welds are to be inspected in accordance with Technical Specification 4.15.1.

      The breaks in compartments 3 and 4 are similar but do not produce differential pressures that would produce incremental collapse of the Intermediate Building interior structures. The method of calculating slab and wall capacity (Yield Line Theory) is reviewed in Section 7.2.1.1 of Appendix 14A and a comparison of capacities with expected differential pressures is summarized in Section 7.2.1.2 of Appendix 14A.

   2) The 12 inch main steam header below elevation 322 ft 0 inch and shown on Figure 14A-1 is positioned directly opposite emergency feedwater valve EF-V1B. The restraint/shield schematically represented on Figure 14A-17 is provided to protect valve EF-V1B from damage due to postulated breaks in the header and in the 12 inch main steam line that connects to the end of the header nearest EF-V1B.

   3) The feedwater line from containment penetration No. 103 runs approximately parallel to and above a 12 inch main steam line, which is also in the overhead of the same compartment as the turbine driven EFW Pump. To prevent both Steam Generators from becoming inoperable and the loss of the turbine driven EFW pump, the postulated breaks in this section of the pipe have been reevaluated with respect to References 15 and 16. The results of this evaluation indicate that this section of pipe can be classified as "superpipe" and that no breaks have to be postulated in this area (i.e., Break Locations 1 and 7 have been changed to 13 and 28 on Figure 14A-12). The pipe whip restraints

installed on the feedwater line to reduce pipe deflections due to a pipe break at locations 1 and 7 shall remain to reduce pipe deflections from pipe breaks at locations 13 and 28. Figure 14A-18 illustrates the protection provided.

The design procedure applied to the design of this restraint and the one discussed in Section 4.2.d.2. of Appendix 14A conform to the procedures presented in Section 7.2.2 of Appendix 14A.

4) Section 4.5 of this appendix provides an analysis of cooldown capability without using the Emergency Feedwater System.

5) An evaluation was made in Reference 13 for different alternatives available in order to mitigate flooding in the Intermediate Building in case of a postulated feedwater line break. The flood protection modifications were implemented to mitigate the effects of flooding due to a postulated main feedwater line break (MFLB) in the Intermediate Building by allowing water to flow into the tendon access gallery and the alligator pit. By removing the upper half of the western water "stop wall" on the alligator pit and opening the doors at entrance "A" and "B" to the tendon access gallery, there will be approximately 25 minutes before flooding in the Intermediate Building adversely affects the emergency feedwater system components not qualified for submergence would be adversely affected (Reference 14). Intermediate Building flood detection alarm system has been added to alert the operator in the Control Room to flooding conditions as a result of a MFLB. The alarm will provide the operator with sufficient time, approximately 20 minutes, to take corrective action to prevent damage to the EFW pumps.

e. Postulated breaks in the Turbine Building cannot adversely affect equipment utilized for the safe shutdown of the reactor. Since there is no affected reactor safety equipment in this area, the review of breaks in the Turbine Building is complete.

4.3 AUXILIARY, FUEL, AND INTERMEDIATE AREA (PERSONNEL ACCESS)BUILDINGS (AUXILIARY AREA)

Those reactor protection systems and engineered safeguards systems that could be affected by postulated pipe breaks are all located in the Auxiliary Building.

The effects of breaks in the Intermediate or Turbine Building on the Auxiliary Building ambient atmosphere will be minimal and momentary as this steam leakage to the Auxiliary Building area must pass through the west turbine building wall, through the controlled access Hot Tool Room door, and travel approximately 150 feet before entering the Auxiliary or Fuel Handling Buildings. This leakage would be continuously condensed and dissipated by the outside air ventilation exhaust systems in the corridor and in the Auxiliary and Fuel Handling Buildings.

The postulated breaks in the auxiliary area will not require protective action because they do not deplete primary system inventory or impair the normal heat removal systems (main steam and feedwater).

Another characteristic of pipe breaks in this area is that they are substantially lower in energy than breaks in the Intermediate or Turbine Building. No lines in this area larger than 1 inch

carry fluids that exceed both 200ºF and 275 psig. The largest line in this area with postulated crack breaks is 14 inches versus 24 inches in the Intermediate or Turbine Building.

All break locations in the auxiliary area will meet the established criteria (one string of each engineered safeguards system will remain operable). The break will not damage both strings of any engineered safeguards system due to the separation of engineered safeguard components. In a few cases where cabling for both trains of an engineered safeguards system runs in the vicinity of a postulated break, the effects on the cabling will be analyzed and, if necessary, protected by a barrier or rerouted as required.

The capability to detect and sustain the flooding as a result of breaks in the auxiliary area is discussed in Section 6.4.5 of the FSAR.

Postulated high energy sample line breaks will not affect any engineered safeguard/reactor protection system equipment nor render any vital building areas permanently uninhabitable.

The HVAC has been designed to confine the consequences of reactor coolant small line breaks to relatively small areas of the Fuel Handling Building at elevation 281 ft 0 inch and the radioactive sample station in the Control Room tower which are of no immediate importance in the event of a high energy line rupture. The radiation monitoring system and Control Room controls of the Reactor Building isolation valves will be used to terminate any postulated high energy sample line breaks.

The letdown line to the makeup system normally operates below 200°F and is therefore subject only to crack breaks according to the criteria prescribed in Section 2.0 of this Appendix. However, a full diameter break would result in flow sufficiently high to render the coolers ineffective and the temperature of the flow from the break would exceed 200°F. To assure sufficient conservatism, full size breaks have been postulated in the letdown line. For a break downstream of the breakdown orifice, the flow would be slightly above normal and the effectiveness of the cooler inside containment would not be significantly reduced.

Analyses performed by GPUN (Reference 17) determine postulated line break locations and address the effects of those breaks on the letdown line and nearby equipment due to jet impingement. The result of the evaluation show that the letdown line is structurally adequate to ensure that pipe whip is not an issue during a postulated line break event and that the pipe will remain stationary. Furthermore, jet impingement from the postulated rupture points does not pose a threat to any other safety related system in the plant. The compartment pressurization resulting from the break has been analyzed and found to be below that which would cause damage to the building.

The temperature, pressure, and humidity environment in the vicinity of the break has been determined by analysis (References 18, 19). The safeguard equipment in the vicinity of the break consists of valves, cabling, and pumps. The valves and cabling have been qualified for the environment expected to result from the postulated break. The pumps are in separate compartments and are not affected by the local environment. The potential flooding from a postulated letdown line break does not result in a water level high enough to impact safety related equipment or circuits.

A postulated break will be automatically isolated by closure of the Reactor Building isolation valves when high temperature is sensed in the letdown line.

Since the Makeup System in the auxiliary building is not a High Energy Line per the criteria of Section 2.0, a High Energy Line Break (HELB) need not be postulated. Therefore, this is not a Design Basis Event for TMI-1 and 10 CFR 50.49 does not apply. The environmental evaluation was performed to respond to initial licensing questions but is not in the scope of 10 CFR 50.49 (Reference 20, 21).

4.4 DIESEL GENERATOR BUILDING

The Diesel Generator Building has no direct doorway or access connection to either the Intermediate Building or the Turbine Building. However, ventilation air comes from the machine shop. Steam leakage into the machine shop would be minimal as closed doorways separate the machine shop and Turbine Building. Also, the machine shop has a separate air supply and exhaust system that would dissipate any steam leakage.

Prolonged steam conditions in this space are not to be expected. The north diesel room has no direct doorway, access, or ventilation connection to either the Intermediate Building or Turbine Building or any other area expected to become steam laden following the pipe break. Therefore, the north diesel room ambient atmosphere will not experience any change as a result of the pipe break.

4.5 COLD SHUTDOWN CAPABILITY

The adequacy of the borated water storage tank as an interim heat sink for the Three Mile Island Nuclear Station Unit-1 Reactor Coolant System has been evaluated for the following set of assumptions:

a. Steam line break occurs inside the Intermediate or Turbine Building during rated power operation

b. Reactor trips

c. Loss of all feedwater to both steam generators occurs

d. Loss of offsite power occurs

In addition to this set of assumptions, this evaluation is valid for any situation where Reactor Coolant System energy removal through the steam generators is no longer available because of a HELB in containment.

There are three primary areas of concern for this condition. These areas are prevention against core uncovering, protection against excessive Reactor Building pressure, and the ability to achieve cold shutdown conditions.

The B&W digital computer code CRAFT, Reference 10, has used to determine the characteristics of this accident with regard to core uncovering and mass energy releases to the containment. The mass and energy release data from CRAFT were used in the digital computer code CONTEMPT, Reference 11, for Reactor Building pressure calculations. The assumptions and results of the analysis are summarized in Table 14A-6. A single steam generator blowdown was considered as the most conservative case since, for a double

blowdown, the high pressure injection (HPI) pump would be started almost instantaneously on low Reactor Coolant System pressure actuation, meaning a lower probability of core uncovering.

Core uncovering is prevented by pumping water from the borated water storage tank via the Makeup and Purification System (HPI) into the Reactor Coolant System. With one makeup and purification (HPI) pump started 15 minutes after the break, the minimum coolant level in the reactor vessel occurs at approximately 140 minutes and at no time falls below the top of the core. Operator action is assumed to occur 15 minutes after the break in starting the makeup and purification pump (HPI).

The building pressure increases during the transient as boiloff occurs through the pressurizer safety valves (2515 psia). Assuming the boiloff goes directly to the building atmosphere with no credit for the quench tank, the building pressure reaches the Reactor Building cooler and high pressure injection set point (4 psig) 38 minutes after the break. With one building cooler operative at this time, the building pressure reaches a maximum value of 24 psig and never exceeds the design pressure limit. Furthermore, the reactor spray actuation set point (30 psig) is not reached, and a single building cooler provides adequate protection throughout the transient against excessive Reactor Building pressure.

High pressure injection of BWST water continues until the BWST is depleted (approximately 24 hours, assuming one HPI pump is operating). At this time, further cooldown is achieved by using the decay heat (low pressure injection) pumps drawing from the Reactor Building sump to supply suction to the makeup and purification (HPI) pumps. The sump recirculation continues until the Decay Heat Removal System (LPI) can be actuated to reduce the system to cold shutdown. Cold shutdown is then achieved by venting the system pressure and actuating the Decay Heat Removal System to recirculate the reactor coolant through the decay heat coolers.

5.0 SUMMARY AND CONCLUSIONS

The results of this design review are summarized as follows:

a. A rupture of the high energy piping systems is considered highly unlikely. The systems other than auxiliary steam piping have been conservatively designed in accordance with the criteria in USAS B31.1.0, Code for Power Piping. Materials, fabrication, and quality assurance requirements of the Code have been utilized. In addition, the main steam piping has been subject to 100 percent radiography of welds from the steam generators to the turbine stop valves, and the feedwater piping has been subject to 100 percent radiography from the steam generators to the feedwater pumps. Quality assurance provisions of USAS B31.7, Code for Nuclear Piping, have been implemented for nuclear systems. The auxiliary steam piping system has been shown to be adequate for combined effects of operating conditions and the SSE. Quality assurance provisions are implemented to ensure configuration of the auxiliary steam piping system is maintained.

b. The Atomic Energy Commission criteria had been implemented in identifying postulated break locations in high energy piping systems.

c. All of the equipment required for shutdown is protected from the postulated ruptures by virtue of location (remote from high energy lines), restraints, inspection, or barriers.

d. The Control Room will remain habitable and operable following a postulated high energy pipe break due to its remote location from such breaks.

e. The plant can be brought to cold shutdown conditions by utilizing either feedwater system or the Emergency Core Cooling System and Reactor Building cooling.

f. This analysis shows that the borated water storage tank serves as an adequate interim heat sink from the standpoint of core covering, Reactor Building pressure considerations and achieving cold shutdown.

g. Further, the analysis shows that the core remains covered, the Reactor Building is protected against excessive pressure, the plant can be taken to cold shutdown, and containment integrity can be maintained.

After Cycle 17, the original OTSGs were replaced with enhanced OTSGs. The existing HELB-related analyses and evaluations contained in Appendix 14A were reviewed in Reference 124 and confirmed to remain applicable following the TMI SG replacement.

6.0 INSPECTION REQUIREMENTS

It is intended to ultrasonically inspect the welds identified on Figures 6 and 9 in compartments 2 and 5 to the extent possible. This inspection will be performed periodically as required by Technical Specification 4.15.

7.0 METHODS OF ANALYSIS AND DESIGN

7.1 THERMAL-HYDRAULIC ANALYSIS

A detailed thermal-hydraulic analysis was performed in order to evaluate the consequences of a postulated high energy line rupture in the Intermediate Building. The analysis included the calculation of system transients and blowdown loads as well as compartment pressurization for use in the structural evaluation.

7.1.1 BLOWDOWN ANALYSIS

7.1.1.1 Main Steam Blowdown Analysis

Selection of Line Breaks

A total of Two (2) double ended guillotine pipe breaks in the Main Steam System assumed at the worst locations from an equipment qualification viewpoint; i.e., breaks are located in areas where the environmental response was most severe. Piping stress levels were not considered as an input to break location.

The selected break located at the 322 ft elevation is a guillotine break of a 24 inch main steam line upstream of the main steam line isolation valve. Guillotine breaks give the maximum total flow and most severe conditions in the building.

The location of the worst case break for the steam supply line to the EFW pump turbine is at the 295 ft elevation and is a guillotine break of the steam supply line.

Computer Model

An EDSFLOW Computer model was constructed for the Intermediate Building and was utilized to evaluate the main steam line breaks. The building was nodalized as volumes. Portals such as doorways and stairwells were modeled as connecting pathways between the volumes.

The Intermediate Building was nodalized into sixteen volumes. The 295 ft and 305 ft elevations were more precisely modeled since most of the safety related electrical equipment is located in these spaces, and a more accurate prediction of the pressure and temperature response is required. The area of the walls and ceiling of each compartment was calculated and modeled as an exposed concrete surface to simulate their heat sink effect.

The heat transfer coefficient (h) to the walls was calculated by the EDSFLOW Computer Code to be 5 Btu/F-hr-ft^2 throughout the transient for the 24 inch main steam line break case. The heat transfer coefficients calculated internal to the code as a function of transient time and is based upon fundamental natural and forced convection heat transfer correlations. This value is conservative since the effect of wall condensation was not considered. In accordance with the Uchida Condensing Heat Transfer Correlations, heat transfer coefficients substantially higher (up to a maximum of 280 Btu/F-hr-ft^2 at an air to steam ratio of 0.1) than the value calculated could have been used shortly after break initiation. For the steam supply to the emergency feed pump turbine line break case a heat transfer coefficient of 50 Btu/F-hr-ft^2 was utilized during blowdown, and 5 Btu/F-hr-ft^2 after blowdown was terminated. These values are well

below the actual condensing coefficients. The results of the analysis, therefore, are considered conservative in nature.

The thermal properties of concrete used in this analysis are listed in Table 14A-9.

The blowdown mass and energy release rates for the 24 inch line break were based on a circumferential break. The mass and energy release rates are included in Table 14A-8.

The blowdown mass and energy release rates for the 12 inch line break were calculated utilizing a modified Darcy's equation. The mass and energy release rates are 512 1bm/sec-ft^2 and 6.154 x 10^5 Btu/1bm-ft^2, respectively. It is conservatively assumed that this blowdown is constant for ten minutes without depressurizing the steam generators. After ten minutes, it is assumed emergency feedwater to the faulted OTSG would be terminated by the operator.

Key Analytical Assumptions

The following assumptions and approximations were used in this analysis:

1. Initial Intermediate Building pressure, temperature and humidity were assumed to be 14.7 psia, 90F, and 60 percent, respectively.

2. All steam lines were assumed to be at their maximum operating pressures and temperatures.

3. Forward and reverse flow shock losses were accounted for where junction flow areas were reduced or expanded in relation to the adjacent volume flow areas.

4. For the EFW pump turbine steam supply line break, doors are assumed to be in their most conservative position at the initiation of the event. If the door is closed, it is assumed to open at a differential pressure of 2 psid when opening away from the door jamb, and 4 psid when opening against the door jamb.

5. The analysis was performed without the 92 ft^2 vent area at the 322 ft elevation available. There is 75 ft^2 total vent area from the building when the opening at the 322 ft elevation is not assumed available.

6. Operator action to terminate EFW flow to the faulted OTSG feedwater is required to mitigate the EFW pump turbine steam supply line break within ten minutes.

7. The steam vent pathway out of the Intermediate Building is assumed available after a 2 psid pressure buildup in the building.

7.1.1.2 Feedwater Blowdown Analysis

The RELAP-3 digital computer code, Reference 2, was used in analyzing the feedwater blowdown transients for the determination of thrust loads. The system was represented by an assemblage of control volumes connected by flow paths or junctions. The effects of valves, pumps, heat exchangers, and check valves are included in the code.

The steam generators were modeled so that the feedwater inlet nozzles were above the steam generator water level. In this representation, backflow through the inlet nozzles would be steam.

The feedwater lines were divided into several volumes for each case. The volumes were selected so that volume size and junction location would provide optimum system representation for the particular case being analyzed.

It was assumed that for the durations of these analyses, the feedwater pumps would continue to operate and that flow would be a function of head. It was further assumed that for the duration of these analyses, an unlimited constant pressure supply of water was available at the feedwater pump suction. Both main feedwater pumps were combined and modeled as a single pump.

In modeling flow nozzles, the actual nozzle throat area was used if flow to the leak was in a forward direction through the flow nozzles. Where flow to the leak was in a reverse direction through the nozzle, an effective flow area was calculated and used in the model.

Basic Assumptions

a. Reactor operation at full load conditions.

b. Steam generator nominally at 925 psi with feedwater inlet at 1000 psi and 462^{o}F.

c. Feedwater line check valve fails to close.

d. Pumps do not trip.

e. Circumferential and longitudinal breaks were considered.

f. Break volumes were selected to account for the segment of piping up to the first elbow on either side of the break.

7.1.2 COMPARTMENT PRESSURIZATION ANALYSIS

7.1.2.1 Main Steam Line Ruptures

The pressure temperature transients resulting from the postulated rupture of a main steam line in the Intermediate and Turbine Buildings were investigated.

The transients for subcompartment integrity evaluation were calculated by extending the short term main steam blowdown model to include control volumes representing compartments of the Intermediate Building with their interconnecting vent area. Figure 14A-16 presents the Intermediate Building compartment designations. Double-ended circumferential ruptures were considered as the limiting case. Table 14A-4 presents the maximum wall and slab pressure differentials obtained for the Intermediate Building subcompartments where over-pressurization was investigated as a potential problem.

7.1.2.2 Feedwater Line Ruptures

With its significantly lower energy content, the feedwater line rupture does not represent a problem with respect to compartment pressurization.

7.1.3 PIPE THRUST AND JET IMPINGEMENT

The thrust forces developed by a jet flow being expelled from a ruptured pipe and calculated from the internal pipe pressure and the density of the mass being accelerated out of the rupture area. The conditions used to calculate this thrust assume that the rupture occurs over 1 millisecond with the maximum thrust force being established by using the corresponding pressure and mass flow conditions. As the blowdown of the system progresses, the pipe stagnation pressure is assumed to be equal to the static pressure at the rupture location for both longitudinal breaks and for circumferential breaks. The thrust calculation used for these analyses includes the integrated pressure and momentum effects and does not take advantage of upstream flow restrictions to reduce the pipe pressure.

Thrust forces on those lines not affected by blowdown characteristics during the first 15 seconds were assumed as 1.26 PA for steam and 2.0 PA for subcooled fluid.

The forces on targets in the path of escaping fluids are dependent on the size and shape of the target and its distance from the rupture area. Since the actual shape of the rupture dictates the flow field shape being generated, it was assumed that a typical rupture is circular and the free stream expansion of the jet to be conical with an included angle of 30 degrees. Calculation of the force on an object was determined by assuming that the dynamic pressure developed at the rupture exit is for the maximum mass flow and pressure conditions in the high energy line and is inversely proportional to the cross section area of the conical expansion being generated. This dynamic pressure was applied on the targets assuming a target drag coefficient of 2.0, i.e., complete stagnation of the escaping fluid.

7.2 STRUCTURAL ANALYSIS AND DESIGN

7.2.1 WALL AND SLAB CAPACITIES

Postulated ruptures of the main steam lines at the containment penetrations were investigated to determine differential pressures in compartments 2 through 5. Peak differential pressures are presented in Section 7.1.2. To investigate the retention or loss of structural integrity due to breaks within the small compartments, the load carrying capacity of slabs and walls was calculated and compared to expected differential pressure.

7.2.1.1 General Yield-Line Theory and Shear Capacity

Yield-Line Theory (References 3, 4, and 5) takes into consideration the inelastic behavior of the reinforced concrete structural element (wall or slab) in developing a mechanism prior to loss of structural integrity. In brief, the steps involved in the evaluation of the uniform pressure capacity of slabs and walls are as follows:

a. The ultimate moment capacity of the cross section is calculated at various negative and positive moment regions, i.e., at the boundaries and at midspan. The sections are typically doubly reinforced and the moment capacity is given conservatively by:

$$M_u = 0.9A_sF_y\ (1d - {}^{a}/2)$$

where:

M_u = moment capacity/ft

A_s = area of tension steel/ft

F_y = yield strength of reinforcing steel

d = effective depth from steel centroid to extreme compression fiber

a = depth of equivalent compression concrete

The estimate is conservative in that it neglects the increase in strength of the tension steel after yielding occurs. Also, section capacities are reduced by a factor of 0.9.

b. The support conditions for the slab or wall under consideration are determined. Where the supports of a given wall or slab are connected to slabs or walls of approximately equal thickness and when the reinforcing steel is sufficiently anchored to develop its strength, the support condition is taken as being fixed. Where the above conditions are not satisfied, the support condition is assumed to be simple support or free in the case of free edges adjacent to containment.

c. The correct yield-line pattern is established by trial and error using the Principle of Virtual Displacements or The Equilibrium Method. Sufficient trails are executed to determine the uniform load-carrying capacity (absolute minimum of all trials). A 10 percent reduction is applied to all results to allow for corner effects.

The uniformly distributed load corresponding to the correct yield-line pattern is taken to be the differential pressure capacity of the wall or slab in bending.

d. Each wall or slab is checked for punching shear (two way action) around its periphery (the perimeter is defined at a distance d/2 from the support lines) and for local shear (one way action) at a distance d from support faces.

The concrete shear strength for two way action is taken as $4\sqrt{f'c}$ and the slab capacity is calculated by:

$$W_{ps} = \frac{4\sqrt{f'c}\, b_o d}{A_p}$$

where:

W_{ps} = pressure-producing punching shear capacity

b_8 = perimeter length

d = effective depth

A_p = slab or wall surface area bounded by b

The concrete shear strength for one way action is taken as:

$$V_c = 1.9\sqrt{f'c} + 2500\ P\ \frac{Vd}{Mu} \qquad 3.5\ \sqrt{f'c}$$

as defined in Reference 6. The slab capacity is then calculated by considering the most critical section at d distance from a support face.

The wall or slab capacity as controlled by shear is taken as the lesser of the two values described above.

e. The maximum differential pressure capacity is then the lesser of the two values from c. (as controlled by moment) and d (as controlled by shear).

7.2.1.2 <u>Compartment Differential Pressures Versus Capacities</u>

The differential pressure results defined in Section 8.1.2 for circumferential breaks in compartments 2 through 5 are presented in Table 14A-5. A comparison of peak differential pressures to the wall or slab capacities indicates the following:

a. Structural slabs and walls in compartments 2 and 5 may experience excessive differential pressures due to postulated breaks. Protection will be provided as described in Section 4.2.d.1 of this Appendix.

b. Postulated breaks in compartments 3 and 4 should not result in wall or slab failure and no protection is required.

7.2.1.3 <u>Jet Impingement And Combined Differential Pressure</u>

Jet impingement is a direct result of either a longitudinal or circumferential pipe break. Longitudinal pipe rupture, which can occur at any orientation about the circumference of the pipe at the break point, results in a jet axis perpendicular to the longitudinal axis of the pipe. Circumferential (guillotine) breaks result in a jet axis parallel to the longitudinal axis of the pipe. The jet is assumed to diverge from the break at a 30 degree conical angle, and the total integrated force on an object is determined by assuming that the dynamic pressure developed at the rupture exit is inversely proportional to the cross sectional area of the conical expansion cone at any station. Thus, the walls and slabs are evaluated for the effects of the jet.

After investigating various postulated pipe ruptures, the most severe case of jet impingement from a main steam break in compartments 3 and 4 is a longitudinal break acting on the wall between compartments 4 and 5. This is due to the fact that the main steam pipe is approximately 2 feet away from the wall, and the jet impingement force from a postulated side split break at the containment vessel penetration strikes the wall at its unsupported edge. Jet

loads in compartments 2 and 5 were not investigated since protection against those breaks is provided as stated in Section 4.2.d.1 of this Appendix.

The capability of the wall between compartments 4 and 5 to resist the jet impingement force was analyzed using Yield-Line Theory. The jet impingement load was conservatively taken as a point load acting at the free edge of the wall. This analysis assumed a semicircular fan-shaped crack pattern and resulted in an ultimate capacity approximately three times higher than the jet force.

The wall was then evaluated for its ability to resist the punching shear caused by the jet impingement. The punching shear capacity of the wall was found to be approximately four times the punching shear stress caused by the jet impingement force.

If the impingement force is assumed to be instantaneous and a dynamic load factor of two is used, both the moment and shear checks stated above are more than satisfactory. The jet load in this case was not combined with pressure since the peak jet force occurs at the instant of break and the peak pressure occurs later.

The wall between compartments 4 and 5 was also analyzed for the combined effects of jet impingement and pressurization at a time subsequent to pipe break. The ultimate moment capacities of the wall needed to resist the pressurization load and the jet impingement was less than the available ultimate moment capacity of the wall. The technique for combining the two loadings is described in Reference 5. In a similar manner, the combined shear stress caused by pressurization and jet impingement was found to be less than the available shear capacity of the wall. All other cases of combined pressurization and jet impingement in compartments 3 and 4 were less critical than the case discussed.

7.2.2 RESTRAINT DESIGN PROCEDURE

7.2.2.1 Assumptions

The assumptions involved in designing restraints are as follows:

a. Guaranteed minimum yield strength of pipe steel reduced in accordance with operating temperature.

b. Guaranteed minimum ultimate strength of the pipe steel is unaffected by temperature.

c. Ultimate strain of both piping and restraint material is one half of guaranteed minimum percent elongation.

d. Guaranteed minimum values of yield strength and ultimate strength for restraint material A 36 are taken from applicable ASTM specifications.

e. A 10 percent increase in material properties is applied to allow for strain rate effect.

The design procedure discussed in the following section utilizes computer program DYREC (Reference 7). Final designs of restraints were investigated using the techniques presented.

7.2.2.2 Analytical Method

The dynamic analysis of lumped mass models of the rupturing pipe and restraint system was performed by direct numerical time integration of the equations of motion. The computer program DYREC includes the following capability:

a. Element types; bilinear beam, bilinear axial or rotational spring, "special" axial or rotational spring.

b. Plastic, elastic, or elasto-plastic impact after closing specified gaps.

c. Constant zero or non-zero nodal boundary conditions.

d. Piecewise linear force-time histories.

7.2.2.3 Material Properties

Pipe Material - A 106 Grade B

F_y = Yield strength at 600F	= 25.9 ksi x 1.1 = 28.5 ksi
F_{ult} = Ultimate strength	= 60 ksi x 1.1 = 66.0 ksi
% Elongation	= 22 percent
E = Modulus of elasticity	= 25.7 x 10^3 ksi

Structural Steel - A36

F_y = 36 ksi x 1.1	= 39.6 ksi
F_{ult} = 60 ksi x 1.1	= 66.0 ksi
% Elongation	= 20 percent
E = Modulus of elasticity	= 30 x 10^3 ksi

7.2.2.4 Section Properties

a. Pipe Cross Section

The bilinear moment versus curvature relationship is defined by the following points:

$M_y = F_y S$

$M_{ult} = F_y Z + (F_{ult} - F_y)S$

Where:

S = Elastic Section Modulus

Z = Plastic Section Modulus

E_y = F/E = Yield Strain

E_{ult} = (% elongation)/2 = Ultimate Strain

$\varnothing_y$ = $2\ E_y/d$ = Yield Curvature

$\varnothing$ ult = $2\ E_{ult}/d$ = Ultimate Curvature

Where:

d = O.D. of pipe cross-section

The shear versus shearing strain relationship for pipe cross section is defined by:

F_{sy} = Shear Yield Stress* = $Fy/\sqrt{3}$

A_s = Effective Shear Area 8 = 0.53 A pipe

Y_y = Shear Yield Force = $A_s\ F_y$

G = Shear Modulus = 0.4E

Y_y = Shear Yield Strain = $Y_y/(A_s\ G)$

Y_{ult} = Ultimate Shear Force = $2Y_y$

Y_{ult} = Ultimate Shear Strain = 309

* Von Mises Criteria

b. Structural Wide Flange Cross Section

In defining the moment versus curvature relationship and shear versus shearing strain, the moment is assumed to be carried by the flanges and the shear by the web.

M_y = $F_y\ A_f\ (d-t_f)$

Where:

A_f = Area of One Flange

d = Depth of Cross-Section

t_f = Flange Thickness

$M_{ult} = F_{ult}\ a\ (d\text{-}t_f)$

$E_y = 2\ E_y/d$

$E_{ult} = 2\ E_{ult}/d$

$Y_y = F_{sy}\ A\ W_{eb}$

$V_y = V_y/(A_{web}\ G)$

$V_{ult} = 2\ V_y$

$V_{ult} = 30\%$

The bilinear moment versus curvature and shear versus shearing strain curves are illustrated on Figure 14A-19.

7.2.2.5 Example Model

The pipe condition at rupture, shown on Figure 14A-20 part A, is representative of a postulated break in the 12 inch main steam header. Figure 14A-20 B part illustrates the dynamic model of the same condition.

The restraint is positioned so as not to interfere with normal piping operation. Therefore, the model has a gap of 4 inches between nodes 3 and 8 at the instant F(t) is applied, i.e., at the time postulated rupture occurs. Node 8, in this case, is representative of the mass of the header that is assumed to break away. The modeling is conservative in that credit is not taken for the restraint offered by the 12 inch branch line coming into the 12 inch header.

Nodes 1 through 7 and bilinear beam elements 1 through 6 represent two structural wide flange sections embedded in the existing slab.

The fineness of lumping as well as the magnitude of time increment for numerical integration are selected to ensure a reasonable approximation of the dynamic transient. The total time of execution of the mathematical model on the computer is set to allow multiple impacts of the rupturing pipe and observance that the selected restraint is indeed bringing the rupturing pipe to a stable steady state condition.

7.2.2.6 Detail Design

The details of the restraint design are considered in two parts, steady state and transient effects.

The restraint is proportioned such that after the transient occurs, the shears, moments, reactions, and so forth, are within the allowable values of applicable codes (e.g., AISC, manual of Steel Construction, 7th Edition). During the transient, the maximum element curvatures and shears are limited to approximately one half of their ultimate values.

8.0 REFERENCES

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TABLE 14A-1
(Sheet 1 of 2)

High Energy Lines Investigated for Consequences of Postulated Pipe Ruptures

Description of Line	# of lines	Nom Line OD, in.	Type of Fluid Contained	Normal Operation		Main Bldg Areas Occupied by these lines				Remarks
				Temperature °F	Pressure psig	Intermediate Bldg Elev.	Turbine Bldg Elev.	Auxiliary Bldg Elev.	Fuel Handling Bldg Elev.	
Main Steam (MS)	4	24	Superheated Steam	570	900	337' 7" & 347' 1"	347' 1"	None	None	None
Main Steam Dump	2	12	Superheated Steam	570	900	346' 0"	345' 8" & 342' 9"	None	None	None
MS supply to EFW Pump Turbine and MS Dump Lines	4	8 indr takeoffs	Superheated Steam	570	900	338' 8"	None	None	None	8 inch takeoffs rise off MS Line in small Intermediate Building cubicles and drop vertically into 10 inch headers that terminate at end of main headers
	2	10 Sub header				~ 314'				
	1	12 header				297' 0"				
Main Feedwater (FW)	2	20 FW Pump Disch	Hot Water (subcooled)	455	>950	346' 0" to 324' 0" & 316' 2"	30" – 324' 3" 20" – 346' 0" & 324' 0"	None	None	Pressure varies due to HD loss in piping system. One 20 inch discharge from common 30 inch manifold is in Int. Building. The rest are in Turbine Building
	1	30 Common Man.								
	2	20 Outlets from common man.								
Aux Steam (AS) to EFW Pump Turbine	1	4	Saturated steam	382	185	344' 0" to 303' 6"	None	None	None	None
Emergency Feedwater Pump Discharge (EFW)	1	6	Cold Water	90	>1000	304' 0" & 306' 6'	None	None	None	The 3 EFW pump discharge lines tie into common 6 inch manifold and two 6 inch outlet lines feed out of this into the Reactor Building.
	2	4				297' 7" & 306' 6'				
	2	6				298' 3" & 318' 0"				
Aux Steam (AS) to Auxiliary Building	1	4	Saturated steam	227	5	None	365' 10"	289' 2" to 288' 8"	363' 6" & 362' 6' to 329' 0"	In passageway between Fuel Handling to Building and Control Tower. One 2 inch branches off within Fuel Handling Building at 339 ft.
	1	6					366' 7" & 356' 6"	288' 8" to 294' 1"	329' 0" to 328' 8"	
	1	8					356' 6" & 363' 7'	294' 1" to 296' 0"	301' 3" to 289' 2"	

TABLE 14A-1
(Sheet 2 of 2)

High Energy Lines Investigated for Consequences of Postulated Pipe Ruptures

Description of Line	# of lines	Nom Line OD, in.	Type of Fluid Contained	Normal Operation		Main Bldg Areas Occupied by these lines				Remarks
				Temperature °F	Pressure psig	Intermediate Bldg Elev.	Turbine Bldg Elev.	Auxiliary Bldg Elev.	Fuel Handling Bldg Elev.	
Makeup Pump Discharge	3	4 Pump Disch	Cold Water	135	>2250	None	None	287’ 9” to 284’ 3”	291’ 0”	Refer to Figure 3 for details of pipe routing of discharge piping from makeup pumps
	1	4 Comm Manf								
	3	4 Manf Outlet								
Letdown to Makeup & Purification	1	2 1/2	Cold Water	120	~ 2200	None	None	291’ 6’ 289’ 0” & 282’ 0”		Splits into 3 branch manifolds where pressure is broken down to between 25 and 75 psig
Decay Heat Suction	1	12 RC LETDOWN	Water varying from hot to cold	250 to between 110 & 140	300 to atmospheric plus static head	None	None	291’ 6” 263’ 0”	None	
	2	12 BRANCH								
	2	14 PUMP SUCTION								
Decay Heat Discharge	2	10	Water varying from hot to cold	210 to between 100 & 130	~ 400 ~ 150	None	None	272’ 9” 274’ 0” 291’ 6”	None	None
Reactor Coolant Sample	1	3/8	Hot water (subcooled) and steam	Between 540 & 605 Steam is saturated	~ 2200	None	None	312’ 0”	Between Reactor Bldg and FH Bldg	Line passes through passageway between FH Building and Control Tower and enters Control Tower as same place as Steam Gen. sample, below
Steam Generator Secondary Side Sample	2	3/8	Hot water (subcooled)	~ 500	~ 900	None	315’ 6”	None	315’ 6” In passageway between FH Bldg and Control Tower	Line enters radioactive sample station in room at elevation 306 feet 0 inch near northwest corner of Control Tower

TABLE 14A-2
(Sheet 1 of 2)

SYSTEMS REQUIRED FOR SAFE SHUTDOWN FOLLOWING A POSTULATED MAIN STEAM OR FEEDWATER PIPE RUPTURE

System	Building	Equipment Location Elevation	Area
Makeup and Purification System	Auxiliary	281-0	Northeast section
Decay Heat Removal Systems			
D.H. Pumps and Coolers	Auxiliary	261-0	North-central section
Closed Cycle Cooling System Coolers	Auxiliary	271-0	West section
Closed Cycle Cooling Pumps section	Auxiliary	305-0	North-central
River Water Pumps	Screen House	308-0	East-central section
Borated Water Storage Tank (BWST)	Outdoors	306-0	Near northeast corner of Auxiliary Building
Nuclear Services Cooling Water System			
Closed Cycle Cooling Coolers	Auxiliary	271-0	West section
Closed Cycle Cooling Water Pumps	Auxiliary	305-0	North-central section
River Water Pumps	Screen House	308-0	East-central section
Air Handling (ES)			
Reactor Building Cooling Units	Reactor	287-0	East-central section
Cooling Water Piping	Intermediate	295-0	North-central section
Reactor Building Coolers	Reactor	291-0	West end
Diesel Generators			
Diesels	Diesel	306-0	Most of Bldg. area
Fuel Tanks Underground	Outside	Below grade	North of Diesel Generator Building

TABLE 14A-2

(Sheet 2 of 2)

SYSTEMS REQUIRED FOR SAFE SHUTDOWN FOLLOWING A POSTULATED MAIN STEAM OR FEEDWATER PIPE RUPTURE

Equipment Location System	Building	Elevation	Area
Spent Fuel Cooling System	Fuel Handling	281-0	Central west side
Pressurizer Safety Valves and PORV	Reactor	354-0	On top pressurizer
RPS and ES Actuation	Reactor and Auxiliary		
RPS and ES Actuation Racks	Control		
Emergency Feedwater	Intermediate	295-0	East section
Atmospheric Dump Valves	Intermediate	295-0	EF-P-1 & Bypass Header Room

TABLE 14A-3
(Sheet 1 of 1)

TABULATION OF SYSTEMS AND BREAK LOCATIONS

System	Figure No. and Reference Isometrics	Break Locations Indicates on Isometrics	Secondary Street, % of Allowable (Thermal)	Primary + Secondary Stress, % of Allowable (deadweight + Thermal ±Seismic)
Generator "B"	Figure 14A-11	29	63.9	57.5
to Header		43 end	7.4	21.2
at EL 297'-0"		13 end	6.7	59.3
		12	20.6	2.8
		10	29.7	43.6
		9 end	32.4	45.5
Feedwater	ISO	13	4.3	36.5
from 30" Header		28	27.0	49.1
to Penetration	Figure 14A-12	30	28.9	50.6
103		44 end	22.3	55.4
Feedwater	ISO	1 end	50.7	49.4
from 30" Header		12	27.8	46.1
to Penetration	Figure 14A-13	8	36.7	44.2
227		53 end	42.3	51.7
Main Stream	Figure 14A-1	1 end	00.7	21.2
Dump to		116	89.0	95.8
Intermediate		115	87.3	88.8
Wall		117 end	44.2	59.7
Main Stream	Figure 14A-1	1 end	43.5	53.8
Dump to		29	72.3	77.7
Intermediate		28	71.0	77.2
Wall		43 end	27.8	46.0

TABLE 14A-4
(Sheet 1 of 2)

MAXIMUM COMPARTMENT PRESSURES

Circumferential Break

Steam Line Rupture in Compartment 2

Maximum Pressure in Compartment 2 = 82.1 psia = 67.4 psig
maximum Differential Pressure,
Subcompartment 2
to Subcompartment 3 = 65.7 psid
6 = 63.0 psid
1A = 66.4 psid
12 = 66.5 psid
Reactor Bldg = 67.4 psid
Turbine Bldg = 67.4 psid

Steam Line Rupture in Compartment 3

Maximum Pressure in Compartment 3 = 45.1 psia = 30.4 psig
Maximum Differential Pressure,
Subcompartment 3
to Subcompartment 2 = 27.4 psid
4 = 29.8 psid
1B = 26.6 psid
7 = 28.8 psid
13 = 3.8 psid
Reactor Bldg = 30.4 psid

Steam Line Rupture in Compartment 4

Maximum Pressure in Compartment 4 = 36.3 psia = 21.6 psig
Maximum Differential Pressure,
Subcompartment 4
to Subcompartment 3 = 20.8 psid
5 = 20.8 psid
1C = 12.9 psid
14 = 2.4 psid
8 = 18.8 psid
Reactor Bldg = 21.6 psid

TABLE 14A-4
(Sheet 2 of 2)

MAXIMUM COMPARTMENT PRESSURES

Circumferential Break

Steam Line Rupture in Compartment 5

Maximum Pressure in Compartment 5 = 70.1 psia = 55.4 psig
Maximum Differential Pressure,
Subcompartment 5

to Subcompartment	11	= 54.3 psid
	4	= 54.3 psid
	1D	= 53.5 psid
	15	= 54.3 psid
	9	= 52.3 psid
Reactor Bldg		= 55.4 psid

Longitudinal Break

Steam Line Rupture in Compartment 3

Maximum Pressure in Compartment 3 = 33.07 psia = 18.37 psig
Maximum Differential Pressure,
Subcompartment 3

to Subcompartment	2	= 15.99 psid
	4	= 17.27 psid
	1B	= 16.05 psid
	7	= 16.62 psid
	13	= 1.32 psid
Reactor Bldg		= 18.37 psid

Steam Line Rupture in Compartment 4

Maximum Pressure in Compartment 4 = 21.67 psia = 6.97 psig
Maximum Differential Pressure,
Subcompartment 4

to Subcompartment	3	= 3.57 psid
	5	= 3.52 psid
	1C	= 1.62 psid
	14	= 1.02 psid
	8	= 2.87 psid
Reactor Bldg		= 6.97 psid

TABLE 14A-5
(Sheet 1 of 2)

COMPARISON OF WALL AND SLAB
CAPACITY TO DIFFERENTIAL PRESSURE

Structural Member	Break In Compartment No.	Capacity (psi)	Peak Differential Pressure (psi)
Slab, Compartment 2	2	47.8	66.4
Wall, Compartment 2 to 3	2	44.4	65.7
Wall, Compartment 2 to 6	2	39.0	63.0
Roof, Compartment 2	2	54.0	66.5
Slab, Compartment 3	2	32.3	26.6
Wall, Compartment 3 to 4	2	43.7	29.8
Wall, Compartment 3 to 7	3	39.0	28.8
Wall, Compartment 3 to 12	3	44.4	27.4
Roof, Compartment 3	3	36.5	3.8
Slab, Compartment 4	4	9.3	12.9
Wall, Compartment 4 to 3	4	43.7	20.8
Wall, Compartment 4 to 5	4	53.2	20.8

TABLE 14A-5
(Sheet 2 of 2)

COMPARISON OF WALL AND SLAB
CAPACITY TO DIFFERENTIAL PRESSURE

Structural Member	Break In Compartment No.	Capacity (psi)	Peak Differential Pressure (psi)
Wall, Compartment 4 to 8	4	40.2	18.8
Roof, Compartment 4	4	33.5	2.4
Slab, Compartment 5	5	19.2	53.5
Wall Compartment 5 to 4	5	53.2	54.3
Wall, Compartment 5 to 9	5	47.0	52.3
Roof, Compartment 5	5	50.1	54.3

TABLE 14A-6
(Sheet 1 of 1)

CHRONOLOGY OF EVENTS FOR HIGH ENERGY PIPE BREAK

TIME (SEC)	EVENT
0	Double-ended break of a 24 inch diameter steam line on the secondary side
1	Reactor trip on variable low pressure; turbine stop valves close, isolating the unaffected steam generator
47	Damaged steam generator blows dry
450	Unaffected steam generator provides no more heat sink; minimum system pressure of about 1550 psia is reached
900	Operator action starts one HPI pump
1200	Primary loop becomes solid with subcooled water; pressurizer code relief valve opens at set point of 2515 psia
2300	Reactor Building cooler actuation set point of 4 psig is reached
5700	Steam first appears in the core
8500	Minimum coolant level in reactor vessel is reached; core remains covered
8800	Containment Building pressure reaches the maximum value of 24 psig

TABLE 14A-7
(Sheet 1 of 1)

THREE MILE ISLAND NUCLEAR STATION - UNIT-1

INTERMEDIATE BUILDING ENVIRONMENTAL CONDITIONS

Main Steam Line Break at 322'-0" Elevation

Elevation (ft)	Peak (F) Temperature	Peak (psia) Pressure	Pressure & Temperature Profiles
355	187.0	24.2	-
322	326.0	24.2	Figure 14A-23
295	322.0	23.6	Figure 14A-21

Main Steam to EFW Pump Turbine Line Break

Elevation (ft)	Peak (F) Temperature	Peak (psia) Pressure	Pressure & Temperature Profiles
355	95.0	16.7	-
322	205.0	16.7	Figure 14A-24
295	273.0	16.7	Figure 14A-22

TABLE 14A-8
(Sheet 1 of 3)

MASS AND ENERGY RELEASE RATES

FOR 24 INCH MAIN STEAM LINE BREAK

Time After Rupture (sec)	Piping Blowdown (lbm/sec)	Enthalpy (Btu/lbm)	OTSG Blowdown (lbm/sec)	Enthalpy (Btu/lbm)
0.0	0.0	0.0	0.0	0.0
0.05	4.973+3	1241.3	6.476+3	1234.0
0.1	5.962+3	1234.9	4.715+3	1234.3
0.1001	5.000+2	1234.9	4.715+33	1234.3
0.2	5.000+2	1234.9	4.411+3	1204.6
0.3	5.000+2	1234.9	4.214+3	1199.7
0.4	5.000+2	1234.9	4.038+3	1200.5
0.5	5.000+2	1234.9	3.876+3	1201.2
0.75	5.000+2	1234.9	3.530+3	1202.5
1.0	5.000+2	1234.9	3.252+3	1203.5
1.5	5.000+2	1234.9	2.923+3	1204.3
2.0	5.000+2	1234.9	2.700+3	1204.9
3.0	5.000+2	1234.9	2.224+3	1205.4
4.0	5.000+2	1234.9	1.948+3	1205.0
5.0	3.518+2	1174.6	2.186+3	1214.1
6.0	3.00 +2	1172.4	2.299+3	1206.3

TABLE 14A-8
(Sheet 2 of 3)

MASS AND ENERGY RELEASE RATES

FOR 24 INCH MAIN STEAM LINE BREAK

Time After Rupture (sec)	Piping Blowdown (lbm/sec)	Enthalpy (Btu/lbm)	OTSG Blowdown (lbm/sec)	Enthalpy (Btu/lbm)
7.0	3.10+2	1172.9	2.465+3	1205.8
8.0	3.29+2	1173.7	2.556+3	1205.9
9.0	3.38+2	1174.0	2.630+3	1205.4
10.0	3.46+2	1174.4	2.689+3	1205.1
11.0	3.50+2	1174.6	2.702+3	1204.9
12.0	3.49+2	1174.5	2.693+3	1204.9
13.0	3.47+2	1174.4	2.669+3	1205.0
14.0	3.43+2	1174.3	2.637+3	1205.0
15.0	3.39+2	1174.1	2.604+3	1205.1
17.0	3.21+2	1203.8	2.534+3	1205.2
20.0	3.07+2	1205.0	2.440+3	1205.3
25.0	2.88+2	1205.0	2.286+3	1205.4
30.0	2.69+2	1205.6	2.150+3	1205.3
35.0	2.50+2	1205.4	2.001+3	1205.2
40.0	2.33+2	1205.2	1.864+3	1204.8

TABLE 14A-8
(Sheet 3 of 3)

MASS AND ENERGY RELEASE RATES

FOR 24 INCH MAIN STEAM LINE BREAK

Time After Rupture (sec)	Piping Blowdown (lbm/sec)	Enthalpy (Btu/lbm)	OTSG Blowdown (lbm/sec)	Enthalpy (Btu/lbm)
45.0	2.18+2	1204.8	1.748+3	1204.5
50.0	2.05+2	1204.4	1.642+3	1204.0
55.0	1.93+2	1203.9	1.543+3	1203.5
60.0	1.81+2	1203.3	1.452+3	1202.9
70.0	1.57+2	1201.9	1.200+3	1201.3
75.0	5.79+1	1197.2	6.128+2	1189.9
75.0-3600	0.0	0.0	8.700+1	1234.4

TABLE 14A-9
(Sheet 1 of 1)

CONCRETE PROPERTIES

Temperature (F)	Thermal Conductivity K, (Btu/hr - Ft^{20}F)	Volumetric Heat Capacity (Btu/ft^{30}F)
20	0.80	30.0
200	0.81	30.0
300	0.82	30.0