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REVIEW OF HIGH ENERGY LINE ANALYSIS FOR THE MONTICELLO NUCLEAR GENERATING STATION

Prepared for:

Northern States Power Company

Prepared by:

NUTECH Engineers, Inc.
Minneapolis, Minnesota

Prepared By *Patrick M. Donnelly*
P. M. Donnelly, P.E.
Project Engineer

Reviewed By *Art McSherry*
A. J. McSherry
Engineering Manager

Prepared By *E. R. Nelson*
E. R. Nelson
Senior Consultant

Issued By *Thomas N. Vogel*
T. N. Vogel, P.E.
Project Manager

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NAME / TITLE	INITIALS
E. R. Nelson/Senior Consultant	ERN
P. M. Donnelly, P.E./Project Engineer	PMD
A. J. McSherry, Engineering Manager	AJM
J. D. Freichels/Consultant I	J.D.F.

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1.0 INTRODUCTION

The following report summarizes the results of an evaluation performed for the Northern States Power Company, on the 1973 Monticello High Energy Line Break (HELB) Analysis. The objective of the evaluation was to independently review the systems and conclusions identified in the 1973 HELB evaluation for accuracy, and to evaluate the effects of system configuration changes on the original conclusions for the components, equipment, and structures essential for safe shutdown of the plant.

This report defines the high energy lines associated with the systems of interest, and for each, provides detailed information regarding the method of compliance and/or additional evaluations required to demonstrate compliance with the required HELB criteria. Each HELB system was evaluated individually, for the effects of pipe whip, jet impingement, flooding, equipment qualification, and safe shutdown paths.

The remaining sections of this report provide background information, the evaluation approach and methodology, safe shutdown system requirements, the results of the evaluation, and conclusions and recommendations for additional activites based on the evaluation results.

It should be noted that the conclusions contained in this report reflect the information available at the time of its generation. Revisions to the results and conclusions may be required as additional information is received.

2.0 BACKGROUND

On December 18, 1972 a letter was transmitted to all operating plants by the Atomic Energy Commission (AEC) (Reference 7.1) requesting information relative to the effects of a postulated rupture of piping containing high energy fluid.

In response to this request NSP performed a High Energy Line Break (HELB) analysis for the Monticello Plant and submitted the summary report in September 1973 (Reference 7.2). For this analysis, high energy lines were chosen as those lines whose service temperature exceeded 200°F and whose design pressure was greater than 275 psig. The basis for selection of break locations were those places where the greatest impact on safe shutdown would be experienced. These locations were evaluated for pipe whip, jet impingement, compartmental pressurization, and flooding.

In 1980, IE Bulletin 80-11, Masonary Wall Design (Reference 7.3), was issued to alert owners to the possible failure of block walls that could damage safety related equipment. The review performed for the Monticello Plant indicated that to protect the block walls in the Turbine Building pipe chase (IX/19C), installation of jet impingement shields would be required on the condensate piping, C4A-16-GB & C4B-16-GB. However, no shields were recommended for installation on the feedwater lines. In February 1986 it was noted by NSP that a compartmental pressurization of Fire Zone IX/19C could damage both Division (Div.) I cables and the essential Div. II motor control center (MCC 143). This information was reported to the NRC on April 4, 1986.

In April 1986 NUTECH Engineers Inc. was contracted by NSP to perform a review of the scope and accuracy of the 1973 analysis to determine if additional problems, similar to those found in Fire Zone IX/19C exist. The review covered those systems previously identified in the original report as containing high energy fluid.

3.0 EVALUATION APPROACH AND METHODOLOGY

This report provides a thorough review of those systems identified in the 1973 HELB analysis as having high energy lines. Individual high energy lines and the associated break locations were identified, and the effects on Safe Shutdown (SSD) systems, components, and structures from the postulated breaks were evaluated. The criteria identified in the December 18, 1972 Giambusso letter as clarified by Standard Review Plan (SRP) 3.6.1 (Reference 7.4), were utilized as the basis for the determination of the high energy lines, break locations, and the evaluation of effects on SSD equipment.

This section describes the criteria, assumptions and methodology used to select the high energy lines, determine the break locations, and evaluate the effects of the high energy breaks on SSD capability.

3.1 General Criteria

The evaluation requirements for high energy line breaks outside of the containment are contained in Criteria No. 4 of the General Design Criteria listed in Appendix A of 10 CFR Part 50 and in the Giambusso letter (Reference 7.1) . These criteria require that systems and components required to ensure a cold shutdown of the plant be capable of withstanding all of the expected conditions resulting from high energy line breaks outside of the containment including pipe whip, jet impingement and environmental effects. The application of these criteria is restricted to those systems, structures, and components required to bring the plant to, and maintain a cold shutdown condition.

3.2 Assumptions

The following assumptions were made during the HELB evaluation:

3.2.1 A postulated break was assumed to occur during normal steady state operating conditions at 100% of rated power (Criterion B.3.a -Reference 7.4).

3.2.2 Loss of offsite power concurrent with the line break was assumed, unless it was more conservative to assume the availability of offsite power, as would be the case for a break in a feedwater line (Criterion B.3.b.1 - Reference 7.4).

3.2.3 No fires or other simultaneous line breaks or accidents were considered in the evaluation of the data (Criterion B.3.a - Reference 7.4).

3.2.4 A single active component failure was assumed in systems used to mitigate consequences of the postulated piping failure and to shut down the reactor. The single active component failure is assumed to occur in addition to the postulated piping failure and any direct consequences of the piping failure, such as unit trip and loss of offsite power (Criterion 11c _ Reference 7.1 and as clarified in Criterion B.3.b.2 of Reference 7.4)

3.2.5 Circumferential breaks were assumed to result in complete pipe severence with full separation of the two severed pipe ends (i.e., guillotine). The break was assumed to be perpendicular to the longitudinal axis of the pipe. Circumferential breaks postulated at fittings were assumed to be at the fitting-to-pipe weld(s) (Footnote 9 to Criterion 3 -Reference 7.1).

3.2.6 Longitudinal breaks were assumed to be oriented parallel to the longitudinal axis of the pipe. Longitudinal breaks postulated at fittings were assumed to be at the center of the fitting (Footnote 8 to Criterion 3 - Reference 7.1).

3.2.7 Seismic Category I Systems, for which stress data was not available to determine break locations, was treated as a seismic Category II system (Criterion B.3.a.2 - Reference 7.5).

3.3 Identification of High Energy Piping

High energy (HE) piping was assumed to be all piping with a normal operating temperature exceeding $200^{\circ}F$ and the normal operating pressure exceeding 275 psig (References 7.6). If the actual normal operating conditions were not known, the design conditions were conservatively substituted. These criteria were applied to the Primary Steam, RCIC (steam), HPCI (steam), RWCU, Condensate Feedwater, and high energy sampling and instrument sensing systems. The Piping and Instrumenta-

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tion Diagrams and the Piping Line Designation Tables were utilized to generate the list of high energy lines.

3.4 Selection of Break Locations

Break locations were postulated for all HE Piping with a nominal diameter greater than 1 inch (Criterion 3 -Reference 7.1), and a normal operation time exceeding 2% of the total unit operation time (Criterion B.2.e -Reference 7.5) in accordance with the following criteria:

3.4.1 Breaks were postulated for each Seismic Category I HE Line as follows:

(1) At the terminal ends of the pressurized sections of the run (Criterion 2.b.1 -Reference 7.1);

(2) At any intermediate location where normal operating and seismic stresses exceeds $0.8(S_h + S_A)$ or $.8(S_A)$ (Criterion 2.b.2 -Reference 7.1) ;

(3) Not less than two (2) total intermediate locations were selected on the basis of highest relative stress if line stresses were below the criteria specified in Item (2). When choosing postulated break location on this basis, if the stresses differed by less than ten percent at the two highest stressed locations and their locations were not separated by a change in the direction of the piping run, then additional points were selected until a minimum of two locations were identified which were separated by a change in direction of the piping run (Section 5.2.c - Reference 7.7).

(4) If the piping run had calculated stresses lower than the limits established by Item 2 everywhere between its terminal ends and had only one change of direction, only one intermediate rupture location was postulated. Intermediate rupture locations were not postulated on short piping runs (less than 20 pipe diameters in length) with calculated stresses lower

than the limit of Item 2 everywhere between terminal ends and only one change of direction (Section 5.2.C - Reference 7.7).

3.4.2 Break locations postulated for each Seismic Category II (i.e., non-seismic)high energy line were selected as follows (Criterion B.1.d.2 - Reference 7.5):

(1) At the terminal ends of the pressurized sections of the run;

(2) At each intermediate location of potential high stress or fatigue such as pipe fittings, valves, flanges and welded on attachments.

3.4.3 Critical cracks were postulated to occur in piping carrying high energy fluids routed in the vicinity of systems required for safe shutdown of the unit. The critical crack size was taken to be 1/2 the pipe diameter in length and 1/2 the wall thickness in width (Criterion 2 -Reference 7.1).

3.5 Break Types

Circumferential breaks were postulated and evaluated at terminal ends. In addition, circumferential breaks were postulated at intermediate break locations on HE piping with a diameter exceeding 1 inch. Longitudinal breaks were postulated at all intermediate locations on HE piping with a nominal diameter exceeding 4 inches (Criterion 3 - Reference 7.1).

3.6 Site Survey Method and Approach

Preparation for the site surveys included the generation of a HELB Table identifying, by system, all high energy lines. Piping drawing plans and sections were color coded to highlight the high energy lines. Fire area boundaries were then superimposed on these drawings using the Monticello Nuclear Generating Plant Updated Fire Hazards Analysis (Reference 7.8) to define site survey areas. Fire Area/Zones were used, so that a unique designation could be provided for each compartment inspected. The surveys were conducted by compartment or fire area for all systems with high energy lines. HELB Data was recorded by survey team members for each HE line on Data Sheets and compiled into a survey book.

3.7 Evaluation of High Energy Line Break Data

The data obtained during the plant surveys was used to evaluate the hazards of pipe whip, compartment flooding, compartmental overpressurization, environmental conditions and jet impingement on those systems, components, and structures identified as required for safe shutdown. The evaluation techniques for each of the hazards identified was performed by applying the following criteria:

3.7.1 Pipe Whip Effects

For each postulated break, the effects of pipe whip were evaluated. Pipe whip movement was assumed to occur about a plastic hinge located at a point or points selected using the following guidelines (Reference 7.4):

(1) The pipe run terminal point.

(2) The nearest pipe fitting which will experience high bending moment. Generally this will occur at the second elbow from the break point.

(3) A pipe whip restraint.

(4) Any structure or equipment which can reasonably be expected to restrain the movement of the pipe (e.g., concrete wall or pressure vessel).

(5) Ordinary pipe supports were considered ineffective during whip.

Circumferential breaks were assumed to cause pipe whip about the plastic hinge in a plane defined by the pipe geometry and in the direction of jet reaction (Criterion B.3.a.5 -Reference 7.5) . Longitudinal breaks were assumed to cause pipe whip movement in any direction normal to the longitudinal axis of the pipe at the point of the break (Footnote 8 to Criterion 3 - Reference 7.1) . Additionally, it was assumed that the geometry of the pipe segment between the selected hinges remained unchanged throughout the pipe whip path.

The area of influence of pipe whip was considered to be the worst case area determined applying the criteria identified above. In general, circumferential breaks cause the worst case pipe whip conditions.

In evaluating the effect of pipe whip on safe shutdown components and structures, the following criteria was used:

(1) The energy level in a whipping pipe was considered to be insufficient to rupture an impacted pipe of: A) Equal or greater nominal pipe size and B) Equal or heavier wall thickness (Footnote 1 to Criterion 1.d -Reference 7.1).

(2) Impacted pipe of lesser nominal pipe size or lighter wall thickness was assumed to rupture or require further evaluation (Footnote 1 to Criterion 1.d - Reference 7.1).

(3) All electrical cables in cable trays, conduit, or other raceways impacted by a whipping pipe were assumed to be severed or require further evaluation. Other electrical and I & C components were assumed to fail or require further evaluation (Reference 7.2).

(4) Structural components impacted by a whipping pipe were assumed to fail or require further evaluation (Reference 7.2).

3.7.2 Jet Impingement Effects

For each postulated break, the effects of jet impingement were evaluated. The criteria used to evaluate these effects were as follows:

(1) All jets were assumed to be influenced by gravity (Section 7.1 - Reference 7.7). However, because of the jet fluid velocities and the relative proximity of the targets, jet travel in a straight line was assumed.

(2) Jets from a circumferential break were assumed to sweep the arc traveled during the whip (Section 8.1 - Reference 7.7).

(3) A longitudinal break can occur at any azimuth location of the pipe circumference (Footnote 8 to Criterion 3 -Reference 7.1).

(4) A jet discharging saturated steam, saturated water, or subcooled water, with a fluid temperature greater than the saturation temperature at the surrounding environmental pressure was assumed to expand in a 10^{o} half-angle cone. Subcooled water jets with a fluid temperature less than the saturation temperature at the surrounding environmental pressure are characterized by a constant diameter jet (Section 7.2 -Reference 7.7) . The initial area of the jet for a circumferential break was assumed equal to the effective flow area of the pipe at the break location (Section 7.2 - Reference 7.7).

(5) A postulated jet was considered effective until it strikes a solid barrier. All safe shutdown components, pipes, and structures impacted by the jet were considered incapable of performing their safe shutdown function or requiring further evaluation (Section 7.2 -Reference 7.7).

(6) Where the jet must travel a significant distance to impact safe shutdown components or structures, simplified calculations may be used to demonstrate that the jet impingement forces are negligable.

3.7.3 Flooding Effects

In each area or compartment where breaks were postulated, the potential adverse effects from flooding were identified using the following criteria (Section 10 - Reference 7.7):

(1) Vulnerability of safe shutdown equipment due to flooding because of location and configuration.

(2) Potential rupture sizes and the available quantity of water.

(3) Presence of floor drains or doorways which could provide drainage.

(4) Flooding protection in the form of dams or sumps with pumps.

3.7.4 Environmental Effects and Compartment Pressurization

A review was conducted to determine if the safe shutdown components in compartments or areas in which breaks have been postulated are qualified for worst case environmental conditions including the effects of compartment pressurization. Attention was focused on closed compartments which contained high energy lines. The bounding or largest available energy line break for each compartment was compared with that assumed in the Monticello Nuclear Generating Plant Environmental Effects Report (Reference 7.9) to establish limiting condition for the compartment. If other lines appeared to produce a more severe condition, this information was noted.

4.0 SAFE SHUTDOWN SYSTEMS REQUIREMENTS

Safe shutdown was assumed to be achieved by meeting the specific performance goals of the plant. These performance goals were used to establish the equipment required for safe shutdown. The goals include:

- Bringing the reactor core to a subcritical condition.
- Restoring and maintaining reactor vessel water level above the active fuel level precluding fuel cladding failure.
- Preventing overpressurization of the reactor vessel.
- Removal of decay heat generated in the core.

A review of specific plant documents was conducted, to obtain information on the systems required for safe shutdown and the success paths to safe shutdown. This information was gathered from the Updated Safety Analysis Report (Reference 7.10), the 1973 HELB Report (Reference 7.2), and Fire Protection Safe Shutdown Analysis Report (Reference 7.11). From these documents a list of safe shutdown systems was generated for use in identifing potential targets of pipe whip and jet impingement, equipment subject to flooding and environmental concerns, and for determining the safe shutdown path for each postulated break. A description of each system follows:

4.1 Safe Shutdown Systems Summary

The following list identifies those systems, for which credit can be used to bring the unit to safe shutdown:

(1) Reactor Protection System (RPS), Control Rod Drive System, and Control Rods

(2) High Pressure Coolant Injection (HPCI)

(3) Safety/Relief Valves (S/RV's)

(4) Reactor Core Isolation Cooling (RCIC)

(5) Residual Heat Removal (RHR) including:

- Low Pressure Coolant Injection (LPCI) Mode
- Shutdown Cooling (SDC) Mode
- Supression Pool Cooling (SPC) Mode

(6) Core Spray (CS)

(7) RHR Service Water

(8) Shutdown Instrumentation

(9) Auxiliary Support

- Onsite Power and Distribution System
- Emergency Service Water
- Diesel Generator Auxiliary Systems
- D-C Power Systems
- HVAC

4.2 Reactor Protection System, Control Rod Drive System, and Control Rods

The RPS, Control Rod Drive System and Control Rods constitute the Reactivity Control System. Shutdown of the fission chain reaction process is accomplished by insertion of the control rods into the reactor core. With the rods inserted, the reactor is in a subcritical condition. Control Rod insertion into the core is actuated by the RPS, and the Control Rod Drive System provides the motive force to insert the control rods. The Control Rod Drive Pumps can also be used to maintain the reactor vessel water level, while decay heat is removed by the SDC Mode of the RHR System. The Reactivity Control System meets the requirements of the first performance goal.

4.3 High Pressure Coolant Injection (HPCI)

The HPCI system is the high pressure, low flow water injection system whose function is to restore and maintain reactor vessel water and depressurize the reactor. The HPCI System is automatically initiated on reactor low-low water level, and the HPCI System can continue operation down to reactor vessel pressures lower than the shutoff heads of the CS and LPCI pumps. The HPCI pump is powered by means of a steam driven turbine with steam supplied from the reactor vessel and is controlled by Div. II 250V-DC power. The preferred source of water for the HPCI is the Condensate Storage Tank and the safety-related source is the suppression pool. The HPCI system meets the performance goals of both inventory control and depressurization.

4.4 Safety/Relief Valves (S/RV's)

The S/RV's are designed to be capable of remote manual actuation to reduce reactor vessel pressure to enable the low pressure core cooling systems (LPCI and Core Spray) to function. The S/RV's include the valves, air accumulators and associated circuitry for manual operation. For the safe shutdown system function, the valves are manually operated to depressurize the reactor vessel by transferring steam to the suppression pool. Depressurization of the reactor vessel enables these low pressure core cooling systems to effectively function. The S/RVs require only D-C power from the plant battery system for operation. The source of air for valve operation is from the air accumulator on each valve. There are no required automatic system actuation signals for the safe shutdown system function.

Once the reactor vessel is depressurized, the S/RVs can be used as a flow path to remove decay heat from the reactor vessel, if the shutdown cooling line is not available. A loop is established with a CS or RHR pump taking suction from the Suppression Pool and returning the water to the Suppression Pool through the S/RVs.

4.5 Reactor Core Isolation Cooling

The RCIC is designed to restore and maintain reactor vessel water level. Like the HPCI System, the RCIC Pump is powered by a steam driven pump with steam supplied from the reactor vessel, operates on D-C power, is actuated on reactor low-low water level, is capable of supplying water to the reactor vessel over the pressure range from S/RV setpoints to below the shutoff head of the CS and LPCI pumps, and can take pump suction from either the Condensate Storage Tank or the Suppression Pool. The RCIC meets the performance goal of restoring and maintaining reactor vessel level.

4.6 Residual Heat Removal

The RHR System is designed to maintain reactor vessel water level and remove decay heat, after the reactor vessel has been depressurized. The LPCI Mode is used to restore and maintain reactor vessel water level. LPCI Pump suction is taken from the Suppression Pool, and the water is injected through the Recirculation System. The SDC Mode is used to remove decay heat from the reactor vessel. RHR Pump suction is taken from the recirculation piping. The water is pumped through the RHR Heat Exchanger, and the water is returned to the reactor vessel via the recirculation piping. The isolation valves in the recirc loop used for SDC water return to the reactor vessel are closed prior to establishing SDC flow to the recirc loop. The SPC Mode is used to remove decay heat from the Suppression Pool. In this mode Reactor Vessel level is maintained by a CS pump. In the SPC mode, RHR Pump suction is taken from the Suppression Pool. The water is pumped through the RHR Heat Exchanger and returned to the Suppression Pool.

4.7 Core Spray

The Core Spray System is used to maintain reactor coolant inventory and remove decay heat, after vessel depressurization. Reactor vessel coolant inventory is maintained by taking suction from the Suppression Pool and pumping the water through a ring header above the core in the reactor vessel. By opening the SR/V's and operating a CS pump, a loop can be established to remove decay heat from the reactor vessel and transfer the heat to the Suppression Pool. The decay heat is then removed from the Suppression Pool by SPC Mode of RHR.

4.8 RHR Service Water

The RHR Service Water System is used to transfer the decay heat from the RHR System to the river (i.e., Ultimate Heat Sink). River water is pumped through the tube side of the RHR Heat Exchanger and returned to the river. The RHR Service Water System meets the performance goal of decay heat removal.

4.9 Shutdown Instrumentation

The minimum required instrumentation to assure safe shutdown is as follows:

(1) Reactor Vessel Level Indication.

(2) Reactor Pressure Iindication.

(3) Suppression Pool Temperature Monitoring

(4) Suppression Pool Level Indication.

4.10 Auxiliary Support Systems

The following Auxiliary Support Systems are used to support those required for safe shutdown:

(1) The Onsite Power and Distribution System provides the necessary electrical power to run safe shutdown equipment.

(2) The Emergency Service Water System provides cooling water for the Diesel Generators, the RHR and CS Pumps, the RHR/CS pump room coolers, and the HPCI pump room cooler.

(3) The Diesel Generator Auxiliary Systems provide the necessary air for diesel start and the fuel oil for operation.

(4) The D-C Power Systems provide the control power for the HPCI, RCIC, and Diesel Generator Systems and supply the power for the S/RV's electrical operation.

(5) The HVAC Systems are used for HPCI Room cooling and CS/RHR Room cooling.

5.0 RESULTS OF EVALUATION

5.1 General Results

Table 5.1-1 provides a list of all high energy lines considered in the evaluation. In addition to identifying the high energy lines for each system considered, information is provided as to each line's location, terminal points, material, design conditions, and fire area(s) location(s). Also provided for each line is the compliance matrix, which identifies what further evaluation will be required to achieve compliance with the HELB requirements. Information for each HE line within a compartment was compiled by site inspection, and recorded in the survey book (Reference 7.12). The HE line list and compliance matrix reflects the information gathered from the site inspections in the compliance matrix. The systems identified as having high energy lines included:

(1) Main Steam

(2) Feedwater

(3) Condensate

(4) HPCI (steam)

(5) RCIC (steam)

(6) Reactor Water Clean-up (RWCU)

(7) Instrument and Sample Lines

All lines, which could be excluded based upon the 1" or less nominal pipe size or 2 % operating time exclusions, have been identified as excluded in the compliance matrix.

5.2 Single Active Failure

For each compartment containing HE lines, where safe shutdown equipment was adversely affected, a single active failure review was conducted to determine the single active failures that could inhibit safe shutdown of the unit. Determination of the component(s) subject to single active failure was based upon the definition provided in Appendix A to BTP ASB 3-1 (Reference 7.4) and Section 3.2 of Reference 7.6. The single failure review was conducted using the criteria given in Section 3.2.4 of this report. Those scenarios, for which a safe shutdown path could not be determined, are identified as items to be resolved in the conclusions.

The determination of postulated targets and lost systems revealed that for all HELB's within the plant, the RPS and the S/RV's were not affected. Also, there are no single active failures within these systems coincident with loss of offsite power (Sections 6.2.5 and 7.6.1 - Reference 7.10) which could prevent these systems from performing their intended safe shutdown function. Hence, the reactivity control and Reactor Vessel depressurization function could always be achieved.

The Reactor Vessel level control function is achieved by using either the HPCI or RCIC system before RPV depressurization and by using the Core Spray or LPCI Mode of RHR after RPV depressurization. If neither HPCI nor RCIC is affected by a HELB, then either system can achieve the level control function, assuming a single active failure in either system. If both HPCI and RCIC are lost through a HELB or HELB and single active failure, the RPV can be depressurized with the S/RV's and either the LPCI mode or CS can be used to restore and maintain RPV level. The LPCI and CS are redundant to each other, and each essential division has a CS and LPCI System. There are no single active failures in the CS, HPCI, RCIC, and the LPCI Mode of RHR which prevents the reactor vessel level function from being achieved.

The decay heat removal function is achieved by the HPCI System before RPV depressurization and by the SDC Mode of RHR or the CS, S/RV's, and SPC Mode of RHR after depressurization. If the HPCI System is not available as result of a HELB or a single active failure, the RPV can be depressurized using the S/RV's, and decay heat removal established by SDC Mode of RHR or the CS-S/RV's -SPC Mode of RHR. Each essential division contains a CS and RHR system. Hence, these systems provide redundancy for the decay heat removal SSD function on a systems basis. In addition, each RHR Loop is equipped with 2 pumps (only 1 of which is required for SPC or SDC Mode) and locked open suction valves to the suppression pool. Therefore, in either loop of RHR with a single active failure either the SDC Mode or the SPC Mode is available, and the decay heat removal function can be achieved. Therefore, no single active failure within these systems can prevent achievement of the decay heat removal SSD function.

In addition to these systems, the RHR Service Water System is required to remove decay heat from the RHR System and transfer the heat to the river. Each RHR Service Water System loop is equipped with duel pumps, and hence a single active failure of either pump does not affect the ability of the identified loop to function. The only non-redundant active component in either loop is the control valve which regulates RHR Service Water pressure. If this valve is the single active failure, sufficient time exists to manually open the valve or force the valve. This is because for the HELB's postulated either HPCI and RCIC are not affected or both essential power systems are not effected. In either case heat rejection to the river would not be required for several hours, which would be sufficient time to switch to the other essential division or open the affected valve.

For the Emergency Service Water (ESW) System, redundancy within each division does not exist, and the ESW lines are subject to HELB's. Further analysis of this system is given for each compartment where ESW is affected, and the non-acceptable results are identified in the conclusions.

The remainder of the auxiliary systems (i.e., D/G Auxiliary Systems, D-C Power Systems, and HVAC) can perform their function assuming a single active failure. The D/G Auxiliaries include the air start system, cooling system, the oil transfer system, and the electrical control system. Redundant air start valves exist, which are the only active components on the air start system. The cooling system has no active components except for the ESW pump, which was discussed previously. The oil day tank for the diesel generator contains enough fuel for the diesel to run for 8 hours (Section 8.4.1 - Reference 7.10). For a failure of the essential fuel oil transfer pump a gasoline driven portable pump exists to transfer fuel oil from the oil storage tank to either day tank. The electrical control power for the diesel generators is taken from the D-C power system. The only active components on this system are the breakers at the Essential MCC's. A single failure of one of these breakers would not prevent the functioning of the D-C power system. Likewise, for the entire 250V and 125V D-C power system, the only active components are the breakers. No single active failure can prevent the system from functioning, and no components other than the 125V D-C breakers on MCC's 133 and 143 are subject to HELB's.

The only instrumentation required for safe shutdown, which is affected by a HELB, are the Suppress Pool Temperature Monitoring (SPTMS) cables for Division II. No single active failure exists in the SPTMS by a HELB, any postulated single failure or loss of power to one of the essential divisions will not prevent the remaining instrumentation from performing their SSD function.

The remaining system is the essential power system consisting of the two diesel generators, 4KV switchgear, 480V load centers and the essential 480V MCC's. The only active components in this system subject to single active failure are the diesel generators and their respective D/G breaker to each 4KV bus. The other switchgear and breakers on the power distribution system do not change position and are not subject to an active failure. Because of the criteria used to determine whether paths to safe shutdown exist, a HELB , coincident with loss of offsite power and loss of one D/G as the single active failure will result in an unacceptable condition where safe shutdown cannot be achieved. Therefore, additional measures will be required to mitigate this single failure.

5.3 Evaluation By System

5.3.1 Main Steam System

The high energy lines for the Main Steam System are located in 3 compartments, the Main Steam Chase (II/2F) in the Reactor Building, the Condenser Area (X/12C) and the Steam Jet Air Ejector (SJAE) Room (X/12E) in the Turbine Building.

The high energy lines on the Main Steam System include the four main steam lines from the drywell penetrations to the high pressure equalization lines, the turbine bypass lines to the condenser, primary steam to SJAE line, and primary steam to Steam Seal System.

5.3.1.1 Main Steam Chase (II/2F)

The Main Steam Chase contains the four main steam lines and the associated drain piping. Pipe whip from the main steam lines is not considered a problem, since these lines are restrained at several locations within the compartment. Jet impingement from a break in a main steam line can affect the ceiling of the pipe chase, where the Div. II saftyrelated cables are located. Additionally, a jet impingement from line P2-18-ED or PS3-18-ED could impinge upon the HPCI injection line (TW3-12"-ED), the HPCI Steam Line (PS18-8-ED), Emergency Service Water Line (SW30B-3-HF), the RCIC Steam Line (PS17-3"-ED) and the RCIC injection line (FW5-4-ED). Flooding from the Main Steam System is not considered credible, since very little water would be produced. Critical cracks were not postulated, because the effects on safe shutdown equipment in the area from pipe whip and jet impingement envelope the effects of the postulated critical cracks.

The results of the evaluations concluded a safe shutdown of the unit can be accomplished for a break impinging on the ceiling, because the cables required to power the other essential division or transfer diesel power to Div. I are not located in the ceiling. Hence, the unit can be safely shutdown even if these Div. II cables are adversely affected. Further analysis will be required to confirm that safe shutdown can be achieved, if HPCI, RCIC, and one division of Emergency Service Water (Div. II) are damaged in this area, and loss of offsite power and a single active failure of the Div. I diesel generator is assumed.

5.3.1.2 <u>SJAE Room (X/12E)</u>

The main steam piping line (PS9-3-ED) is routed to the SJAE Room from the Condenser Area. No concern exists with respect to pipe whip and jet impingement, since there is no SSD equipment in the area. Flooding is also not a concern, because this line carries steam. A break in the line would produce very little water. Safe shutdown would not be affected from pipe whip, jet impingement, or possible flooding within this compartment. (See Section 5.4.8 for compartment pressurization effects.)

5.3.1.3 Condenser Area (X/12C)

The bulk of Main Steam piping is within the condenser area. Pipe whip from each of the main steam lines can affect either of the Emergency Service Water (ESW) Lines (SW30A-3"-HF & SW30B-3"-HF). A pipe whip of line PS4-18-ED could additionally damage RHR Service Water line (SW9-18-GF). The most critical break would be in the steam bypass line (PS7-10-ED). A whip from this line could damage both of the Emergency Service Water lines, power cables to the HPCI System, and cables of one division of the Suppression Pool Temperature Monitoring System (SPTMS). The pipe whip effects of the other primary steam piping within this compartment could not cause any damage to safe shutdown equipment.

For jet impingement, the worst case event would be a longitudinal break in the bypass steam line (PS7-10-ED), which could impinge on both of the ESW lines (SW30A-3-HF & SW30B-3-HF) and whose pipe reaction would damage the HPCI and SPTMS cables on the other side of the line. All other postulated Main Steam breaks could damage individual piping of safe shutdown systems; but loss of any one line would not result in loss of the safe shutdown capability because only one safety division would be affected, and safe shutdown can be achieved, assuming a single active failure (see Section 5.2).

Flooding is not a consideration since the high energy fluid is steam.

With the exception of losing both divisions of Emergency Service Water in the Condenser Area, all other breaks would damage only one division of redundant SSD systems. Additional review and possible modifications to the unit will be required to mitigate the con-sequences of the break on the steam bypass line (PS7-10-ED), that damages both divisions of ESW.

5.3.2 Condensate System

The high energy Condensate System lines are located in the following compartments:

(1) Condenser Area (X/12C)

(2) Turbine Building Pipe Chase Area (IX/19C)

(3) Reactor Feedwater Pump Area (IX/13B & IX/13C)

The HE Condensate System piping includes the main condensate lines (C4A-16-GB and C4B-16-GB), the Feedwater Pump minimum flow lines (C4A-2-EB and C4B-2-EB), and the condensate cross-tie (C7-16-GB).

Break locations were selected based upon a seismic analysis of the piping and break location criteria established for seismic Category I piping. The pipe runs extended from the terminal points on the third stage intermediate heaters to the suction nozzles of Reactor Feedwater Pumps. The intermediate break locations on line C4A-16-GB are in the Feedwater Pump Area (IX/13B), and the intermediate break locations for C4B-16-GB are in the Condenser Area (X/12C). Also, because the condensate cross-tie line (C7-16-GB) and Feedwater Pump minimum flow lines (C4A-2-EB and C4B-2-EB) are routed totally within the IX/13B fire zone, all break locations for these lines are in fire zone IX/13B.

5.3.2.1 Condenser Area (X/12C)

The Condenser Area contains both of the Condensate Lines (C4A-16-GB and & C4B-16-GB). Each line terminates at the respective inlet and outlet of the Third Stage Intermediate Heaters E-13A & B and at the outlet from Second Stage Intermediate Heaters E-12A & B. These terminal points and the two intermediate break locations on Line C4B-16-GB constitute all of the Condensate System break locations within this compartment.

Of the postulated break locations, only two represent a concern for Safe Shutdown equipment. A longitudinal break at one of the intermediate break locations on C4B-16-GB could impinge upon the Emergency Service Water Line (SW30B-3-HF). Also, a circumferential break at the inlet terminal point of line C4A-16-GB at Intermediate Heater E-13A could impinge on both Emergency Service Water Lines SW30A-3-HF and SW30B-3-HF. A loss of one of the Emergency Service Water lines would not impact Safe Shutdown. (See Section 5.3.1.3.) However, damage to both lines would result in an unacceptable condition and will require further analysis.

Flooding is not a concern because of the size of the Condenser Area. Critical cracks were not postulated, because pipe breaks were already postulated at the points in proximity to the SSD equipment.

5.3.2.2 Turbine Building Pipe Chase (IX/19C)

There were no break locations determined to be within this compartment.

5.3.2.3 Reactor Feedwater Pump Area (IX/13B & 13C)

All five of the HE condensate lines are present within this compartment. The break locations include the terminal end of each condensate line at the respective Feedwater Pump suction nozzle, the terminal ends of the cross-tie line (C7-16-GB) at the Condensate Lines, the terminal ends of the mini flow lines at the connections to the Feedwater Pumps, and the two intermediate break locations on Condensate Line C4A-16-GB.

MCC 133 could be exposed to jet impingement from longitudinal breaks on C7-16-GB and the intermediate break locations on C4A-16-GB. Moreover, the ceiling of this compartment (IX/13B) is subject to jet impingement from these break locations, which could cause damage to MCC 143 on the floor above, by allowing a harsh environment to the room above through a breach in the ceiling or directly from a missile created by failure of the floor.

Flooding is not a concern, since there is a flood cavity below MCC 133. Even if MCC 133 were to flood, safe shutdown could still be accomplished by using the Division I diesel power Div. II, assuming loss of MCC 133, loss of offsite power and single failure of the Div. II diesel generator. The same scenario can be applied to the loss of MCC 133 from jet impingement. Further analysis will be necessary to show that the ceiling to this compartment is not damaged by jet impingement from the condensate piping.

Critical cracks were not postulated because the adverse affects, postulated from line breaks would envelope the effects produced by the critical cracks.

5.3.3 Feedwater System

The HE Feedwater system piping (FW2A-14-EB and FB2B-14-EB) begins in Fire Area IX/13B at the discharge nozzle of feed pumps P-2A and P-2B. The two main Feedwater lines, FW2A-14 and FW2B-14, pass through IX/13C, up into IX/19C and then into the Condenser Area X1/12C. Before entering the Reactor Building Steam Chase II/2F, each Feedwater line is connected to its respective high pressure feedwater heaters (E-14 and E-15) in area X/30.

The two feedwater lines and Feedwater regulating station piping were seismically analyzed. Break locations were selected based upon the seismic analysis of the piping and the break location criteria established for seismic Category I piping. All four intermediate break locations for the feedwater system were identified in fire area/zone IX/13B, the Reactor Feed Pump Area at Elevation 911'. There were no break locations in the Turbine Building Pipe Chase (IX/19C). An additional break location was chosen in the Condenser Area (X/12C), as a result of the seismic analysis.

5.3.3.1 Fire Areas IX/13B and IX/13C, Reactor Feedwater Pump Area

Three break locations for each Feedwater line have been evaluated as follows:

FW2A-14 - Feedwater pump discharge nozzle, upstream weld on control valve CV6-12A, and downstream weld on the horizontal plane elbow on centerline elevation 924'-4".

FW2B-14 - Downstream weld on control valve CV6-12B, upstream weld on the vertical plane elbow at centerline elevation 922'-6", and Feedwater Pump P-2B discharge nozzle.

FW2-6 - Downstream weld on control valve CV-613 and weld on U-bend upstream of valve CV-613.

For the effects of pipe breaks in this location, refer to the previous discussion in Section 5.3.2.3, Condensate System.

5.3.3.2 <u>Fire Area IX/19C Turbine Building Pipe Chase</u>

There are no break locations determined to be within this compartment.

5.3.3.3 <u>Fire Area X/12C - Condenser Area</u>

The break location for Feedwater Line FW2A-14 is located at an elbow between column lines 6 and 7 at a centerline elevation of 934'-10". The Division II Emergency Service Water line (SW30B-3-3HF) is a jet impingement target, and the effects of a loss of this line have been discussed previously in connection with Main Steam (see Section 5.3.1) and Condensate System (see Section 5.3.2) line breaks. There are no pipe whip concerns in this are.

5.3.3.4 <u>Fire Area II/2F - Reactor Building Steam Chase</u>

Break locations in this area consist of the terminal ends at each primary containment penetration and one intermediate break loction on each feedwater line. For a break at the terminal point on line FW2A-14-ED

RCIC flow would be lost, because the RCIC injection line (FW5-4"-ED) connects upstream of the postulated break. Also, the intermediate break location for line FW2A-14-ED is at the weld to valve FW-94-1, which is also downstream of the RCIC injection point. For Feedwater line FW2B-14-ED, a break at the terminal end would cause the loss of the HPCI System, because the injection point is upstream of the break. However, HPCI flow is not affected for the postulated break at the intermediate location, because the break location is upstream of the HPCI injection point and the check valve on the feedwater line (FW2B-14-ED). No safe shutdown equipment can be adversely affected by pipe whip or jet impingement from any pipe break of the Feedwater System in the Reactor Building Steam Chase.

Flooding in the Reactor Building Steam Chase, as a result of a Feedwater line break, would cause the loss of both HPCI and RCIC, since the injection valves for these systems (MO-2068 and MO-2107) would be submerged above the motor operators until the door to the compartment failed. Water exiting the steam chase would flow along the floor of the Reactor Building 935' elevation, and some of the water would flow down to the compartment containing the Control Rod Drive Pumps. Also, some water would flow down to the HPCI pump room. Additional safe shutdown equipment would not be affected by the water exiting the steam chase, because no SSD equipment is located in the affected area.

Since the only SSD systems which would be adversely affected by Feedwater pipe breaks in the Reactor

Building Steam Chase are HPCI and RCIC, safe shutdown is not affected. Previous analysis has demonstrated that the unit can be safely shutdown with loss of both HPCI and RCIC (Reference 7.10).

5.3.3.5 Fire Area X/30 - Turbine Operating Floor

Feedwater lines FW2A and FW2B have break locations at the terminal ends of the inlets and outlets and the Feedwater heaters in this area. There are no safe shutdown components located in this area. Pipe whip or jet impingement targets are limited to the Turbine Building walls which would be postulated to fail.

5.3.4 High Pressure Coolant Injection System

The steam supply line (PS18-B-ED) to the HPCI Turbine begins at the drywell penetration located in the steam chase (Fire Zone II/2F). The steam supply line enters the torus area (IV/1F) and then the HPCI compartment area II/1E. This steam supply line is a high energy line from the drywell to the steam supply valve located on the HPCI turbine.

5.3.4.1 Fire Area II/2F - Reactor Building Steam Chase

Possible pipe whip targets include the Feedwater and Main Steam Lines which are assumed to be unaffected, because they are larger and thicker walled than the HPCI steam line.

Jet impingement targets include the ceiling through which Division II embedded conduits are routed. The effects of jet impingement on the ceiling is discussed in Section 5.3.1.1. Emergency Service Water Piping (SW30B-3-HF) and the RCIC Steam line (PS17-3-ED) can be damaged by jet impingement from a longitudinal line break. This results in loss of HPCI, RCIC and the Div. II ESW. Further evaluation will be required to assure a path to safe shutdown. Critical cracks were not postulated, because pipe breaks are already postulated at the locations in proximity to the SSD equipment.

Compartment flooding is not a consideration in this area. The HPCI steam line will be isolated on high flow, so that only minor flooding can occur.

5.3.4.2 Fire Area IV/1F - Torus Area

No high stress intermediate break locations are located for the HPCI Steam Line in the Torus Area.

5.3.4.3 Fire Area II/1E - HPCI Compartment

Only HPCI components are located in this area. The adjacent compartment (II/10), where pull boxes for the 4 KV power feeds to the Division II core spray and RHR pumps are located, have been qualified for the expected environmental conditions.

5.3.5 Reactor Core Isolation Cooling System

The identified high energy piping for RCIC System is the steam supply line (PS17-3-ED) to the RCIC turbine from the drywell penetration. This line begins in the steam chase (II/2F), runs down into the torus area (IV/1F) and into the RCIC compartment (III/1C).

5.3.5.1 Fire Area II/2F - Steam Chase

A pipe break on the RCIC steam line could impact the HPCI steam line (PS18-8-ED) or the ESW "B" line (SW30B-3-HF). The HPCI steam line is not affected, because it is larger and heavier schedule than the RCIC steam line. The effects of loss of the ESW line have been discussed in Section 5.3.1.3. Possible jet impingement targets include the ceiling which contains embedded Division II conduits and the ESW "B" line (SW30B-3-HF). The impact on safe shutdown from the loss of either of these systems is discussed in Section 5.3.1.3. The HPCI steam line is not affected because it is shielded by the RCIC injection line. Critical cracks were not postulated since the identified pipe break locations were those in proximity to the SSD equipment.

Compartment flooding will not effect any safe shutdown components because the RCIC Steam Line will isolate on high flow, and any resultant flooding will be minor in nature.

5.3.5.2 Fire Area IV/1F - Torus Area

Pipe whip targets include the "B" Emergency Service Water loop (SW30B-3-HF). (See Section 5.3.1.3 for discussion of loss of ESW.) There are no jet impingement targets in this compartment (Reference 7.12).

Compartment flooding is not a concern due to the small relative size of the RCIC steam supply line and the short duration before the break is isolated.

5.3.5.3 Fire Area III/1C - RCIC Compartment

Only RCIC components are located in this compartment are RCIC associated with the exception of a pull box for the core spray and RHR pumps 4 KV power feeds. An RCIC terminal end exists at this location, making the pull box a jet impingement target.

Loss of cables would disable Div. I RHR and CS. However, safe shutdown could be achieved by using Div. II RHR and CS pumps with power supplied by the Div. II D/G. For the worst single active failure, the Div. I D/G can be cross-tied to Div. II to run the RHR and CS pumps. The cables in the pull box are qualified for the compartment environment following a RCIC steam supply line break.

5.3.6 Reactor Water Cleanup - RWCU

The Reactor Water Cleanup high energy line begins at the drywell penetration and outboard RWCU isolation valve. This line supplies reactor water to the RWCU heat exchanger or the RWCU pumps. All high energy lines are in the RWCU compartment, except the return line which connects to RCIC and HPCI injection lines in the steam chase.

5.3.6.1 Fire Area II/3D - RWCU Compartment

Pipe whip targets consist of conduits which supply motive power to the Division II reactor sample line isolation valve, both Division II core spray outboard injection valves, an RHR containment spray valve and the Primary Containment Atmospheric Control isolation valves. Redundant Division I inboard containment isolation valves located inside of the primary containment are available to isolate the CS, RHR containment spray and reactor sample lines for safe shutdown. Therefore, loss of the above valves is not a concern. Both Primary Containment Atmospheric Control Isolation (PCAC) valves fail closed on loss of control cables.

Jet impingement in the RWCU compartment can target any of the above conduits. Additional jet impingement targets include the outboard RWCU isolation valve, both Division II Core Spray injection valves, RHR containment spray valve, Reactor Sample line and isolation valve and the Primary Containment Atmospheric Control Isolation valves. Redundant valves inside the containment or located outside this compartment, mitigate any concerns on loss of the above components with one exception: the loss of air supply to the PCAC isolation valves A02386 and A02387. These valves have air inflated seals which are required for leak tightness and are each supplied by an air line, check valve, and receiver tank located in this compartment. Additional jet impingement and system analysis will be required to confirm primary containment boundary integrity.

Compartment flooding is not a concern as there are no safe shutdown components located below the level of the compartment doors.

Sufficient leak detection instrumentation is available for isolation of the RWCU line by using the inboard isolation valve.

Outside the RWCU compartment in Fire Area II/3D are two Division II cable trays which are jet impingement targets. These same cables have been analyzed as part of Main Steam pipe break analysis in the Reactor Building Steam Chase (see Section 5.3.1.3). There are no pipe whip concerns in this area due to the distance to safe shutdown targets.

5.3.6.2 Fire Area V/3A - MG Set Room

Jet impingement targets are limited to the two power distribution panels for the RHR air compressors and the two wall mounted Containment Atmospheric Monitoring panels, which are not required for safe shutdown. No pipe whip targets are located in this compartment because the closest SSD component is farther than the pipe whip moment arm (Reference 7.11). Loss of both RHR air compressor power feeds is not a concern, since the air receivers are not affected by the jet, and even if compressed air was unavailable, the RHR valves could be opened manually. With the remainder of the SSD systems operable, approximately 8 hours would be available to open the valve.

5.3.6.3 Fire Area II/2C - Reactor Building Elev. 935

Two cable trays located near this line are pipe whip and jet impingement targets. These cable trays contain Division II cables for which redundant Division I cables are available in another location. A discussion of the effects on safe shutdown from the loss of these Div. II cables is given in Section 5.3.1.3.

5.3.6.4 Fire Area II/2F - Steam Chase

There are no pipe whip targets in this area. Any other concerns in this area are bounded by other larger pipes discussed in Sections 5.3.1, 5.3.3, and 5.3.4. Check valves at the RWCU injection points will prevent loss of both HPCI and RCIC flow from a break at either terminal end. Also, the check valves ensure the availability of both HPCI and RCIC for a break on the non-seismic portion of the RWCU line in the Steam Chase area.

5.3.7 Instrument Sensing Lines

The Instrument Sensing Lines from the primary cooling system and reactor vessel to the instrumentation represent high energy lines for that portion of the routing outside of the Primary Containment. Since all of these lines are 1 inch or smaller in nominal size, neither circumferential breaks nor longitudinal are required to be postulated. In addition, all lines are equiped with excess flow check valves, which would mitigate any break within a few seconds of the break occuring.

5.4 Evaluation By Compartment

This section addresses the effects of high energy line breaks in the various Reactor Building and Turbine Building compartments previously discussed. The evaluated effects include compartment pressurization and possible environmental qualification effects. For each compartment containing high energy lines, a re-evaluation was performed to determine that the bounding environmental effects and high energy lines previously identified in Reference 7.9 are still applicable. Worst case conditions were used to identify the mitigating features of each compartment. A comparison of results with the Monticello Nuclear Generating Plant Environmental Effects Due to Pipe Rupture report, Reference 7.9, was made to determine if additional analysis is required to support the environmental qualification process.

5.4.1 Reactor building Steam Chase Area II/2F

The high energy lines in this compartment consist of the following:

(1) Main Steam Lines (PS1 thru 4)

(2) Feedwater lines (FW2A & FW2B)

(3) HPCI Steam Supply (Line PS18)

(4) RCIC Steam Supply (Line PS17)

(5) RWCU Return Line (REW6)

(6) Bypass to Condenser (D4-6)

The bounding conditions are determined by the Main Steam lines and Feedwater lines. Blowout panels located near the ceiling of the steam chase will fail as designed venting steam or flashed feedwater up to the turbine operating deck. The environment at the operating deck has not been calculated in Reference 7.9 for this break location. An instrument rack in this area provides interlocks to the Reactor Protection System. The door from the Reactor Building to the steam chase is expected to fail open and the 935 foot elevationof the Reactor Building to be affected. The environmental effects in the steam chase are not changed from those documented in Reference 7.9. Hence, the HPCI and RCIC injection valves (MO-2068 and MO-2107, respectively), which are qualified for this environment, remain functional following a Main Steam, HPCI Steam or RCIC Steam line break in this compartment.

5.4.2 Open Area of Reactor Building 935' Elevation (II/2C)

The Reactor Water Cleanup return line REW6 is the critical break in this area. Environmental conditions for this area are based on, and bounded by a main steam line break in the steam chase. Therefore, no changes to the environmental qualifications are required, and no compartmental pressurization results.

5.4.3 Reactor Building 962' Elevation Area (II/3D)

The reactor water cleanup return line REW6 is the limiting environmental concern for line breakes in this area. No additional environmental or compartmental pressurization concerns relative to Reference 7.9 have been identified in this area.

5.4.4 MG Jet Room Area (V/3A)

The reactor water cleanup return line REW6 continues into this compartment at the south west corner. There are no environmental conditions identified for this area or pipe breaks assumed for this area in Reference 7.9.

This compartment will not pressurize appreciably due to the large volume relative to the break size and the louvers located on the east wall of the compartment.

5.4.5 Torus Area (IV/1F)

The HPCI and RCIC steam supply lines in this area are the only high energy lines present. The environmental conditions for this area were computed for the HPCI line in Reference 7.9 as the limiting condition. However, no HELB break locations in the torus area are identified for the HPCI steam supply line. Using the HPCI line break as a bounding condition for the environmental qualification is conservative, since RCIC is a smaller diameter line than HPCI.

5.4.6 HPCI Compartment (II/1E)

The environmental effects due to the HPCI steam supply line PS18 are documented in Reference 7.9 and are not changed for this analysis.

5.4.7 RCIC Compartment (III/1C)

The environmental conditions were not specifically calculated for this area but were considered to be the same as the HPCI room. The RCIC compartment vents to the torus area (IV/1F), which is bounded by the HPCI steam line break, and TIP drive room area (III/2A) where no safe shutdown components are located.

5.4.8 Turbine Building Condenser Area (X/12C)

The bounding environmental conditions in the condenser area are the Main Steam line and Feedwater line breaks. This area has not been specifically evaluated in Reference 7.8. However, the original HELB report (Reference 7.2) and NRC contact Docket No. 50-263 (Reference 7.13) give a figure of 1.4 psig for the peak pressure due to a Main Steam line break. A break on a Main Steam pipe will pressurize the compartment, and the steam will vent through hatches and openings for equipment to the turbine operating floor (Elevation 95'). The compartmental pressurization may also blowout the block walls provided for condenser tube removal. No safe shutdown components are located in the postulated path of the block wall. However, Division I cables are routed above this area in a cable vault separated by a catwalk and will be exposed to environmental conditions not yet computed. Environmental conditions for the turbine operating floor for a break in the condenser area have not been computed.

5.4.9 SJAE Area (X/12E)

The main steam line, PS9, in this compartment is the only high energy line to consider from the original HELB report. There are no safe shutdown components in this area and was not evaluated for environmental conditions in Reference 7.9. Compartment pressurization may cause the compartment door to fail and become a missile, which could target Load Center 103. Failure of this door could expose Load Center 103 and Bus 15, both safe shutdown equipment, to adverse temperatures and humidity. The redundant division power distribution equipment will not be effected. A fire barrier between the SJAE compartment and the condenser area located in the pipe chase will most likely fail before the compartment door thus relieving the pressure. Further analysis will be required to resolve this question.

5.4.10 Turbine Building Pipe Chase (IX/19C)

There are no break locations located within this compartment.

5.4.11 Reactor Feedwater Pump Area (IX/13B and IX/13C)

A postulated break in a Feedwater line is the bounding condition for this area. Since the original Reference 7.9 report, changes in pipe support have resulted in break location changes. However, the original conclusions about compartment pressurization and temperature are not changed.

5.4.12 Turbine Operating Floor (X/30)

The Feedwater lines terminate at the inlets and outlets of the high pressure Feedwater heaters in this area. The report (Reference 7.9) does not consider a break in this area. Therefore, no environmental conditions have been determined for breaks located in this area (X/30). An instrument rack located in the east end of the Turbine Operating floor provides interlocks to the Reactor Protection System for the first stage turbine pressure. The Reactor Protection System will still perform its safety function based on Reactor Vessel Low Water Level.

5.5 Table of System Effects

Table 5.5-1 shows the effect of specific high energy line breaks by compartment and system. This table includes the required auxiliary systems which are considered potential HELB targets. The meaning of the letter codes used in the table are as follows:

F - primary failure as a direct result of a line break.

A - system is unaffected by a line break and is available.

$\overline{A}$ - The system is unavailable due to the failure of a required function or component associated with another system.

DATE : 10-Jun-86

HIGH ENERGY LINE DESIGNATION (BECHTEL SYSTEMS)

SYSTEM	LINE DESIGNATION	P&ID(S)	DRAWING ZONE	TERMINAL PL PTS FROM	TERMINAL PL PTS TO	MATERIAL STD	DESIGN OR NORMAL CONDITIONS TEMP (F)	PRESS (PSI)	D/N	BREAK LOCATION(S)	APPENDIX R FIRE AREA(S)	HELB COMPLIANCE MATRIX: NO ANALYSIS REQUIRED	SYSTEMS ANALYSIS	PIPE RUPTURE	PRA	EXCLUSION(S): SM DIA	2%	LBB
PRIMARY STEAM	PS1-18-ED	115/102	C-6	X-7A	SV1	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	I,II			X				
PRIMARY STEAM	PS1-20-B	102	D-5	CV1	HPTURB	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	I	X						
PRIMARY STEAM	LT2-3/4-ED	115	C-5	PS1-18	MS-4	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	I,II					X		
PRIMARY STEAM	D115-1-ED	102	E-6	PS1-10	MS-12	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	II					X		
PRIMARY STEAM	D1-1 1/2-ED	102	E-6	PS1-10	D4-2	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	II	X						
PRIMARY STEAM	PS11-6-ED	102	E-6	PS1-18	PS7-10	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	I	X						
PRIMARY STEAM	PS7-10-ED	102	E-6	PS11-6	PS10-5	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	I			X				
PRIMARY STEAM	PS7-10-ED	102	D-6	PS7-10	CAP	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	I			X				
PRIMARY STEAM	PS7-8-ED	102	D-6	PS7-10	PS7-10	106 GR B	502	1110	D	SEE CALCULATION NSP-30-101	I	X						
PRIMARY STEAM	PS10-5-B	102/103	B-6	PS10-5	SSR	106 GR D	582	1110	D	SEE CALCULATION NSP-30-101	I	X						
PRIMARY STEAM	PS10-5-B	103	E-6	PS10-5	MO-1046	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	II,I	X						
PRIMARY STEAM	D4-2-ED	102	B-6	HDR	COND (18)	106 GR D	582	1110	D	SEE CALCULATION NSP-30-101	II	X						
PRIMARY STEAM	PS15-3-ED	115/102	B-6	X-8	D4-6	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	II	X						
PRIMARY STEAM	D4-6-ED	102	B-5	PS15-3	EI-A	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	I	X						
PRIMARY STEAM	PS21-4-ED	102	D-6	PS7-10	RO-1618	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	I	X						
PRIMARY STEAM	D23-1-ED	102	D-3	PS1-20	D23-2	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	I					X		
PRIMARY STEAM	PS2-18-ED	115/102	D-5	X-7B	SV2	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	I,II			X				
PRIMARY STEAM	PS2-20-B	102	D-5	CV2	HPTURB	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	I	X						
PRIMARY STEAM	LT3-3/4-ED	115	E-5	PS2-18	MS-4	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	II					X		
PRIMARY STEAM	D116-1-ED	102	E-6	PS2-10	MS-12	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	II					X		
PRIMARY STEAM	D2-1 1/2-ED	102	E-5	PS2-10	D4-2	106 GR B	502	1110	D	SEE CALCULATION NSP-30-101	II	X						
PRIMARY STEAM	PS12-6-ED	102	E-5	PS2-18	PS7-10	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	I	X						
PRIMARY STEAM	PS6-8-ED	102	B-5	BCV	COND (4)	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	I	X						
PRIMARY STEAM	PS8-8-ED	102	B-5	BCV	COND (4)	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	I	X						
PRIMARY STEAM	PS9-3-ED	102/104	E-5	PS2-10	RS1-3	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	I		X	X				
PRIMARY STEAM	D26-1-ED	102	D-5	MO-1739	D26-10	106 GR D	582	1110	D	SEE CALCULATION NSP-30-101	I					X		
PRIMARY STEAM	D23-1-ED	102	D-3	PS2-20	D23-2	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	I					X		
PRIMARY STEAM	PS3-18-ED	115/102	D-2	X-7C	SV3	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	I,II			X				
PRIMARY STEAM	LT4-3/4-ED	115	D-2	PS3-18	MS-4	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	II					X		
PRIMARY STEAM	PS3-20-B	102	D-5	CV3	HPTURB	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	I	X						
PRIMARY STEAM	D117-1-ED	102	E-6	PS3-10	MS-12	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	II					X		
PRIMARY STEAM	D23-1-ED	102	D-3	PS3-20	D23-2	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	I					X		
PRIMARY STEAM	D23-2-ED	102/104	D-3	HDR	COND (38)	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	I	X						
PRIMARY STEAM	RS1-3-6B	104	E-4	PS9-2	E-2A/2B	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	I		X	X				
PRIMARY STEAM	D23-1-ED	102	D-3	D23-2	PS4-20	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	I					X		
PRIMARY STEAM	PS4-18-ED	115/102	C-2	X-7D	SV4	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	I,II			X				
PRIMARY STEAM	LT5-3/4-ED	115	C-2	PS4-18	MS-4	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	II					X		
PRIMARY STEAM	PS4-20-B	102	D-5	CV4	HPTURB	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	I	X						
PRIMARY STEAM	D118-1-ED	102	D-5	PS4-10	MS-12	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	II					X		
PRIMARY STEAM	PS12-6-ED	102	E-5	PS2-18	PS7-10	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	I	X						
PRIMARY STEAM	PS13-6-ED	102	E-6	PS3-18	PS7-10	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	I	X						
PRIMARY STEAM	PS14-6-ED	102	E-6	PS4-18	PS7-10	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	I	X						
PRIMARY STEAM	D3-1 1/2-ED	102	E-6	PS3-10	D4-2	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	II	X						
PRIMARY STEAM	D4-1 1/2-ED	102	E-6	PS4-10	D4-2	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	II	X						
PRIMARY STEAM	D40-1 1/2-EBD	102	B-5	PS30-10	D4-6	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	I	X						
PRIMARY STEAM	PS30-18-EBD	102	E-5	PS1-18	PS4-18	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	I							
PRIMARY STEAM	INSTR.-1/2-?	102	D5	PS1-18	PAVGMAN	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	I					X		
PRIMARY STEAM	INSTR.-1/2-?	102	D5	PS2-18	PAVGMAN	106 GR B	592	1110	D	SEE CALCULATION NSP-30-101	I					X		
PRIMARY STEAM	INSTR.-1/2-?	102	D5	PS3-18	PAVGMAN	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	I					X		
PRIMARY STEAM	INSTR.-1/2-?	102	D5	PS4-18	PAVGMAN	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	I					X		
PRIMARY STEAM	PS9-2-ED	104	E-5	PS9-3	PS1-3	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	II	X						

nutech
ENGINEERS

DATE : 10-Jun-86

HIGH ENERGY LINE DESIGNATION (BECHTEL SYSTEMS)

SYSTEM	LINE DESIGNATION	P&ID(S)	DRAWING ZONE	TERMINAL PL PTS FROM	TO	MATERIAL STD	DESIGN OR NORMAL CONDITIONS TEMP (F)	PRESS (PSI)	D/N	BREAK LOCATION(S)	APPENDIX R FIRE AREA(S)	HELB COMPLIANCE MATRIX NO ANALYSIS REQUIRED	SYSTEMS ANALYSIS	PIPE RUPTURE	PRA	EXCLUSION(S) SN BIA	2%	LBB
FEEDWATER	VB4-1-DE	106	C-4	FW2B-14	FW-78-2	106 GR B	305	1550	D	SEE CALCULATION NSP-30-101	IX					X		
FEEDWATER	D119-1-DE	106	D-5	FW2A-14	FW-79-1	106 GR B	305	1550	D	SEE CALCULATION NSP-30-101	IX					X		
FEEDWATER	D120-1-DE	106	C-5	FW2B-14	FW-79-2	106 GR B	305	1550	D	SEE CALCULATION NSP-30-101	II					X		
FEEDWATER	D121-1-DE	106	D-5	FW2A-14	FW-81-1	106 GR B	305	1550	D	SEE CALCULATION NSP-30-101	IX					X		
FEEDWATER	D122-1-DE	106	C-5	FW2B-14	FW-81-2	106 GR D	305	1550	D	SEE CALCULATION NSP-30-101	IX					X		
FEEDWATER	V74-1-DE	106	E-6	FW2A-14	FW-84-1	106 GR B	333	1550	D	SEE CALCULATION NSP-30-101	X					X		
FEEDWATER	V75-1-DE	106	D-6	FW2B-14	FW-84-2	106 GR B	333	1550	D	SEE CALCULATION NSP-30-101	X					X		
FEEDWATER	V76-1-DE	106	E-6	FW2A-14	FW-90-1	106 GR B	400	1550	D	SEE CALCULATION NSP-30-101	X					X		
FEEDWATER	V77-1-DE	106	C-6	FW2D-14	FW-90-2	106 GR B	400	1550	D	SEE CALCULATION NSP-30-101	X					X		
FEEDWATER	D123-1-DE	106	E-6	FW2A-14	FW-86-1	106 GR B	333	1550	D	SEE CALCULATION NSP-30-101	X					X		
FEEDWATER	D124-1-DE	106	D-6	FW2B-14	FW-86-2	106 GR B	333	1550	D	SEE CALCULATION NSP-30-101	X					X		
FEEDWATER	LT6-3/4-EB	115	A-3	FW2A-14	FW-93-1	106 GR B	400	1250	D	SEE CALCULATION NSP-30-101	II					X		
FEEDWATER	LT8-3/4-ED	115	A-3	FW2A-14	FW-96-1	106 GR B	400	1250	D	SEE CALCULATION NSP-30-101	II					X		
FEEDWATER	LT7-3/4-ED	115	A-4	FW2B-14	FW-93-2	106 GR B	400	1250	D	SEE CALCULATION NSP-30-101	II					X		
FEEDWATER	LT9-3/4-ED	115	A-4	FW2B-14	FW-96-2	106 GR B	400	1250	D	SEE CALCULATION NSP-30-101	II					X		
FEEDWATER	RV38-1-DE	106	C-6	FW2D-14	RV1147	53B OR 106B	400	1550	D	SEE CALCULATION NSP-30-101	X					X		
FEEDWATER	RV37-1-DE	106	E-6	FW2A-14	RV1146	53B OR 106B	400	1550	D	SEE CALCULATION NSP-30-101	X					X		
FEEDWATER	V123-1-DE	106	E-6	FW2A-14	FW162-1	106 GR D	400	1550	D	SEE CALCULATION NSP-30-101	X					X		
FEEDWATER	V125-1-DE	106	E-6	FW2A-14	FW160-1	106 GR B	333	1550	D	SEE CALCULATION NSP-30-101	X					X		
FEEDWATER	V127-1-DE	106	E-6	FW2A-14	FW156-1	106 GR D	333	1550	D	SEE CALCULATION NSP-30-101	X					X		
FEEDWATER	V129-1-DE	106	D-6	FW2A-14	FW154-1	106 GR B	305	1550	D	SEE CALCULATION NSP-30-101	X					X		
FEEDWATER	V124-1-DE	106	D-6	FW2B-14	FW162-2	106 GR D	400	1550	D	SEE CALCULATION NSP-30-101	X					X		
FEEDWATER	V126-1-DE	106	D-6	FW2B-14	FW160-2	106 GR B	333	1550	D	SEE CALCULATION NSP-30-101	X					X		
FEEDWATER	V128-1-DE	106	D-6	FW2B-14	FW156-2	106 GR B	333	1550	D	SEE CALCULATION NSP-30-101	X					X		
FEEDWATER	V130-1-DE	106	A-6	FW2D-14	FW154-2	106 GR D	305	1550	D	SEE CALCULATION NSP-30-101	X					X		
FEEDWATER	D200-1-DE	106	D-5	FW2A-14	FW-165-1	106 GR B	305	1550	D	SEE CALCULATION NSP-30-101	X					X		
FEEDWATER	D201-1-DE	106	A-5	FW2B-14	FW-165-2	106 GR B	305	1550	D	SEE CALCULATION NSP-30-101	X					X		
CONDENSATE	C4A-2-EB	106	E-3	P-2A	C4A	106 GR B	305	1550	0	SEE CALCULATION NSP-30-101	IX	X						
CONDENSATE	C4B-2-EB	106	C-3	P-2B	C4B	106 GR B	305	1550	D	SEE CALCULATION NSP-30-101	IX	X						
CONDENSATE	C4A-16-GB	105/106	D-3	E-12A	P2A	106 GR B	302/230	434	D	SEE CALCULATION NSP-30-101	IX,X			X				
CONDENSATE	C4D-16-GB	105/106	C-3	E-12B	P2B	106 GR B	302/230	434	D	SEE CALCULATION NSP-30-101	II,X			X				
CONDENSATE	C7-16-GB	106	D-3	C4A-16	C4B-16	106 GR D	302	434	D	SEE CALCULATION HSP-30-101	IX	X						
CONDENSATE	V131-3/4-DE	106	D-3	C4A-16	RV1128	106 GR B	302	434	D	SEE CALCULATION NSP-30-101	IX					X		
CONDENSATE	VB5-1-DE	106	D-3	C4A-16	FW-62-1	106 GR D	302	434	D	SEE CALCULATION NSP-30-101	II					X		
CONDENSATE	D202-1-DE	106	D-2	C4A-16	FW-60-1	106 GR D	302	434	D	SEE CALCULATION NSP-30-101	II					X		
CONDENSATE	VB6-1-DE	106	D-2	C4A-16	FW-59-1	106 GR B	302	434	D	SEE CALCULATION NSP-30-101	IX					X		
CONDENSATE	V132-1-DE	106	D-2	C4A-16	FW-57-1	106 GR B	302	434	D	SEE CALCULATION NSP-30-101	X					X		
CONDENSATE	V134-1-DE	106	D-2	C4A-16	FW-167-1	106 GR B	302	434	D	SEE CALCULATION NSP-30-101	X					X		
CONDENSATE	V135-1-DE	106	D-2	C4A-16	FW-166-1	106 GR B	230	434	D	SEE CALCULATION NSP-30-101	X					X		
CONDENSATE	D204-1-DE	106	D-2	C4A-16	FW-54-1	106 GR B	230	434	D	SEE CALCULATION NSP-30-101	X					X		
CONDENSATE	V138-3/4-DE	106	C-3	C4B-16	RV1129	106 GR B	302	434	D	SEE CALCULATION NSP-30-101	IX					X		
CONDENSATE	VB7-1-DE	106	C-3	C4B-16	FW-62-2	106 GR B	302	434	D	SEE CALCULATION NSP-30-101	IX					X		
CONDENSATE	D203-1-DE	106	C-3	C4B-16	FW-60-2	106 GR B	302	434	D	SEE CALCULATION NSP-30-101	IX					X		
CONDENSATE	VB8-1-DE	106	C-2	C4B-16	FW-59-2	106 GR B	302	434	D	SEE CALCULATION NSP-30-101	IX					X		
CONDENSATE	V133-3/4-DE	106	C-2	C4B-16	FW-57-2	106 GR B	302	434	D	SEE CALCULATION NSP-30-101	X					X		
CONDENSATE	V136-1-DE	106	C-2	C4B-16	FW-167-2	106 GR B	302	434	D	SEE CALCULATION NSP-30-101	X					X		
CONDENSATE	V137-1-DE	106	C-2	C4B-16	FW-166-2	106 GR B	230	434	D	SEE CALCULATION NSP-30-101	X					X		
CONDENSATE	D205-1-DE	106	C-2	C4B-16	FW-54-2	106 GR B	230	434	D	SEE CALCULATION NSP-30-101	X					X		
CONDENSATE	V140-1-6BD	106	E-2	E13-A	FW-56-1	106 GR B	302	434	D	SEE CALCULATION NSP-30-101	X					X		
CONDENSATE	V141-1-6DB	106	C-2	E13-B	FW-56-2	106 GR B	302	434	D	SEE CALCULATION NSP-30-101	X					X		
CONDENSATE	V142-1-6DB	106	E-2	E13-A	FW-148-1	106 GR B	[illegible]	434	D	SEE CALCULATION NSP-30-101	X					X		

DATE : 10-Jun-86

HIGH ENERGY LINE DESIGNATION (BECHTEL SYSTEMS)

SYSTEM	LINE DESIGNATION	P&ID(S)	DRAWING ZONE	TERMINAL PL PTS FROM	TERMINAL PL PTS TO	MATERIAL STD	DESIGN OR NORMAL CONDITIONS TEMP (F)	PRESS (PSI)	D/N	BREAK LOCATION(S)	APPENDIX R FIRE AREA(S)	HELB COMPLIANCE MATRIX: NO ANALYSIS REQUIRED	SYSTEMS ANALYSIS	PIPE RUPTURE	PRA	EXCLUSION(S): 5N DIA	2%	LBB
CONDENSATE	V143-1-6DB	106	C-2	E13-B	FW-148-2	106 GR B	302	434	D	SEE CALCULATION NSP-30-101	X					X		
CONDENSATE	V144-1-6DB	106	E-2	E13-A	RV1112	106 GR B	302	434	D	SEE CALCULATION NSP-30-101	X					X		
CONDENSATE	V145-1-6DB	106	C-2	E13-N	RV1113	106 GR B	302	434	D	SEE CALCULATION NSP-30-101	X					X		
CONDENSATE	V146-1-6DB	106	E-2	E13-A	FW-147-1	106 GR B	302	434	D	SEE CALCULATION NSP-30-101	X					X		
CONDENSATE	V147-1-6DB	106	C-2	E13-B	FW-147-2	106 GR B	302	434	D	SEE CALCULATION NSP-30-101	X					X		
CONDENSATE	V148-1-6DB	106	E-2	E13-A	FW-55-1	106 GR B	302	434	D	SEE CALCULATION NSP-30-101	X					X		
CONDENSATE	V149-1-6DB	106	C-2	E13-B	FW-55-2	106 GR B	302	434	D	SEE CALCULATION NSP-30-101	X					X		
RWCU	REW3-4-ED	128	C-7	X-14	MO-2398	106 GR B	570	1136	D	SEE CALCULATION NSP-30-101	II	X						
RWCU	REW3-4-DB/DBD	128	C-6	MO-2398	HT E201A	106 GR B	570	1136	D	SEE CALCULATION NSP-30-101	II			X				
RWCU	REW3-4-DB/DBD	128	B-5	HT E201C	HT E202A	106 GR B	570	1136	D	SEE CALCULATION NSP-30-101	II	X						
RWCU	REW3-4-ED/DBD	128	C-4	HT E202B	REW3-3	106 GR B	570	1136	D	SEE CALCULATION NSP-30-101	II	X						
RWCU	REW3-4-DBB	128	C-4	MO-2398	REW3-3	106 GR B	570	1317	D	SEE CALCULATION NSP-30-101	II			X				
RWCU	REW3-4-DBD	128	C-6	RC-102	RC-101	106 GR B	570	1317	D	SEE CALCULATION NSP-30-101	II	X						
RWCU	REW3-3-ED	128	B-4	REW3-4DBD	P204 A/B	106 GR B	570	1136	D	SEE CALCULATION NSP-30-101	II	X						
RWCU	REW6-3-DBD	128	D-5	HT E201A	MO-2399	106 GR B	570	1317	D	SEE CALCULATION NSP-30-101	II,V	X						
RWCU	REW6-3-DC	128	D-6	MO-2399	RC-6	312/376TP304	570	1317	D	SEE CALCULATION NSP-30-101	II,V	X						
RWCU	LT1D-3/4-DC	128	D-6	REW6-3-DC	RC-34	312/376TP304	570	1317	D	SEE CALCULATION NSP-30-101	II					X		
RWCU	REW6-3-DB	128	D-6	RC-6	RC-7-1/7-2	106 GR B	570	1317	D	SEE CALCULATION NSP-30-101	II	X						
RWCU	REW6-3-ED	128	D-7	RC-7-1	TW3-12-ED	106 GR B	400	1250	D	SEE CALCULATION NSP-30-101	II	X						
RWCU	REW6-3-DB	128	D-7/A-3	RC-7-2	FW5-4	106 GR B	400	1250	D	SEE CALCULATION NSP-30-101	II	X						
RWCU	D175-1-DB	128	C-4	REW3-4	RC-77	106 GR B	570	1317	D	SEE CALCULATION NSP-30-101	II					X		
RWCU	LT17-3/4-ED	128	C-7	REW3-4	RC-36	106 GR B	570	1136	D	SEE CALCULATION NSP-30-101	II					X		
RWCU	RV67-3/4-DDB	128	C-5	E-201C	RV-67	106 GR N	570	1317	D	SEE CALCULATION NSP-30-101	II					X		
RWCU	RV6D-3/4-DBB	125	D-6	E-201A	RV-68	106 GR B	570	1317	D	SEE CALCULATION NSP-30-101	II					X		
RWCU	V111-1-DB	128	C-6	REW3-4-DB	RC-11	106 GR B	150 ?	1317	D	SEE CALCULATION NSP-30-101	II					X		
RWCU	V139-3/4-DBD/E	128	C-6	REW3-4-ED	RC-75	106 GR B	570	1317	D	SEE CALCULATION NSP-30-101	II					X		
RWCU	V130-1-DBD	128	D-5	REW6-3DDB	RC-15	106 GR D	570	1317	D	SEE CALCULATION NSP-30-101	II					X		
RWCU	REW3-1-DBB	128	C-6	E-201A	E-201C	106 GR B	570	1317	D	SEE CALCULATION NSP-30-101	II					X		
RWCU	REW4-1-DBD	128	C-5	E-201C	E-201A	106 GR B	570	1317	D	SEE CALCULATION NSP-30-101	II					X		
RWCU	V110-3/4-DBB	128	D-5	REW6-3	RC-40	106 GR B	570	1317	D	SEE CALCULATION NSP-30-101	II					X		
RWCU	D206-1-DBD	128	C-5	REW3-4	RC-20	106 GR B	570	1317	D	SEE CALCULATION NSP-30-101	II					X		
RWCU	V117-3/4-DB	128	D-4	REW3-4	RC-73	106 GR B	150 ?	1317	D	SEE CALCULATION NSP-30-101	II					X		
RWCU	D207-3/4-DBD	128	C-4	REW3-4	RC-26	106 GR B	570	1317	D	SEE CALCULATION NSP-30-101	II					X		
RWCU	V109-1-DBB	128	D-5	E-201A	RC-28	106 GR B	150 ?	1317	D	SEE CALCULATION NSP-30-101	II					X		
RWCU	D162-1-DBD	128	C-5	E-201C	RC-30	106 GR B	150 ?	1317	D	SEE CALCULATION NSP-30-101	II					X		
RWCU	V102-1-DB	128	D-4	E-202A	V103-1	106 GR B	570	1317	D	SEE CALCULATION NSP-30-101	II					X		
RWCU	V103-1-DB	128	D-4	E-202A	RC-22	106 GR B	570	1317	D	SEE CALCULATION NSP-30-101	II					X		
RWCU	RV70-1 1/4 -DC	128	D-3	E-202A	RV8	106 GR B	150 ?	1317	D	SEE CALCULATION NSP-30-101	II							
RWCU	V118-1-DB	128	D-3	E-202A	PBCC-49	106 GR B	150 ?	1317	D	SEE CALCULATION NSP-30-101	II					X		
RWCU	RV69-3/4-HB	128	C-4	E-202B	RV8	106 GR B	570	1317	D	SEE CALCULATION NSP-30-101	II					X		
RWCU	D158-1-ED	128	C-4	E-202B	D159-1	106 GR B	150 ?	1317	D	SEE CALCULATION NSP-30-101	II					X		
RWCU	D159-1-ED	128	C-4	E-202B	RC-24	106 GR B	150 ?	1317	D	SEE CALCULATION NSP-30-101	II					X		
MISC TURBN PIP	PS10-5-LST6	102/103	C-6/E-6	FST-10	MO1045	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	X	X						
MISC TURBN PIP	PS10A-5-LST6	103	E-6	PS10-5	MO1046	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	X	X						
MISC TURBN PIP	D109-1-HB	104	E-6	PS9-2	ST1276	106 GR B	582	1110	D	SEE CALCULATION NSP-30-101	X					X		
MISC TURBN PIP	D108-1-HB	104	E-5	PS9-2	ST1275	106 GR B	582	1110	0	SEE CALCULATION NSP-30-101	X					X		
NUCLEAR BOILER	D209-1-DC	115	E-2	X28F-1	IDV-6	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	X28F-1-DC	115	E-2	X-28F	PS2-102	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	X33A-1-DC	115	D-3	X-33A	FT6-51C	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	I					X		
SENSING LINE	X33D-1-DC	115	D-3	X-33D	FT6-60C	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	I					X		
SENSING LINE	X33F-1-DC	115	C-2	X-33F	FT6-51D	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	I					X		

nutech
ENGINEERS

DATE : 10-Jun-86

HIGH ENERGY LINE DESIGNATION (BECHTEL SYSTEMS)

							DESIGN OR NORMAL CONDITIONS					HELB COMPLIANCE MATRIX				EXCLUSION(S)		
SYSTEM	LINE DESIGNATION	P&ID(S)	DRAWING ZONE	TERMINAL PL PTS FROM	TO	MATERIAL STD	TEMP (F)	PRESS (PSI)	D/B	BREAK LOCATION(S)	APPENDIX R FIRE AREA(S)	NO ANALYSIS REQUIRED	SYSTEMS ANALYSIS	PIPE RUPTURE	PRA	SN DTA	22	LBB
SENSING LINE	X33E-1-DC	115	C-2	X-33E	PT6-60D	312/376TP304	582	1110	N	SEE CALCULATION NSP-30-101	I					X		
SENSING LINE	X33B-1-DC	115	D-5	X-33B	FT6-51B	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	I					X		
SENSING LINE	X33C-1-DC	115	D-5	X-33C	PT6-60B	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	I					X		
SENSING LINE	X27B-1-DC	116	C-5	X-27B	FT2-3-64A-K	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	X28A-1-DC	116	C-5	X-28A	PT6-53A	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	X28AA-1-DC	116	C-5	X28A-1	PS2-3-53A	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	X28AB-1-DC	116	C-5	X28A-1	LT2-3-112B	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	X28AC-1-DC	116	C-5	X28A-1	LT2-3-73A	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	X28B-1-DC	116	D-5	X-28B	LT2-3-61	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	X28BA1-DC	116	D-5	X28B-1	DPT2-3-65	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	X28C-1-DC	116	D-5	X-28C	LT2-3-85B	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	X49E-1-DC	115	C-5	X-49E	DP-116D	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	X49F-1-DC	115	C-5	X-49F	DP-116D	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	X28D-1-DC	116	D-5	X-28D	PT2-3-60A	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	X28E-1-DC	116	D-5	X-28E	LT2-3-61	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	X29A-1-DC	116	D-4	X-29A	PT6-53B	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	I					X		
SENSING LINE	X-29AA-1-DC	116	D-4	X29A-1-DC	PS2-3-53B	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	I					X		
SENSING LINE	X-29AB-1-DC	116	B-3	X29A-1-DC	LT2-3-112A	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	I					X		
SENSING LINE	X-29AC-1-DC	116	D-3	X29A-1-DC	LIS2-3-73B	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	I					X		
SENSING LINE	X29B-1-DC	116	D-4	X-29B	LT6-52B	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	I					X		
SENSING LINE	X29C-1-DC	116	D-4	X-29C	LT2-3-180B	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	I					X		
SENSING LINE	X29B-1-DC	116	D-4	X-29D	PT2-3-178D	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	I					X		
SENSING LINE	X40AE-1-DC	116	C-5/D-6	X-40A-E	FT2-3-63A	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	X40AD-1-DC	116	C-5/D-6	X-40A-D	FT2-3-63A/64A	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	X40AC-1-DC	116	C-5/D-6	X-40A-C	FT2-3-64B	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	X40AB-1-DC	116	C-5/D-5	X-40A-B	FT2-3-64C	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	X40AA-1-DC	116	C-5/D-5	X-40A-A	FT2-3-64D	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	X40BA-1-DC	116	C-5/D-5	X-40B-A	FT2-3-64E	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	X40BF-1-DC	116	C-5/A-5	X-40B-F	FT2-3-63B	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	X40BD-1-DC	116	C-5/A-5	X-40B-D	FT2-3-63B/64F	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	X40BC-1-DC	116	C-5/A-5	X-40B-C	FT2-3-64G	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	X40BB-1-DC	116	C-5/A-5	X-40B-B	FT2-3-64H	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	X40BE-1-DC	116	C-5/A-5	X-40B-E	FT2-3-64J	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	X40AF-1-DC	116	C-5/A-6	X-40A-F	FT2-3-64K	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	X40CA-1-DC	116	C-4/B-6	X-40C-A	FT2-3-63C	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	I					X		
SENSING LINE	X40DB-1-DC	116	C-4/B-6	X-40D-B	FT2-3-64V	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	I					X		
SENSING LINE	X40DD-1-DC	116	C-4/B-6	X-40D-D	FT2-3-64U	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	I					X		
SENSING LINE	X40DC-1-DC	116	C-4/B-6	X-40D-C	FT2-3-64T	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	I					X		
SENSING LINE	X40DF-1-DC	116	C-4/B-6	X-40D-F	FT2-3-63D/64S	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	I					X		
SENSING LINE	X40DE-1-DC	116	C-4/B-6	X-40D-E	FT2-3-63D	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	I					X		
SENSING LINE	X40DA-1-DC	116	C-4/A-6	X-40D-A	FT2-3-64R	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	I					X		
SENSING LINE	X40CB-1-DC	116	C-4/A-6	X-40C-B	FT2-3-64P	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	I					X		
SENSING LINE	X40CD-1-DC	116	C-4/A-6	X-40C-D	FT2-3-64N	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	I					X		
SENSING LINE	X40CC-1-DC	116	C-4/A-6	X-40C-C	FT2-3-64M	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	I					X		
SENSING LINE	X40CF-1-DC	116	C-4/A-6	X-40C-F	FT2-3-63C/64L	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	I					X		
SENSING LINE	X40CE-1-DC	116	C-4/A-6	X-40C-E	FT2-3-63C	312/376TP304	582	1110	D	SEE CALCULATION NSP-30-101	I					X		
SENSING LINE	X27A-1-DC	116/122	C-5/D-3	X-27A	DPS14-47AB	312/376TP304	562	1148	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	REW13A-1-1-EF	117-1	B-2	X-31A	DPT2-111A	312/376TP304	562	1148	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	REW13A-3-1-EF	117-1	A-3	X-31D	DPIS2-139A	312/376TP304	562	1148	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	REW13B-1-1-EF	117-1/120	C-5	X-51B	PS-2-128A	312/376TP304	562	1148	D	SEE CALCULATION NSP-30-101	I					X		
SENSING LINE	REW13B-3-1-EF	117-1/120	C-5	X-51C	PS-2-128B	312/376TP304	562	1148	D	SEE CALCULATION NSP-30-101	I					X		

DATE : 10-Jun-86

HIGH ENERGY LINE DESIGNATION (BECHTEL SYSTEMS)

SYSTEM	LINE DESIGNATION	P&ID(S)	DRAWING ZONE	TERMINAL PL PTS FROM	TERMINAL PL PTS TO	MATERIAL STD	DESIGN OR NORMAL CONDITIONS TEMP (F)	DESIGN OR NORMAL CONDITIONS PRESS (PSI)	D/N	BREAK LOCATION(S)	APPENDIX R FIRE AREA(S)	HELB COMPLIANCE MATRIX NO ANALYSIS REQUIRED	SYSTEMS ANALYSIS	PIPE RUPTURE	PRA	EXCLUSION(S) SM DIA	2%	LBB
SENSING LINE	REW13B-5-1-EF	117-1	D-6	X-32A	DPT2-111D	312/376TP304	562	1148	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	REW13D-21-1-EF	117-1	A-6	X-32D	DPT2-136B	312/376TP304	562	1148	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	REN13A-9-1-DC	117-1	D-3	X-52E	FT2-110C	312/376TP304	562	1248	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	REW13A-11-1-DC	117-1	D-3	X-52F	FT2-110C	312/376TP304	562	1248	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	REW13A-15-1-DC	117-1	A-2	X-31B	DPIS-136A	312/376TP304	562	1248	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	REW13B-19-1-DC	117-1	B-6	X-32B	DPIS-2-139B	312/376TP304	562	1248	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	REW14-1-1-DC	117-1	E-3	X-51F	C121	312/376TP304	562	1248	D	SEE CALCULATION NSP-30-101	I					X		
SENSING LINE	REW15-1-1-DC	117-1	E-3	X-51A	C122	312/376TP304	562	1248	D	SEE CALCULATION NSP-30-101	I					X		
SENSING LINE	REW19-1-1-DC	117-1	E-6	X-51E	C121	312/376TP304	562	1248	D	SEE CALCULATION NSP-30-101	I					X		
SENSING LINE	REW20-1-DC	117-1	E-6	X-51D	C122	312/376TP304	562	1248	D	SEE CALCULATION NSP-30-101	I					X		
SENSING LINE	REW16-1-1-DC	117-1	E-3	X-52B	C125	312/376TP304	562	1248	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	REW17-1-1-DC	117-1	E-3	X-52C	C122	312/376TP304	562	1248	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	REW-22-1-1-DC	117-1	E-6	X-52B	C212	312/376TP304	562	1248	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	REW-23-1-1-DC	117-1	E-6	X-52A	C122	312/376TP304	562	1248	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	REN13B-23-1-DC	117-2	C-5	X-32E	C57	312/376TP304	562	1248	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	REW13D-14-1-DC	117-2	C-5	X-32F	C57	312/376TP304	562	1248	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	REW13B-5-1-DC	117-2	C-5	X-31F	C57	312/376TP304	562	1248	D	SEE CALCULATION NSP-30-101	II					X		
SENSING LINE	REW13D-7-1-DC	117-2	C-5	X-31E	C57	312/376TP304	562	1248	D	SEE CALCULATION NSP-30-101	II					X		
SAMPLE LINE	SL19-3/8-DC	102	E-6	PSI-1N	SX3028	312/376TP304	540	975	N	SEE CALCULATION NSP-30-101	I					X		
SAMPLE LINE	SL20-3/8-DC	106	D-2	C4A-16	SX3029	106 GR B	302	434	D	SEE CALCULATION NSP-30-101	I					X		
SAMPLE LINE	SL21-3/8-DC	106	C-2	C4B-16	SX3030	106 GR B	302	434	B	SEE CALCULATION NSP-30-101	I					X		
SAMPLE LINE	SL22-3/8-DC	106	E-6	FW2A-14	SX3031	312/376TP304	540	975	N	SEE CALCULATION NSP-30-101	I					X		
SAMPLE LINE	SL23-3/8-DC	106	D-6	FW2B-14	SX3032	312/376TP304	540	975	N	SEE CALCULATION NSP-30-101	I					X		
SAMPLE LINE	REW32-1-3/4-DC	117-1	D-6	X-41	CT-3013	312/376TP304	562	1248	0	SEE CALCULATION NSP-30-101	II					X		
SAMPLE LINE	REW32-1-3/4-ED	128	B-4	REW3-3-ED	HWC-5	106 GR B	570	1136	D	SEE CALCULATION NSP-30-101	II					X		
SAMPLE LINE	REW13B-15-1-DC	117-1	D-5	X-50C	C57	312/376TP304	562	1248	D	SEE CALCULATION NSP-30-101	II					X		
SAMPLE LINE	REN13B-17-1-DC	117-1	D-5	X-50B	C57	312/376TP304	562	1248	D	SEE CALCULATION NSP-30-101	II					X		

TABLE 5.5-1

SYSTEM INTERACTION TABLE

	STEAM CHASE II/2F				TORUS AREA IV/1F		HPCI ROOM II/1E	RCIC ROOM III/1C	CONDENSER AREA X/12C					SJAE ROOM X/12E	TURBINE BLDG. FEED PUMP AREA IX 113 B,C				REACTOR BLDG 935 II/2C		RWCU PUMP ?/3D	MG SET ROOM ?/34	REACTOR BLDG II/3D	TURB BLDG IX/14C	TURB BLDG X/3C
	MAIN STEAM PS 1,2,3,4	HPCI PS18	RCIC PS17	FEEDWATER	HPCI PS18	RCIC PS17	HPCI PS18	RCIC PS17	CONDENSATE C4A	CONDENSATE C4B	FEEDWATER FW2A (2)	STEAM BYPASS PS7	MAIN STEAM PS4,3,2,1	MAIN STEAM PS9	CONDENSATE C7	CONDENSATE C4A	FEEDWATER FW2A	FEEDWATER FW2B	RWCU REWG		REWG REW3	REWG RWCU	REWG RWCU	(1)	(6)
HPCI ○ □	F	F	A	F FW 2B	(1)	A	F	A	A	A	A	F	A	A	A	A	A	A	A		A	A	A		
RCIC □	F	F	F	F FW 2A		F	A	F	A	A	A	A	A	A	A	A	A	A	A		A	A	A		
S/RV'S (7) △ ○	A	A	A	A		A	A	A	A	A	A	A	A	A	A	A	A	A	A		A	A	A		
RHR - LPCI DIV 1 □	A	A	A	A		A	A	A	Ā	A	A	Ā	Ā	Ā	Ā	Ā	Ā	Ā	A		A	A	A		
RHR - LPCI DIV 2 □	Ā	Ā	Ā	Ā		Ā	A	A	Ā	Ā	Ā	Ā	Ā	A	A	A	A	A	A		A	A	A		
RHR - SPC DIV 1 △	A	A	A	A		A	A	A	Ā	A	A	Ā	Ā	Ā	Ā	Ā	Ā	Ā	A		A	A	A		
RHR - SPC DIV 2 △	Ā	Ā	Ā	Ā		Ā	A	A	Ā	Ā	Ā	Ā	Ā	A	A	A	A	A	A		A	A	A		
RHR - SDC DIV 1 △	A	A	A	A		A	A	A	Ā	A	A	Ā	Ā	Ā	Ā	Ā	Ā	Ā	A		A	A	A		
RHR - SDC DIV 2 △	Ā	Ā	Ā	Ā		Ā	A	A	Ā	Ā	Ā	Ā	Ā	A	A	A	A	A	Ā (5)		F (5)	A	A		
RHR - SERVICE WATER DIV 1 △	A	A	A	A		A	A	A	Ā	A	A	Ā	(3) F	Ā	A	A	A	A	A		A	A	A		
RHR - SERVICE WATER DIV 2 △	Ā	Ā	Ā	Ā		Ā	A	A	Ā	Ā	Ā	Ā	Ā	A	A	A	A	A	A		A	A	A		
CORE SPRAY DIV 1 □ △	A	A	A	A		A	A	A	Ā	A	A	Ā	Ā	Ā	Ā	Ā	Ā	Ā	A		A	A	A		
CORE SPRAY DIV 2 □ △	Ā	Ā	Ā	Ā		Ā	A	A	Ā	Ā	Ā	Ā	Ā	A	A	A	A	A	Ā		F	A	F		
EMERGENCY SERVICE WATER DIV 1 ⬡	A	A	A	A		A	A	A	F	A	A	F	F	Ā	A	A	A	A	A		A	A	A		
EMERGENCY SERVICE WATER DIV 2 ⬡	F	F	F	F		F	A	A	F	F	F	F	F	A	A	A	A	A	A		A	A	A		
EMERGENCY POWER DIV 1 ⬡	A	A	A	A		A	A	A	Ā	A	A	Ā	Ā	Ā	F (4)	F (4)	F (4)	F (4)	A		A	A	A		
EMERGENCY POWER DIV 2 ⬡	Ā	Ā	Ā	Ā		Ā	A	A	Ā	Ā	Ā	Ā	Ā	A	A	A	A	A	A		A	A	A		
RPS CONTROL ROD DRIVES, CONTROL RODS ◊	A	A	A	A		A	A	A	A	A	A	A	A	A	A	A	A	A	A		A	A	A		
HVAC ⬡ DIV 1	A	A	A	A		A	A	A	Ā	A	A	A	Ā	Ā	Ā	Ā	Ā	Ā	A		A	A	A		
HVAC ⬡ DIV 2	Ā	Ā	Ā	Ā		Ā	A	A	Ā	Ā	Ā	Ā	Ā	A	A	A	A	A	A		A	A	A		

KEY

F = FAILURE DIRECTLY CAUSED BY A HIGH ENERGY LINE BREAK.

A = SYSTEM IS AVAILABLE.

Ā = SYSTEM IS UNAVAILABLE DUE TO THE FAILURE OF A REQUIRED FUNCTION OF ANOTHER SYSTEM.

SAFE SHUTDOWN FUNCTION

◊ - REACTIVITY CONTROL

□ - RESTORE AND MAINTAIN RPV WATER LEVEL

○ - RPV DEPRESSURIZATION

△ - DECAY HEAT REMOVAL

⬡ - AUXILIARY SUPPORT

NOTES

(1) BY STRESS ANALYSIS, THERE ARE NO BREAK LOCATIONS IN THIS AREA.

(2) THERE ARE NO SSD TARGETS IN THIS AREA FOR FW2B

(3) APPLIES TO PS4 ONLY.

(4) MCC 133 A B.

(5) DIV 2 CONTAINMENT SPRAY ONLY

(6) THERE ARE NO SSD TARGETS IN THIS AREA

(7) THE S/RV'S ARE RESPONSIBLE FOR RPV OVERPRESSURE PROTECTION

6.0 CONCLUSIONS & RECOMMENDATIONS

A systematic review of the high energy piping at the Monticello Plant was performed for the Northern States Power Company. It is concluded that paths to safe shutdown exist for each postulated break with coincident loss of offsite power and a single active failure (as described in Section 5.2) in all compartments with the following exceptions:

6.1 A longitudinal break of the steam bypass line (PS7-10"-ED) could impinge on both divisions of the Emergency Service Water System (SW30A-3-HF and SW30B-3-HF). As an observation it should also be noted that the reaction of pipe in the opposite direction would damage the HPCI System power cabling and the cables to Div. II of SPTMS. A circumferential break on condensate line C4A-16-GB on the inlet to heater E-13A could also jet impinge on both divisions of the Emergency Service Water Lines referred to above.

6.2 A jet impingement in the main steam chase from a break in either the Main Steam line (PS2-18-ED or PS3-18-ED) or the HPCI Steam line (PS18-8-ED) could damage the RCIC steam line (PS17-3-ED)and the Emergency Service Water line (SW30B-3-HF). Safe Shutdown without HPCI, RCIC, and one division of ESW, coincident loss of offsite power, and failure of the opposite division diesel generator, as the single active failure, has not been investigated.

6.3 A break on either Feedwater Line (FW2A-14-ED or FW2B-14-ED) within the Reactor Feedwater Pump Area (IX/13C and IX/13B) would subject MCC's 133 & 134 to adverse environmental effects. This break could also cause a breach in the ceiling to this area from a jet impingement or pipe whip. A breach in the ceiling to this area could also expose MCC's 142 and 143 to adverse environmental effects.

6.4 Compartment pressurization within the SJAE Room from a break in line PS9-3-ED could open the door to the room and expose Load Center 103 and BUS 15 as well as the 480V equipment to adverse environmental conditions. If the single active failure is the diesel generator (G11) of Div. I, shutdown of the unit could not be accomplished.

6.5 Jet impingement in the RWCU Area from line REW3-4"-DB could damage the copper tubing air lines to the seals on Primary Containment Atmospheric Control System isolation valves AO-2386 and AO-2387 and could result in loss of primary containment isolation function. (It should be noted that these valves are not required for safe shutdown. However, in order to meet the intent of Criterion 21 of the Giambusso letter identification of this concern is required.)

In order to mitigate each of the identified concerns, the following modifications and procedural changes are proposed:

<u>Items 6.1, 6.2, & 6.3</u>

Manual isolation valves will be installed in the Intake Structure on the Emergency Service Water Lines immediately downstream of the tees in each discharge pumping. The purpose of the tee is to separate cooling water flow to the D/G from the flow to the room coolers. By installing a manual isolation after the tee on the pipe transporting water to the room coolers, any postulated break in the ESW lines as a result of a HELB can be isolated from cooling flow necessary for the D/G's. Also, manual isolation valve ESW 58-2 will be relocated in the Torus Area from the Main Steam Pipe Chase. The relocation of the valve will allow isolation of any postulated break on line SW30B-3-HF. With the break isolated, cooling water for the Division II room coolers can be provided by the Service Water System from a Service Water Pump capable of being powered by the Diesel Generator. The new manual isolation valves on the ESW lines in the Intake Structure will be closed, so that cooling flow to the D/G's will not be lost due to HELB affecting SW30A-3-HF or SW30B-3-HF.

Additionally, an emergency operating procedure will be generated, which details how an electrical realignment could be made such that one diesel generator could power the other essential division.

With these modifications, concerns about HELB affecting the Emergency Service Lines with a single active failure of a Diesel Generator or Emergency Service Water Pump is mitigated.

<u>Item 6.3</u>

For Feedwater and Condensate line breaks in the Feedwater Pump area, calculations will be performed to show that the ceiling to the room can withstand the force of a direct jet impingement without failure for the Condensate Line and the 6" Feedwater crosstie pipingbreaks. For the Main Feedwater Lines (14") jet impingement shields will be installed to protect the ceiling. Therefore, no break that adversely affects MCC 133 will also affect MCC 143.

Item 6.4

For the SJAE Room , a compartmental pressurization analysis will be performed to quantify the differential pressure across the door to the compartment. Based upon this pressure, the door will be redesigned or replaced to be able to accept the differential pressure without opening. All other possible vent paths from the SJAE Room to the 4KV switchgear area will be eliminated. Therefore, a HELB in the SJAE Room will not affect the switchgear area.

Item 6.5

Tests will be conducted on valves AO-2386 and AO-2387 to establish whether allowable leak rates can be maintained with the seals deflated. If the results of the tests are unacceptable, relocation of the accumulators, check valves, and tubing will be considered, or consideration will be given to providing jet impingement shields to protect the equipment.

7.0 REFERENCES

7.1 Letter, from A. Giambusso, Deputy Director for Reactor Projects, to Northern States Power Company, Subject: High Energy Breaks Outside of the Containment, Dec. 18, 1972 (File # NSP730.0009).

7.2 Postulated Pipe Failures Outside Containment, Monticello Nuclear Generating Plant, Monticello, Minnesota, August 1973 (File # NSP730.0008).

7.3 IE Bulletin No. 80-11, Masonary Wall Design, Issued by the Nuclear Regulatory Commission, May 8, 1980 (File # NSP730.0017)

7.4 Standard Review Plan 3.6.1, Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment, Branch Technical Position ASB 3-1, Rev. 0, March 1975. (File # NSP730.0009)

7.5 Standard Review Plan 3.6.2, Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping, Branch Technical Position, MEB 3-1, Revision 1, July 1981 (File # NSP730.0009).

7.6 United States Atomic Energy Commission - Safety Evaluation by the Directorate of Licensing, Docket No. 50-263, Monticello Nuclear Generating Plant - "Analysis of the Consequences of High Energy Piping Failures Outside Containment", July 29, 1974 (File # NSP730.0009).

7.7 ANSI/ANS-58.2-1980, American National Standard, Design Basis for Protection of Light Water Nuclear Power Plants Against Effects of Postulated Pipe Rupture, December 31, 1980 (File #NSP730.0018).

7.8 Monticello Nuclear Generaing Plant, Re-examination of Appendix R Separation Analysis, September 1985 File #NSP730.0014).

7.9 Report, 01-000910-1137, Monticello Nuclear Generaing Plant Environmental Effects Due to Pipe Rupture, Rev. 1, August 1985 (File #NSP730.0019).

7.10 Updated Safety Analysis Report for the Monticello Nuclear Generating Plant, Northern States Power Company, Revision 3, December 1984.

7.11 NEDO-22087, Fire Protection and Safe Shutdown Systems Analysis Report, Monticello Nuclear Generating Plant, Northern States Power Company, June 1982 (File #NSP730.0014).

7.12 Survey Book, High Energy Lines in Systems Considered in 1972 HELB Analysis, May 1986 (File #730.0016).

7.13 Letter, Dennis L. Zeimann to Northern States Power Company, subject: Questions on Monticello High Energy Line Breaks Outside Containment, January 18, 1974 (File #NSP730.0016).

APPENDIX A

HIGH ENERGY LINE

BREAK LOCATIONS

A.0 HIGH ENERGY LINE BREAK LOCATIONS

The postulated break locations for each high energy line was based on the criteria established in the Giambusso letter (Reference A.7-1). The systems evaluated were the Primary Steam, HPCI, RCIC, Feedwater, Condensate, RWCU, and miscellaneous Sensing and Sample Lines. Calculations were performed in accordance with the "HELB" Evaluation Criteria document (NSP730.0020-001) to determine where the break locations would be postulated for all seismic category I piping (Reference A.7-2). For each system, the calculation results are as follows.

A.1 Primary Steam

The following break locations are postulated for the four Main Steam Lines. The break locations are identified on Figure A-1.

PS1TE-1: Terminal End at penetration X-7A in the Reactor Building Steam Chase Area.

PS1B-1: Approximately 14'-3" north of column line K and 9'-0" west of column line 6 at El. 942'-6" in the Turbine Condenser Area (TBCA).

PS1B-2: Approximately 19'-3" north of column line K and east of support PS5 on the end of the elbow at El.942'-6" in the Turbine Building Condenser Area.

PS1TE-2: Terminal end at stop valve SV1 in the TBCA.

PS2TE-1: Terminal end at penetration X-7B in the Reactor Building Steam Chase.

PS2B-1: Approximately 14'-3" north of column line K and 15'-6" east of column line 6 at El. 942'-6" in the TBCA.

PS2B-2: Approximately 16'-3" north of column line K and east of PS10 on the end of the elbow at EL. 942'-6" in the TBCA.

PS2TE-2: Terminal end at stop valve SV2 in the TBCA.

PS3TE-1: Terminal end at penetration X-7C in the Reactor Building Steam Chase Area.

PS3B-1:	Approximately at 8'-3" north of column line K and 3'-0" east of column line 6 at El. 942'-6" in the TBCA.
PS3B-2:	At first elbow end south of stop valve SV3 at El. 938'-0" in the TBCA.
PS3TE-2:	Terminal end at stop valve SV3 in the TBCA.
PS4TE-1:	Terminal end at penetration X-7D in the Reactor Building Steam Chase.
PS4B-1:	Approximately 9'-0" east of column line 6 next to PS30, El. 942'-6" in the TBCA.
PS4B-2:	At first elbow end south of stop valve SV4, El. 938'-0" in the TBCA.
PS4TE-2:	Terminal end at stop valve SV4 in the TBCA.
PS1TE-3:	Terminal end at control valve CV1 in the TBCA.
PS1B-3:	At the first bend north of control valve CV1, El. 923'-0" in the TBCA.
PS1B-4:	At the first elbow south of turbine, El. 961'-11" in the Turbine Building.
PS1TE-4:	Terminal end at turbine, El. 961'-11" in the Turbine Building.
PS2TE-3:	Terminal end at control valve CV2 in the TBCA.
PS2B-3:	At the first bend north of control valve CV2, El. 923'-0" in the TBCA.
PS2B-4	Approximately 6'-2" north of column line Gh on last bend before turbine, at El 920'-0" in the TBCA.
PS2TE-4:	Terminal end at turbine, El. 951'-7 1/2" in the Turbine Building.
PS3TE-3:	Terminal end at control valve CV3 in the TBCA.
PS3B-3 :	At the first bend north of control valve CV3, El. 923'-0" in the TBCA.
PS3B-4:	Approximately 10'-2" north of column line Gh on last bend before turbine, at El. 920'-0" in the TBCA.

PS3TE-4: Terminal end at turbine, El. 951'-7 1/2" in the Turbine Building.

PS4TE-3: Terminal end at control valve CV4 in the TBCA.

PS4B-3: At the first bend north of control valve CV4, El. 923'-0" in the TBCA.

PS4B-4: At the first elbow north of turbine, El. 961'-11" in the Turbine Building.

PS4TE-4: Terminal end at turbine, El. 961'-11" in the Turbine Building.

PS30B-1: At the first elbow down from PS4-18"-ED, El. 940'-1 3/8" in the TBCA.

PS30B-2: At the first elbow down from PS1-18"-ED, El. 940'-1 3/8" in the TBCA.

The following break locations are for Primary Steam piping connecting to the four mainsteam lines. All break locations ar within the Turbine Building Condenser Area and are identified on Figure A-2 (NX-13142-15).

PS11TE-1: Terminal end at PS1-18"-ED, El. 941'-6" in the Turbine Building.

PS12TE-1: Terminal end at PS2-18"-ED, El. 941'-6" in the Turbine Building.

PS12B-1: Approximatly 14'-3" north of column line K and 2'-9" west of column line 8, El. 940'-8" in the Turbine Building.

PS13TE-1: Terminal end at PS3-18"-ED, El. 941'-6" in the Turbine Building

PS14TE-1: Terminal end at PS4-18"-ED, El. 941'-6" in the Turbine Building.

PS7B-1: Approximately 4'-3" south of column line Jc and 7'-5 3/4" east of column line 9, El. 940'-3" in Turbine Building.

PS7B-2: Approximately 23'-0" north of column line Jc and 7'-5 3/4" east of column line 9, El. 939'-11 9/16" in the Turbine Building.

PS7TE-1: Approximately 59'-0" north of column line Jc and 2'-10 1/2" east of column line 9, El. 939'-11 5/8" in the Turbine Building.

PS7TE-2: Terminal end on north side of bypass control valve, El. 940'-3/8" in the Turbine Building.

PS7TE-3: Terminal end on south side of bypass control valve, El. 940'-3/8" in the Turbine Building.

PS8TE-1: Terminal end on west side of bypass control valve, El. 940'-3/8" in the Turbine Building.

PS8B-1: Approximately 40'-6" north of column line Jc and 1'-5 1/4" west of column line 9, El. 940'-8 11/16" in the Turbine Building.

PS8B-2: Approximately 4'-6" south of turbine center line and 1'-5 1/4" west of column line 9, El. 912'-11" in the Turbine Building.

PS8TE-2: Terminal end at 16 x 8 flow nozzle at penetration 4, 4'-6" south of Turbine center line, El. 912'-11" in the Turbine Building.

PS6TE-1: Terminal end west side of bypass control valve, El. 940'-3/8" in the Turbine Building.

PS6B-1: Approximately 43'-6" north of column line Jc and 1'-5 1/4" west of column line 9, El. 940'-8 11/16" in the Turbine Building.

PS6B-2: Approximately 44'-6" north of column line Jc and 1'-5 1/4" west of column line 9, El. 940'-8 11/16" in the Turbine Building.

PS6B-3: At support PS-51 and 3'-11 1/2" north of turbine center line, El. 912'-11" in the Turbine Building.

PS6TE-2: Terminal end at 16 x 8 flow nozzle at penetration 4, 3'-11 1/2" north of turbine center line, El. 912'-11" in the Turbine Building.

PS21TE-1: Terminal end at orifice RO-1618 on line PS21-4"-ED.

The following break locations are for the Primary Steam Drain Line in the Steam Chase Area.

PS15TE-1: Terminal end at penetration X-8 in the Reactor Building Steam Chase Area.

PS15B-1: Approximately 23'-3" south of column line L and at valve MO-2374, El. 933'-0" in Reactor Building Steam Chase Area.

PS15B-2: Approximately 23'-3 south of column line L and 10'-9" west of valve MO-2374, El. 933'-0" in Reactor Building Steam Chase Area.

PS15TE-2: At support SR 140 which has been removed and replaced by a new support SR 721. The new support SR 721 is not shown in Figure A-3A, so we will use SR 140 as a reference point, El. 933'-0" in the Reactor Building Steam Chase Area.

These postulated break locations are identified on Figure A-3A.

All other high energy lines, forming part of the primary steam system, are identified as non-seismic lines. Hence, break locations were postulated at all fittings (e.g., elbows, tees, valves, brach lines, etc.) as defined by the break location criteria.

A.2 High Pressure Coolant Injection Break Locations

The following break locations are postulated for the HPCI steam line, PS18-8. These postulated break locations are identified on Figure A-3B (NX-13142-42).

PS18TE-1: Terminal end at penetration X-11, El. 949'-0" in the Reactor Building Steam Chase Area.

PS18B-1: Approximately 24'-6" south of column line L and 9'-0" west of column line 6 at valve MO-2035, El. 949'-0" in the Reactor Building Steam Chase Area.

PS18B-2: Approximately 3'-8" south of column line L and 0'-2 1/4" west of column line 6 at support SR-85, El.931'-0" in the Reactor Building Steam Chase Area.

PS18TE-2: Terminal end at HPCI pump S-201 nozzle, El. 905'-6" in HPCI Room.

All other high energy lines, of the High Pressure Coolant Injection System (steam side), are identified as non-seismic lines. Hence, break locations were postulated at all fittings (e.g., elbows, tees, valves, branch lines, etc.) as defined by the break location criteria.

A.3 Reactor Core Isolation Cooling System Break Loacations

The following break locations are postulated for the RCIC steam line, PS17-3. These postulated break locations are identified on Figure A-4 (NX-13142-43).

PS17TE-1: Terminal end at penetration X-10, El. 948'-6" in the Reactor Building Steam Chase Area.

PS17B-1: Approximately 3'-10" south of column line L and 7'-4 5/8" east of column line 6, El. 936'-6" in the Reactor Building Steam Chase Area.

PS17B-2: Approximately 3'-10" south of column line L and 2'-3 3/4" east of column line 6, El. 937'-6" in the Reactor Building Steam Chase Area.

PS17TE-2: At support PSH 112, 3'-10" south of column line L and 4'-1 3/4" east of column line 6 in the Reactor Building Torus Area.

A.4 Feedwater Break Locations

The following break locations are postulated for the Feedwater piping. These postulated break locations are identified on Figure A-5 (NX-13142-52) and A-6 (NX-13142-53).

FW2ATE-1: Terminal end at penetration X-9B, El. 940'-6" in the Reactor Building Steam Chase Area.

FW2AB-1: Approximately 1'-7 1/2" north of penetration X-9B, El. 940'-6" in the Reactor Building Steam Chase Area.

FW2AB-2: Approximately 17'-9" south of column line Jc and 6'-0" east of column line 6, El. 960'-4 5/8" on the turbine operating floor.

FW2ATE-2: Terminal end at heater E15-A nozzle, 12'-6" south of column line Jc and 9'-0" east of column line 6, El. 960'-4 5/8" on the turbine operating floor.

FW2ATE-3: Terminal end at heater E14-A nozzle, 2'-6" south of column line Jc and 9'-0" east of column line 6, El. 951'-7 5/8" on the turbine operating floor.

FW2AB-3: Approximately 15'-7 1/2" south of column line Hj and 7'-8" west of column line 10, El. 924'-4" in the Feedwater Pump Area.

FW2B-1: Approximately on column line 10 south end of valve CV-613, El. 919'-5" in the Feedwater Pump Area.

FW2ATE-4: Terminal end of Feedwater Pump P-2A, El. 918'-4 1/2" in the Feedwater Pump Area.

FW2BTE-1: Terminal end at penetration X-9A, El. 940'-6" in the Reactor Building Steam Chase Area.

FW2BB-1: Approximately 6'-0" west of column line 6 where TW3-12"-ED taps into FW2B-14"-ED, El.940'-6" in the Reactor Building Steam Chase Area.

FW2BB-2: Approximately 17'-9" south of column line Jc and 9'-0" west of column line 6, El. 960'-4 5/8" on the turbine operating floor.

FW2BTE-2: Terminal end at heater E15-B nozzle, 12'-6" south of column line Jc and 9'-0" west of column line 6, El. 960'-4 5/8" on the turbine operating floor.

FW2BTE-3: Terminal end at heater E14-B nozzle, 2'-6" south of column line Jc and 9'-0" west of column line 6, El. 951'-7 5/8" on the turbine operating floor.

FW2BB-3: Approximately 21'-0" south of column line Hj and 7'-8" west of column line 10, El. 922'-6" in Feedwater Pump Area.

FW2BTE-4: Terminal end at feedwater pump P-2B, El. 918'-4 1/2" in Feedwater Pump Area.

FW2B-2: Approximately 0'-10" south of FW2-14"-DE on column line 10, El. 919'-5" in Feedwater Pump Area.

FW2B-3: Approximately 2'-5" south of FW2-14"-DE and 6'-6" west of column line 10, El. 919'-5" in Feedwater Pump Area.

A.5 Condensate System Break Locations

The following break locations are postulated for the Condensate System. These postulated break locations are identified on Figure A-7 (NX-13142-1) and A-8 (NX-13142-2).

C4ATE-1: Terminal end at heater E-13A nozzle, El. 937'-0" in TBCA.

C4AB-1: Approximately 1'-2" south of column line Jc and 7'-0" west of column line 10, El. 928'-4" in Feedwater Pump Area.

C4AB-2: Approximately 1'-2" south of column line Jc and 2'-5" east of column line 10, El. 926'-4" in the Feedwater Pump Area.

C4ATE-2: Terminal end at feedwater pump, P-2A, El. 915'-5" in the Feedwater Pump Area.

C4BTE-1: Terminal end at heater E-13B nozzle, El. 937'-0" in the TBCA.

C4BB-1: Approximately 6'-9" south of E-13B center line and 10'-5" west of column line 5, El. 937'-0" in the TBCA.

C4BB-2: Approximately 5'-0" north of C2B-16"-GB and 4'-6" east of column line 6, El.936'-11" in TBCA.

C4BTE-2: Terminal end at feedwater pump P-2B, El. 915'-5" in Feedwater Pump Area.

All other high energy lines of the Condensate System, are identified as non-seismic lines. Hence, break locations were postulated at all fittings (e.g., elbows, tees, valves, branch lines, etc.) as defined by the break location criteria.

A.6 Reactor Water Cleanup Break Locations

All portions of the RWCU System, identified as having high energy piping, are classified as non-seismic lines. Hence, break locations were postulated at all fittings (e.g., elbows, tees, valves, branch lines, etc.) as defined by the break location criteria.

A.7 References

A.7-1 Letter, from A. Giambusso, Deputy Director for Reactor Projects, to Northern States Power Company, Subject: High Energy Breaks Outside of the Containment, December 18, 1972 (File # NSP730.0009).

A.7-2 NUTECH file, "Determination of high energy pipe locations for the systems, Bechtel looked at in 1973." NUTECH file No. NSP730.0101-001 Rev. 0.

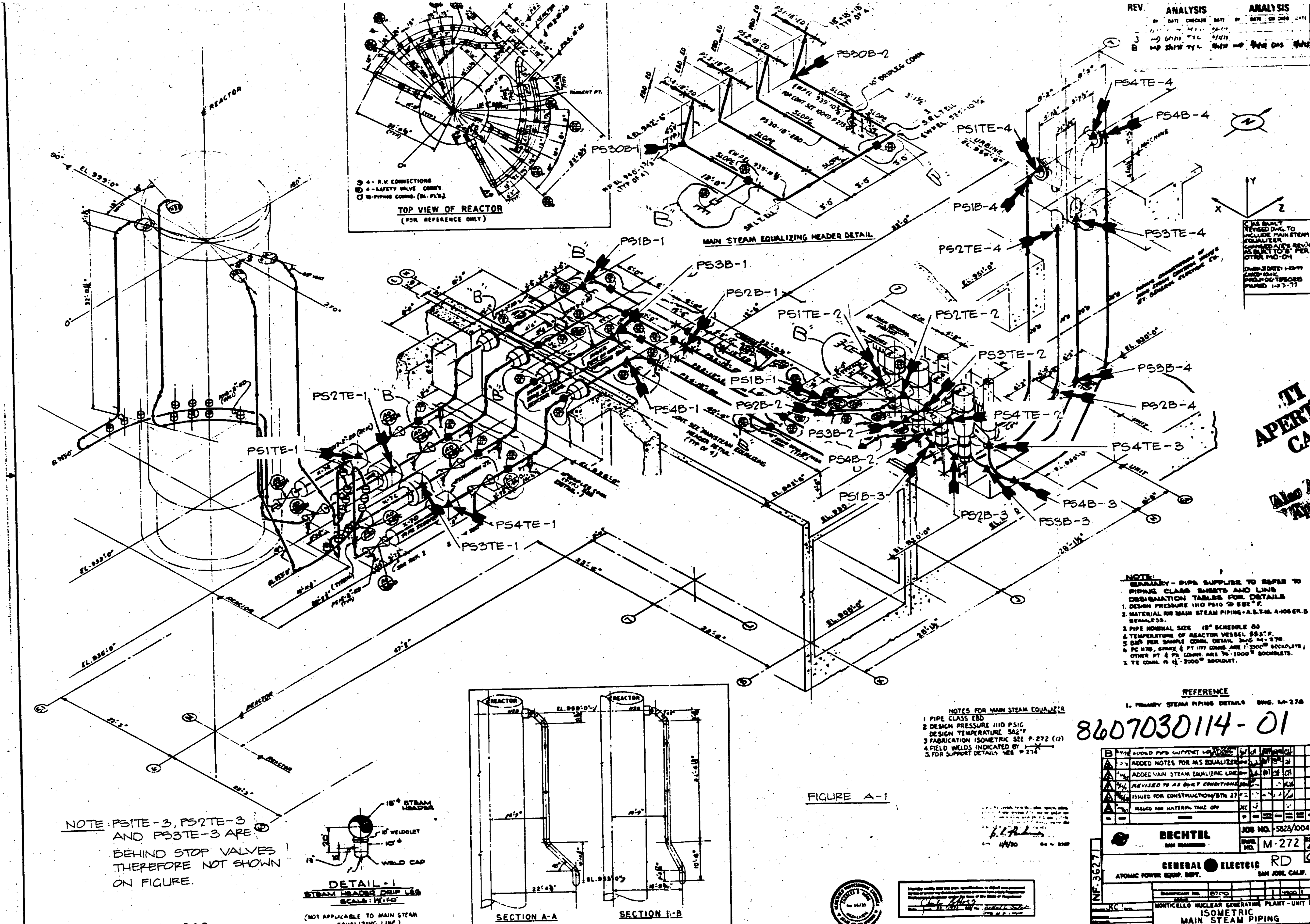
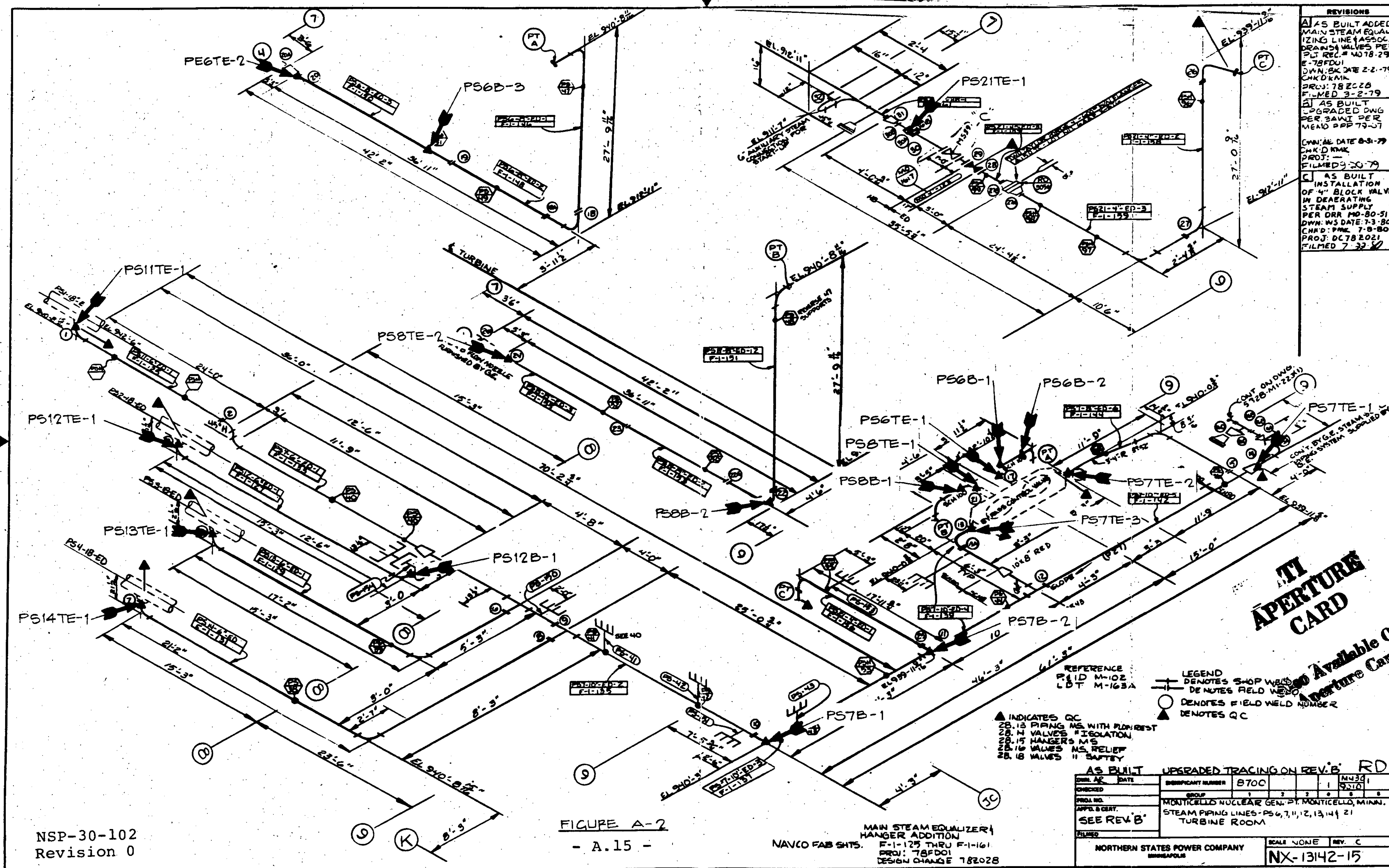

TOP VIEW OF REACTOR
(FOR REFERENCE ONLY)
4 - R.V. CONNECTIONS
4 - SAFETY VALVE CONN'S.
MAIN STEAM EQUALIZING HEADER DETAIL
PS30B-1
PS30B-2
PS1B-1
PS3B-1
PS2B-1
PS1TE-2
PS2TE-2
PS3TE-2
PS1B-2
PS2B-2
PS3B-2
PS4B-1
PS4B-2
PS4TE-2
PS4TE-3
PS1B-3
PS2B-3
PS3B-3
PS4B-3
PS1TE-4
PS4TE-4
PS4B-4
PS1B-4
PS2TE-4
PS3TE-4
PS3B-4
PS2B-4
PS2TE-1
PS1TE-1
PS4TE-1
PS3TE-1
REACTOR
NOTE: PS1TE-3, PS2TE-3 AND PS3TE-3 ARE BEHIND STOP VALVES THEREFORE NOT SHOWN ON FIGURE.
DETAIL-1
STEAM HEADER DRIP LEG
SCALE: 1½"=1'-0"
(NOT APPLICABLE TO MAIN STEAM EQUALIZING LINE)
18" STEAM HEADER
WELD CAP
SECTION A-A
SECTION B-B
NOTE:
SUMMARY - PIPE SUPPLIER TO REFER TO PIPING CLASS SHEETS AND LINE DESIGNATION TABLES FOR DETAILS
1. DESIGN PRESSURE 1110 PSIG @ 582°F.
2. MATERIAL FOR MAIN STEAM PIPING - A.S.T.M. A-106 GR.B SEAMLESS.
3. PIPE NOMINAL SIZE 18" SCHEDULE 80
4. TEMPERATURE OF REACTOR VESSEL 553°F.
5. SEE FOR SAMPLE CONN. DETAIL DWG M-278.
6. PC 1170, SPARE & PT 1177 CONNS. ARE 1"-3000# SOCKOLETS; OTHER PT & PX CONNS. ARE ¾"-3000# SOCKOLETS.
7. TE CONN. IS 1½"-3000# SOCKOLET.
REFERENCE
1. PRIMARY STEAM PIPING DETAILS DWG. M-278
NOTES FOR MAIN STEAM EQUALIZER
1 PIPE CLASS EBD
2 DESIGN PRESSURE 1110 PSIG
DESIGN TEMPERATURE 582°F
3 FABRICATION ISOMETRIC SEE P-272 (Q)
4 FIELD WELDS INDICATED BY
5 FOR SUPPORT DETAILS SEE P-274
8607030114-01
ADDED PIPE SUPPORT LOCATIONS
ADDED NOTES FOR MS EQUALIZER
ADDED MAIN STEAM EQUALIZING LINE
REVISED TO AS BUILT CONDITIONS
ISSUED FOR CONSTRUCTION/BTN 27
ISSUED FOR MATERIAL TAKE OFF
BECHTEL
JOB NO. 5828/10040
DWG. NO. M-272
REV. 4
GENERAL ELECTRIC
ATOMIC POWER EQUIP. DEPT.
SAN JOSE, CALIF.
RD
MONTICELLO NUCLEAR GENERATING PLANT - UNIT 1
ISOMETRIC
MAIN STEAM PIPING
NORTHERN STATES POWER COMPANY
NF-36271
NF-3627 B
TI APERTURE CARD
Also Available On Aperture Card

FIGURE A-1



NSP-30-102
Revision 0

FIGURE A-2

- A.15 -

8607030114-02

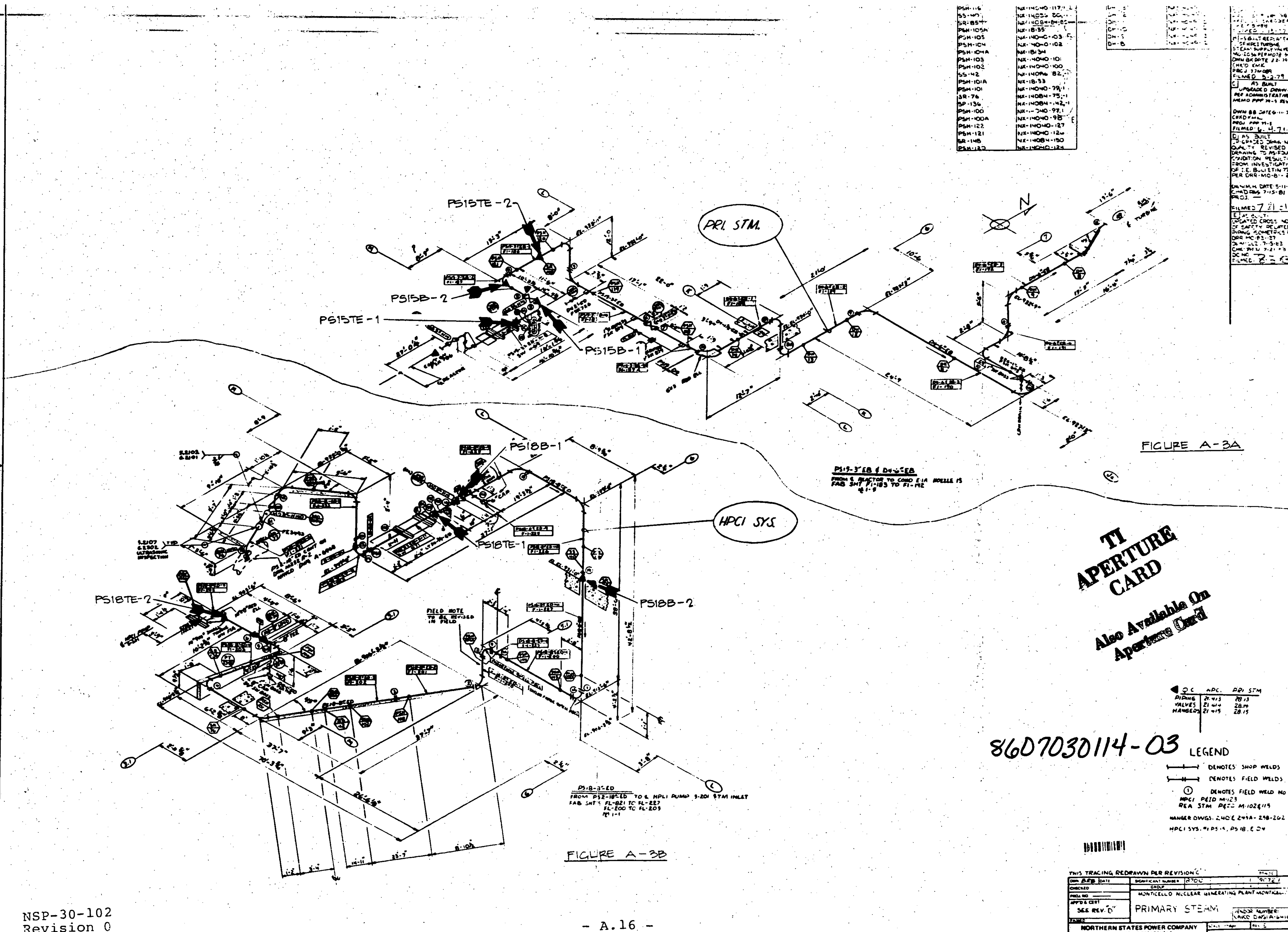


FIGURE A-3A

FIGURE A-3B

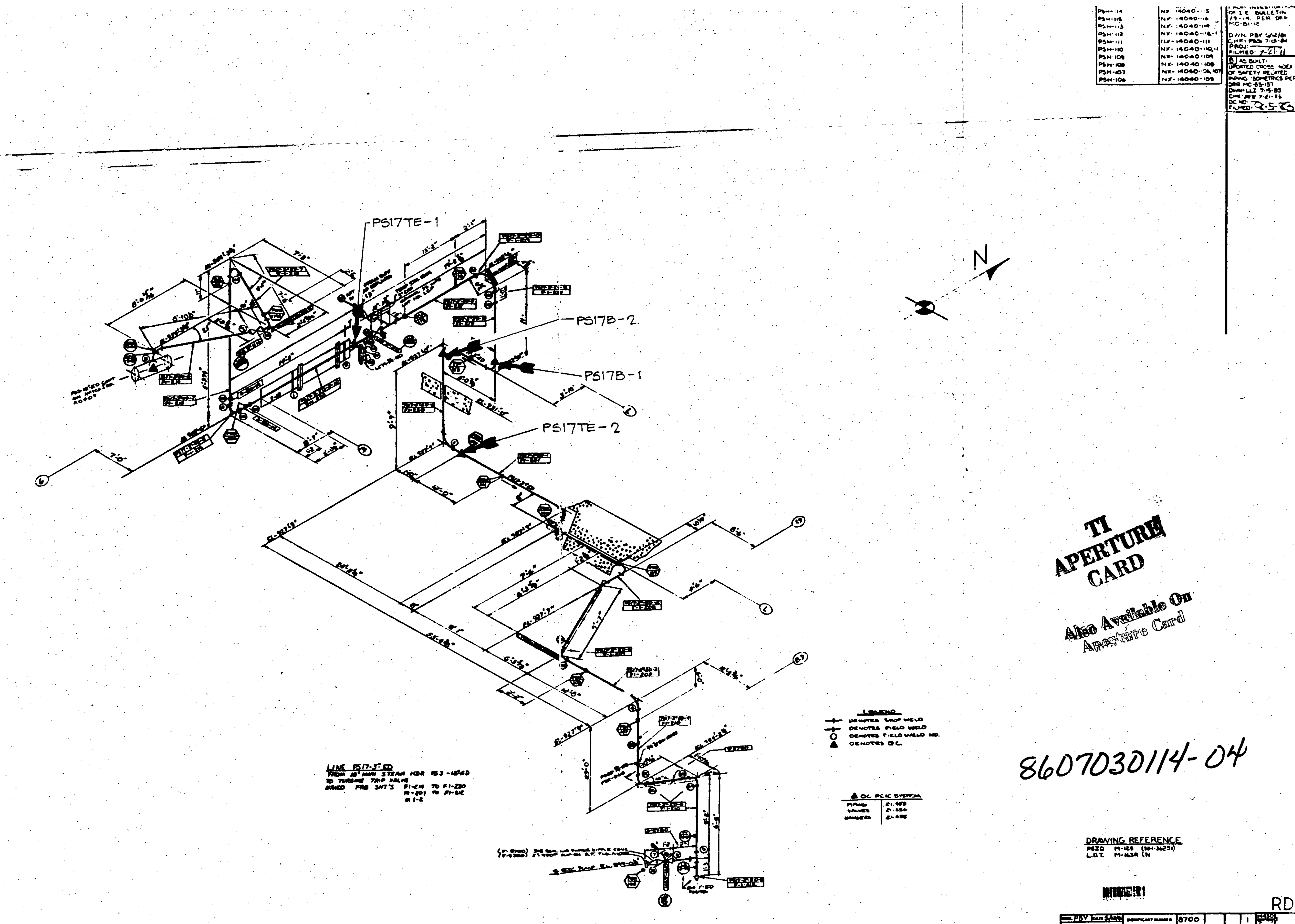
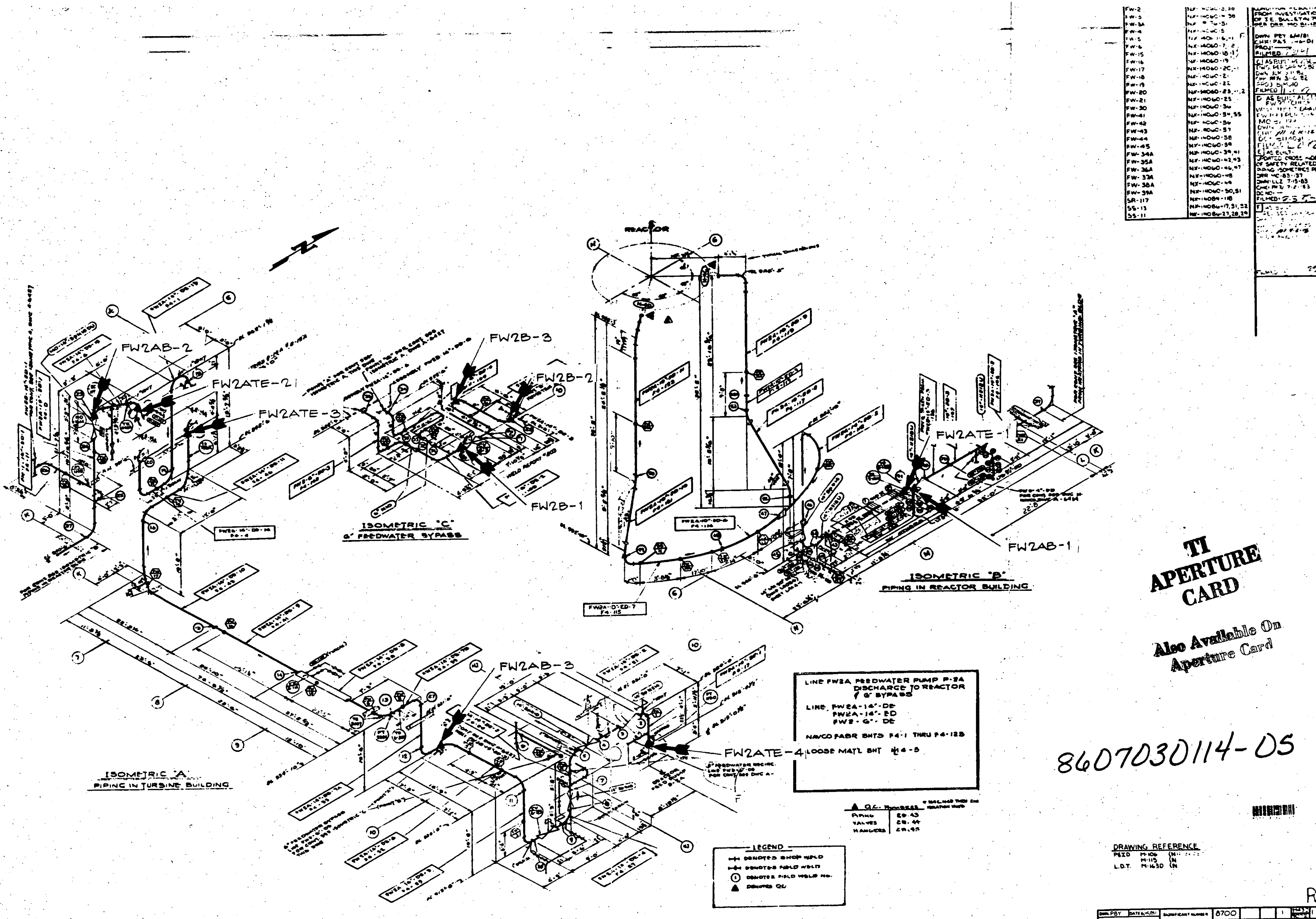


FIGURE A-4

NSP-30-102
Revision 0



FW2AB-2
FW2ATE-2
FW2ATE-3
FW2B-3
FW2B-2
FW2B-1
REACTOR
FW2ATE-1
FW2AB-1
ISOMETRIC "C"
6" FEEDWATER BYPASS
ISOMETRIC "B"
PIPING IN REACTOR BUILDING
ISOMETRIC "A"
PIPING IN TURBINE BUILDING
FW2AB-3
FW2ATE-4
LINE FW2A FEEDWATER PUMP P-2A DISCHARGE TO REACTOR & 6" BYPASS
LINE FW2A-14"-DE
FW2A-14"-ED
FW2-6"-DE
NAVCO FABR SHTS F4-1 THRU F4-125
LOOSE MATL SHT 14-5
LEGEND
DENOTES SHOP WELD
DENOTES FIELD WELD
DENOTES FIELD WELD NO.
DENOTES QC
TI APERTURE CARD
Also Available On Aperture Card
8607030114-05
DRAWING REFERENCE
MONTICELLO NUCLEAR PLT
FEEDWATER PUMP DISCHARGE
NORTHERN STATES POWER COMPANY
NX-13142-52

FIGURE A-5

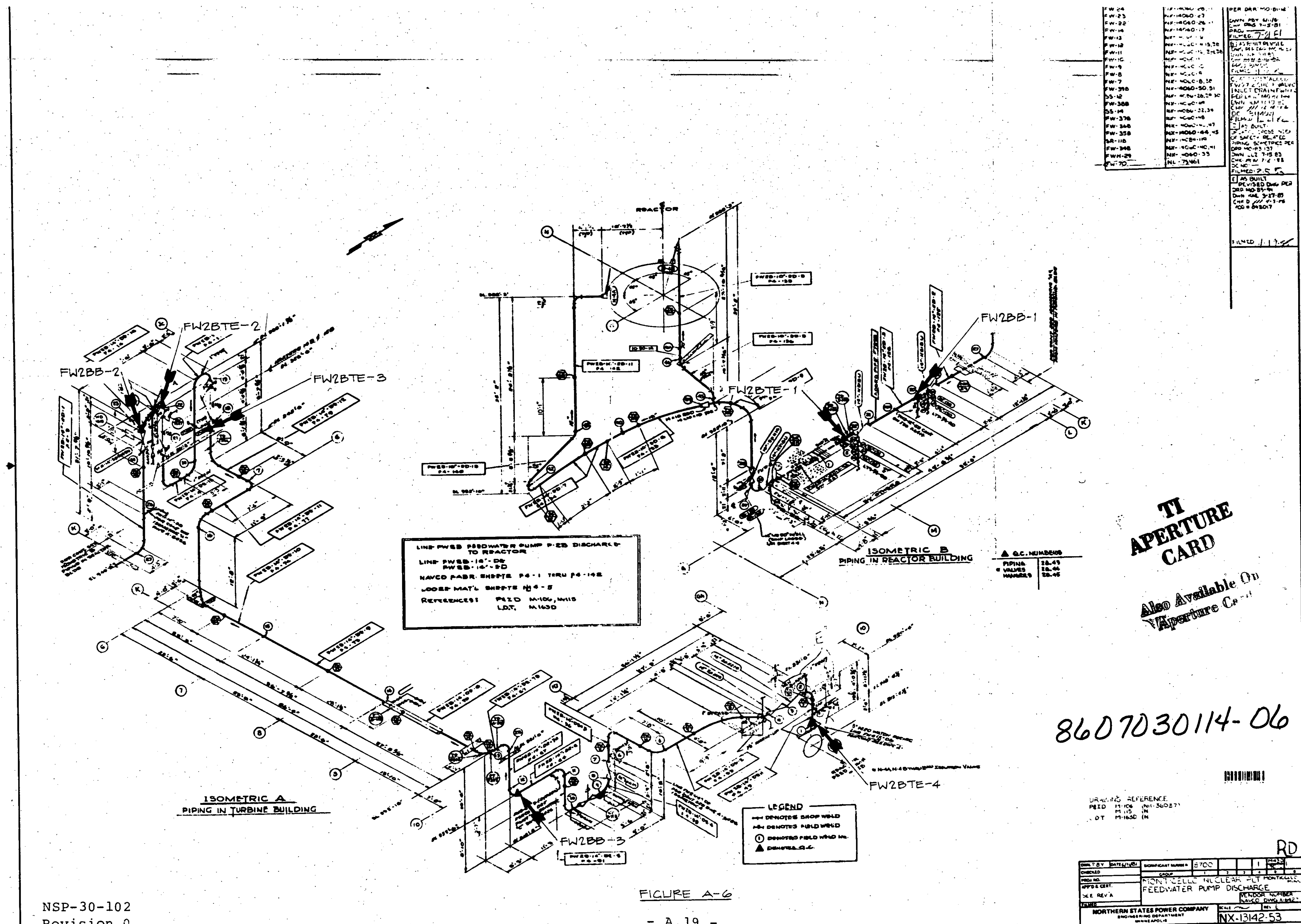
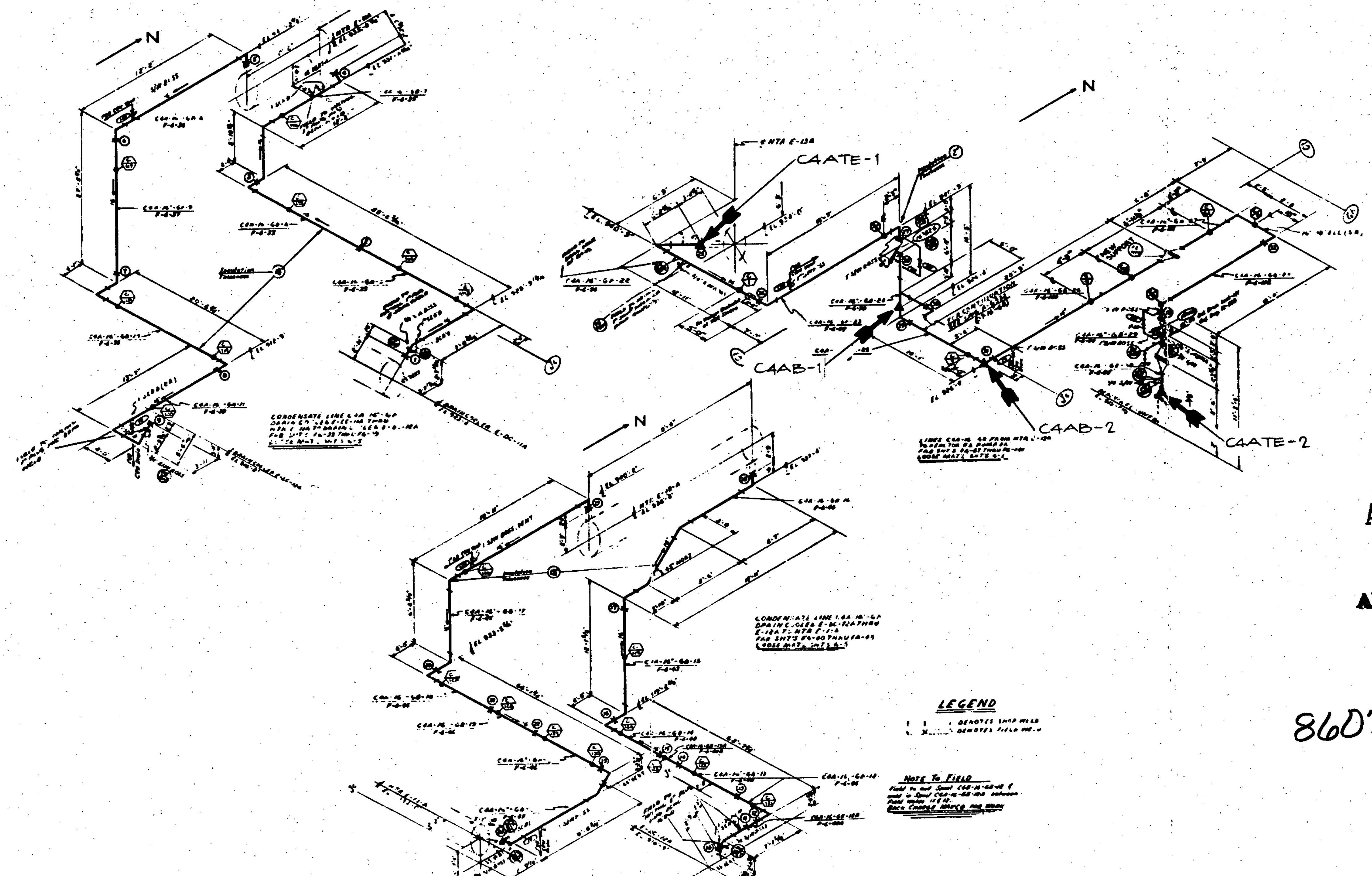

REACTOR
FW2BTE-2
FW2BB-2
FW2BTE-3
FW2BTE-1
FW2BB-1
FW2BTE-4
FW2BB-3
ISOMETRIC A
PIPING IN TURBINE BUILDING
ISOMETRIC B
PIPING IN REACTOR BUILDING
LEGEND
TI APERTURE CARD
Also Available On Aperture Card
8607030114-06
NORTHERN STATES POWER COMPANY
FEEDWATER PUMP DISCHARGE
NX-13142-53

FIGURE A-6



TI APERTURE CARD

Also Available On Aperture Card

8607030114-07

MONTICELLO NUCLEAR PLT.
CONDENSATE PIPING
NORTHERN STATES POWER COMPANY
8700
REV B
NX-13142-1

NSP-30-102
Revision 0

FIGURE A-7

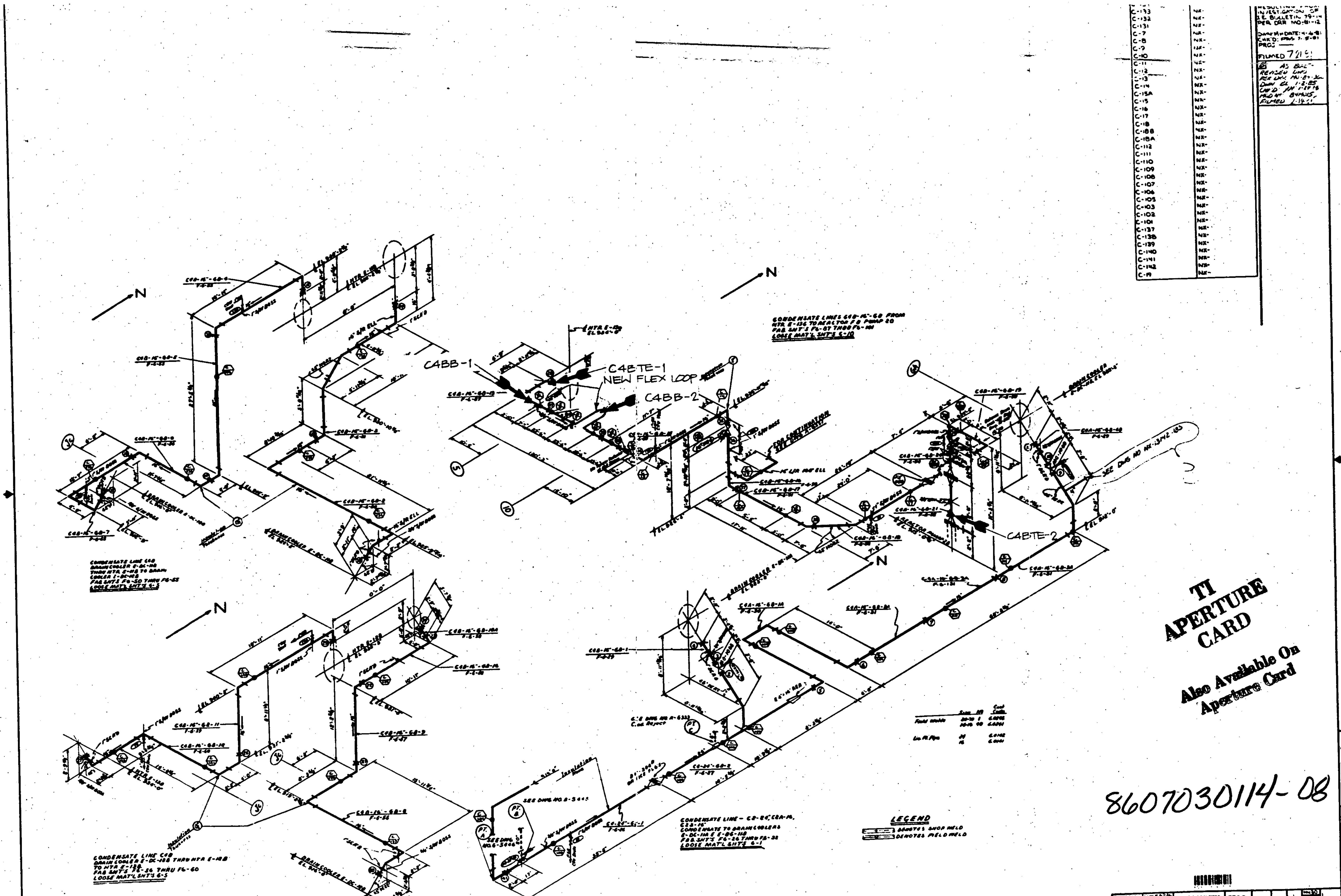

N
C4BB-1
C4BTE-1
NEW FLEX LOOP
C4BB-2
C4BTE-2
TI
APERTURE
CARD
Also Available On
Aperture Card
8607030114-08
NX-13142-2
LEGEND
MONTICELLO NUCLEAR PLT.
CONDENSATE PIPING
NORTHERN STATES POWER COMPANY
NX-13142-2

FIGURE A-8