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March 21, 1973

Roll 7208
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SUPPLEMENT NO. 1

TO

SAFETY EVALUATION

BY THE

DIRECTORATE OF LICENSING

U. S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

NORTHERN STATES POWER COMPANY

PRAIRIE ISLAND UNITS 1 & 2

DOCKET NOS. 50-282 & 50-306

Items missing

1. Amendments 28, 29, 31
~~2. [illegible] Dynamic Analysis [illegible]~~
2. Large Drawings of Typical Pipe [illegible]
3. Large Isometric Steam

INTRODUCTION

The Safety Evaluation Report (SER) for the Prairie Island Nuclear Generating Plant, Units 1 and 2, dated September 28, 1972, concluded that the application for a facility operating license filed by Northern States Power Company (the applicant) complies with the provisions of the Atomic Energy Act of 1954, as amended, and the Commission's rules and regulations and that there is reasonable assurance the activities authorized by the license can be conducted without endangering the health and safety of the public. The Advisory Committee on Reactor Safeguards (ACRS) received a letter during its review of the Prairie Island plant in October 1972 that raised significant questions regarding the capability to achieve a safe plant shutdown following a postulated rupture of a steam pipe in the auxiliary building.

The Regulatory staff reviewed the location of steam pipes relative to engineered safety features in the auxiliary building and the consequences of a postulated rupture of the pipe, as presented in the applicant's letter of November 6, 1972, and the attached response to the staff's questions. The staff concluded that design changes would be required at the Prairie Island plant to provide for safe shutdown following a postulated steam pipe rupture. In its December 12, 1972; January 11, 1973; and February 9, 1973 letters to the applicant, the staff requested a description of proposed plant modifications that will be made to provide assurance that the plant can be safely shut down following the postulated rupture of any pipe carrying a high-energy fluid outside of containment, including the double ended rupture of a main steam pipe or feedwater pipe. Enclosures to the letters provided criteria and requirements for such modifications. A summary of the criteria and requirements that were incorporated in our request is set forth below:

(1) Protection of equipment necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming a concurrent and unrelated single active failure of protected equipment, should be provided from all effects resulting from ruptures in pipes carrying high-energy fluid, up to and including a double-ended rupture of such pipes, where the temperature and pressure conditions of the fluid exceed 200°F and 275 psig. Breaks should be assumed to occur in those locations specified in the "pipe whip criteria." The rupture effects on equipment to be considered include pipe whip, structural (including the effects of jet impingement) and environmental.

(2) In addition, protection of equipment necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming a concurrent and unrelated single active failure of protected equipment, should be provided from the environmental and structural effects (including the effects of jet impingement) resulting from a single open crack at the most adverse location in pipes carrying high-energy fluid routed in the vicinity of this equipment, where the temperature and pressure conditions of the fluid exceed 200°F and 275 psig. The size of the cracks should be assumed to be 1/2 the pipe diameter in length and 1/2 the wall thickness in width.

This supplement to the SER (Supplement No. 1) provides the staff's evaluation of proposed Prairie Island plant modifications that will be provided to cope with postulated high-energy pipe ruptures outside of the containments. The proposed modifications are described in Amendments 25, 28, 29, and 31. The evaluation is designated subsection 6.5 of the Safety Evaluation Report.

6.5 Emergency Shutdown Cooling Capability

6.5.1 Routing of High-Energy Lines Outside Containment and Their Relation to Engineered Safety Features

The two units of the Prairie Island plant share a common auxiliary building, a common turbine building, and a common screenhouse. The arrangement of plant equipment in these buildings is shown in FSAR Figures 1.2-1 through 1.2-11.

The buildings house the engineered safety features, auxiliary systems, components of the main steam and feedwater systems, and the electrical equipment and protection systems that control the reactors and actuate engineered safety features. The equipment associated with each unit is generally housed in compartments of the buildings that are closest to the reactor for that unit. The main controls for the plant, the control room ventilation equipment, control rod drive and trip equipment, 480-volt buses, and the diesel generators are located in separate compartments of the auxiliary building that are isolated from the other auxiliary building compartments through which high-energy fluid lines pass. Batteries, auxiliary feedwater pumps, and 4160-volt buses are located in separate compartments in the Class I (seismic) corridor of the turbine building. The diesel-driven cooling water pumps are in the screenhouse.

Each unit of the plant has two steam generators in its containment. Steam is delivered via one main steam pipe from each steam generator to the turbine. Feedwater is provided to each steam generator through one main feedwater pipe. These four main steam pipes and four main feedwater pipes for the two units pass through several compartments of the auxiliary building. Isometric sketches of these pipes for Unit 1 are shown in FSAR Appendix I, Figure I.3-1 and Figure I.4-1. These sketches show the complete routing of the high-energy fluid systems outside of containment including those systems that pass through the auxiliary building and turbine building. There are three other piping systems that transport high-energy fluids in the auxiliary building and the turbine building: the reactor coolant letdown line (FSAR Figure IA-1); the steam generator blowdown system (FSAR Figure IA-2); and the steam supply for the auxiliary feedwater pump turbines (FSAR Figure IA-3). An isometric sketch of the interconnected compartments of the Unit 1 half of the auxiliary building that would be exposed to an adverse environment after a pipe rupture is shown in FSAR Appendix IA, Figure IA-18. Figures 6.5-1 through 6.5-5 of this supplement to the SER show the routing for Unit 1 of each of the pipes carrying a high-energy fluid in the interconnected compartments of the auxiliary building. The compartments and routing for Unit 2 are similar to to those for Unit 1.

6.5.2 Equipment Required for Safe Shutdown Following the Postulated Pipe Rupture

During hot shutdown and reactor cooldown to 350°F and 400 psig, residual heat from the decay of fission products is transported from the reactor core to the steam generators by the primary coolant system and from the steam generators to the main condensers by the main steam bypass system. Condensate is returned to the steam generators by the main feedwater system. If the main steam and main feedwater systems are out of service, residual heat can be removed by the steam generator safety valves, relief valves, or the atmospheric steam dump valves with the auxiliary feedwater system providing feedwater to a steam generator.

During reactor cooldown from 350°F and 400 psig to cold shutdown conditions, residual heat from the decay of fission products is transported (a) from the reactor core to the residual heat removal (RHR) heat exchanger by the RHR system, (b) from the RHR heat exchanger to the component cooling system heat exchanger by the component cooling system, and (c) from the component cooling heat exchanger to the river by the cooling water system.

During reactor cooldown, concentrated boric acid water is added to the primary coolant system by the chemical and volume control system (CVCS) to keep the reactor subcritical. If the CVCS is not operable, concentrated boric acid water can be added by the safety injection system (SIS).

The applicant has identified the major equipment required for a safe shutdown following small and large breaks in the five high-energy fluid piping systems (FSAR Appendix I, Tables I.3-1, I.4-1, I.5-1, I.6-1, and I.7-1. In the event of a high-energy pipe rupture in the auxiliary building, the auxiliary feedwater system and the steam generator relief valves will be used for reactor cooldown and residual heat removal, and the safety injection system will be used to add concentrated boric acid water to the primary coolant system. These systems, as well as components and protection systems required for reactor trip, isolation of nonessential systems, and control of primary coolant system pressure, temperature, and water levels will be protected from pipe whip, fluid jets, adverse environments, and high compartment pressures following a postulated pipe rupture.

We have evaluated the proposed major equipment required for use in emergency shutdown cooling following this accident. The equipment has been designed to Class I (seismic) criteria. Piping systems have redundant active components such that a single failure of a valve to function or failure of a pump in any piping system would not prevent the flow of water from the river to the steam generators or the relief of steam to the atmosphere. The remote manual containment isolation valve in each auxiliary feedwater pipe that is provided for long-term post-accident use, if needed, will be locked open to ensure that at least one flow path is open to the isolated steam generators for short term post-accident cooling. Required electrical equipment can be operated by the onsite power supply if offsite power is not available. The protection systems that trip the reactors and actuate the required equipment for emergency shutdown cooling meet our guidelines for redundancy. For certain pipe breaks, it may be necessary to shut down both units of the plant. Each unit has one steam driven auxiliary feedwater pump and one motor driven pump.

The applicant will prepare operating procedures for emergency shutdown cooling using the auxiliary feedwater system and other required equipment that will be protected from high compartment pressure, pipe whip, fluid jets, and adverse environments. The procedures will identify all equipment and times required to bring the reactors to a cold shutdown (reactor coolant temperature less than 200°F). Our review of the FSAR, as amended, indicates that in order to bring the plant to a cold shutdown, some equipment in addition to that listed as required in FSAR Appendix I may be needed to (a) circulate water in the reactor coolant system and (b) remove hot water from the secondary side of the steam generators. The reactor coolant pumps and associated controls and offsite power supply can be used to circulate reactor coolant. If offsite power is not available at the time of the accident, the units can be maintained in the hot shutdown condition until offsite power becomes available. The steam generator blowdown lines can be used to remove hot water from the steam generators, but pumps and additional discharge pipes may be needed when the steam generators are depressurized. Alternately, the residual heat removal system and the component cooling system can be used to bring the plant to a cold shutdown. Both residual heat removal and component cooling systems are designed as engineered safety features with redundant active components. The pumps and heat exchangers for these two systems are in compartment "E" that will be isolated from the adverse environment following the rupture of a high-energy fluid pipe.

The applicant will submit a complete evaluation of a spectrum of postulated accidents, identifying all equipment required to operate in the post-accident mode considering a single active failure in addition to the failures due to the accident. Modifications will be made so that required equipment and systems will be protected from pipe whip, fluid jets, adverse environments, and high compartment pressures following a postulated pipe rupture.

Subject to confirmation after evaluation of the final design analysis referred to above, we conclude that the equipment required for safe cold shutdown has been adequately identified.

6.5.3 Protection Against Dynamic Effects of Pipe Whip

Protection of vital structures, systems, and components is required from possible adverse effects due to pipe impact subsequent to a postulated rupture of high-energy fluid system piping. Such protection is provided by a system of structural steel restraints anchored firmly to concrete or structural members of the building, with sufficient strength to limit and control the potential for pipe whipping as the contained fluid discharges from the pipe break.

The design criteria used in determining the pipe systems to be restrained, design basis break locations, and orientation are consistent with the AEC criteria and requirements identified in the introduction to this supplement. Piping systems operating at 275 psig or greater at temperatures equal to or greater than 200°F were considered as high-energy fluid systems by virtue of the inherent potential of the contained fluid to possess sufficient energy to cause pipe whipping in the event of a pipe break.

Design basis break locations in each piping run of the high-energy lines were assumed in accordance with the AEC pipe whip criteria which identify the more highly stressed locations in the piping during normal plant operation. At these locations where the liklihood of structural failure is more probable because the pipe material is subjected to higher stresses relative to other sections of the piping, a complete pipe severance (circumferential break) was assumed, as well as an axial pipe split (longitudinal break) for the purpose of calculating the magnitude and direction of fluid jet thrust and reactions which subsequently develop as the fluid discharges.

The applicant's design review of high-energy fluid systems identified only one location where the calculated stress (36,700 psi) qualified the location as a design basis break point. In fact, the calculated stress levels in the remainder of the piping were at least 20 percent below the yield strength and 40 percent below the ultimate strength of the pipe materials. Despite these low stress levels, the applicant selected at least two intermediate break locations in each pipe run, as an added measure of conservatism, for the purpose of designing restraints to provide additional protection.

Although adequate analyses were performed to determine the forces and response expected at break locations, the applicant agreed to perform a more detailed non-linear elastic-plastic analysis using a lumped mass mathematical model of the piping and restraint system in order to confirm the degree of conservatism when taking into account load-time variation of jet forces, including impact effects of pipe against restraint steel members.

Design basis leakage cracks (other than pipe breaks) were also assumed to occur anywhere along the pipe run in order to make certain that, in the event of a high-energy line leak, the equipment in the vicinity of the pipe crack could be verified as capable of withstanding the resulting steam and water environment created by fluid issuing from the crack and perform the intended safety function.

The placement of rigidly anchored structural steel restraints to accommodate the forces developed at design basis break locations assures added protection for breaks, although unlikely to occur, at locations on either the upstream or downstream side of the design basis break location. Added protection of structures, systems, and components is provided by jet impingement barriers, in the form of steel plates, and encapsulation sleeves. An encapsulation sleeve which consists of an independently supported steel jacket around the design basis break location will not only reduce the rate at which high-energy fluid could escape from the break and reduce the pressurization of the surrounding building compartment, but also constrains the broken pipe from whipping. Design stress limits for the imposed loads on the encapsulation sleeves are consistent with the acceptable levels prescribed by the ASME Nuclear Power Plant Components Code Section III for components subject to faulted plant conditions.

Subject to the dynamic analysis of the piping systems, we find that the structures, systems, and components required for the safe shutdown of the plant in the event of pipe breaks will be adequately protected against the dynamic effects of pipe motions and jet forces due to rupture of high-energy fluid lines outside containment.

6.5.4 Protection Against High Compartment Pressure

The unrestricted flow of high-energy fluid from design basis breaks of large piping systems in small compartments of the auxiliary building that do not have large vent areas would result in pressures in the compartments that exceed the structural capability of the compartment walls. Design basis breaks of high-energy fluid pipes in the turbine building would not result in excessive building pressure because of its large volume and the design of its exterior walls to function as blowout panels. This subsection describes our evaluation of the applicant's proposed modifications to restrict flow from postulated pipe ruptures and provide adequate compartment venting and the capability of compartments to withstand the pressure resulting from blowdown of high-energy fluids from the postulated pipe ruptures.

Large pipe breaks including the double ended rupture of the largest pipes in a system and small leakage cracks up to the design basis crack size have been considered for each of the five high-energy fluid systems. Rupture of the 30-inch diameter main steam pipe at the selected design basis break locations is postulated to result in blowdown of all the steam and water in one steam generator. Rupture of a 16-inch diameter main feedwater pipe at design basis break locations in the auxiliary building would not result in blowdown of a steam generator because reverse flow in the feedwater line would be prevented by the check valve inside containment. Rupture of the 3-inch diameter steam supply pipe at selected locations for the auxiliary feedwater pump turbine, or the 2-inch diameter steam generator blowdown line would not result in significant compartment pressures but could create an adverse environment. Rupture of the 2-inch diameter reactor coolant system letdown line at selected locations could also create an adverse environment.

The location of design basis breaks in the five piping systems identified as high-energy fluid systems in accordance with our criteria (FSAR Figures I.3-1, I.4-1, IA-1, IA-2, and IA-3) have been discussed in Section 6.5.3 of this supplement. The compartments of the auxiliary building in which design basis breaks are postulated to occur are shown in Figures 6.5-1 through 6.5-5 of this supplement. Design basis leakage cracks are assumed to occur anywhere in the surface of the high-energy fluid system. A design basis break of either the main pipe run or of a branch pipe run is designated in these figures by a Roman numberal and a capital letter.

The portion of the Unit 1 main steam piping system which attained the calculated stress levels designated for design basis breaks is located in compartment "Y" of the auxiliary building (see Figure 6.5-1 of this supplement). New walls will be built within existing auxiliary building compartments (north walls of compartments "Y" and "X" in Figure 6.5-1) to assure isolation of the postulated breaks in the auxiliary building. Existing openings in the exterior walls (south walls of compartments "Y" and "X") of the isolated compartments or blowout panels (if required to obtain adequate vent area) will be used to provide pressure venting. An encapsulation sleeve will be provided to limit the steam blowdown rate into compartment "Y." The steam is vented to compartment "X" through a floor grating. There is no equipment in compartments "X" or "Y" that would be required to bring the plant to safe cold shutdown as a result of the steam line rupture in compartment "Y."

Design basis break locations of the steam branch pipe runs are also shown in Figure 6.5-1 of this supplement. Encapsulation sleeves will be installed over the branch pipe at the locations of the design basis breaks and the pipe will be designed to prevent separation of the ruptured pipe from the sleeves. The sleeves will be designed to restrict the flow from a ruptured pipe so that overpressurization of the compartments will not occur.

The applicant has calculated the pressure and temperature transients in compartments "Y" and "X" with an encapsulation sleeve to limit the blowdown area to 75 square inches. The calculated peak pressure in both compartments is less than 2.5 psig and the peak occurs in about 0.5 seconds. Compartment "Y" was calculated to have sufficient structural capability to withstand a compartment pressure of 7.0 psig. Compartment "X" has a calculated structural capability to withstand 5.0 psig. Peak calculated pressures in other compartments are: 0.75 psig in compartment"A;" 1.1 psig in compartment "B;" and less than 0.5 psig in compartment "C." We have verified by independent analyses using a modified CONTEMPT-LT computer code that the applicant's peak pressure calculation for compartments "Y" and "X" is conservative.

The applicant has designed Class I (seismic) structures for the load combinations and working stresses as indicated in FSAR Tables B.6-1; B.6-2; and B.6-3. The applicant will include as a part of the final design analysis a review and evaluation of the structural adequacy of any new structures or modifications to existing structures required for protection against the consequences of the postulated high-energy pipe rupture. This analysis will use load combinations in FSAR Appendix IA Table IA-1 and will include dead load, live load, earthquake load, jet or pressure load, equipment reactions, and pipe whip load. For these combinations, the limiting design stresses will be $f_c = 0.85\ f'_c$ for concrete and $f_s = 0.90\ f_y$ for reinforcing bars and structural steel. The handling of concrete shear stresses will be in accordance with the ACI 318-71 code.

The bellows and the seals in penetrations of the shield building have been tested for a differential pressure of (internal or external pressure) up to 20 psi which is substantially greater than the 2.5 psig pressure resulting from a design basis pipe break. The integrity of these seals and bellows can be expected to be assured and therefore the shield building will function as required.

Subject to the acceptance of the final design analysis, we conclude that the descriptions of modifications and design procedures for modifications in the FSAR, as amended, provide assurance that the modified structures, when completed, will withstand the pressures and jet loadings resulting from the postulated rupture of a high-energy fluid pipe outside the containment.

6.5.5 Protection Against Adverse Environment

The applicant states it will protect the required equipment, including electrical equipment and protection systems, against excessive temperature and moisture due to design basis leakage cracks in high-energy fluid systems and against excessive temperatures or forces due to high-energy fluid jets. Most of the equipment is protected because it will be in compartments of the buildings that can be isolated from the adverse environments by using existing structures (see Section 6.5.1 of this supplement). Compartment "E" (Figure 6.5-1 of this supplement) containing the safety injection pumps will also be isolated from the adverse environment by enclosing the stair wells. Motor control centers for the main feedwater isolation valves and starters for valves now in compartment "B" will be moved to compartment "E."

The normal ventilation systems will be evaluated and included by the applicant in the final design analysis to assure isolation of compartments that contain required equipment. The auxiliary building ventilation ducts are presently rectangular in cross section. If final design compartment pressure analyses show that the differential pressure exceeds the structural capability of this ducting, they will be replaced with cylindrical ducts. Redundant isolation dampers capable of withstanding the post-accident pressure will be provided to isolate those compartments that contain equipment required for a safe shutdown. Dampers will be closed by temperature signals from sensors located in the ducts. Compartments that will be provided with automatic isolation of the ventilation duct include the Class I (seismic) portion of the turbine building that houses the auxiliary feedwater pumps, and compartments "X" and "Y."

Blowout panels and flexible seals will be provided where necessary to isolate a steam environment resulting from a design basis pipe break. The applicant has provided test results which demonstrate that these structural elements will perform as intended.

The control room has its own independent, heated, filtered, and humidified ventilation system. The post-accident environment in the control room will be unaffected by the temperature, humidity, and airborne activity in other compartments or buildings.

The applicant states that the power and control cables used in the auxiliary building have the same environmental qualifications as those used in the containments. They have been prototype-tested to meet LOCA accident conditions inside containment. Terminal connections for cables that will be in a steam environment will be sealed with epoxy varnish. Fuses for required equipment will be enclosed in a sealed box that has been environmentally qualified by tests. Required instruments that are exposed to the adverse environment will be modified so they are the same as those that were qualified for the LOCA conditions inside containment.

The temperature and force of jets from design basis cracks in the high-energy fluid pipes have been calculated by the applicant as a function of distance from the pipe. Cable trays and structures, such as closed doors, that are required for safe shutdown will be protected by jet impingement barriers if they are close enough to be damaged by the jet temperature or force. The locations of impingement barriers for the high-energy lines are shown on FSAR Figures I.3-1, I.4-1, IA-1, IA-2, and IA-3. In compartment "C," a jet impingement sleeve will be put around the main steam pipe to protect nearby cables and cable trays (see Figure 6.5-1 of this supplement).

The major equipment required to prevent fuel damage and to safely bring the units to cold shutdown has been identified, and equipment located in spaces exposed to an adverse environment either has been or will be qualified by tests. As noted in Section 6.5.1 of this supplement, the applicant will submit a complete evaluation of a spectrum of postulated pipe ruptures during hot shutdown and power operation. These analyses will identify the manual and automatic operations of plant equipment and the time at which these actions are required.

Subject to the acceptance of the final design analysis, we find that with the satisfactory implementation of the design including fluid jets resulting from a high-energy fluid line failure, will not result in the loss of the equipment necessary

6.5.6 Conclusions

Subject to the acceptance of the final design analysis of plant modifications and the dynamic analysis of the piping systems, we find that the proposed plant modifications will assure a safe cold plant shutdown following the postulated rupture of any pipe carrying a high-energy fluid outside the containment. We will review the final design analyses and the installation of the modifications prior to plant startup tests to assure that the criteria have been appropriately implemented.

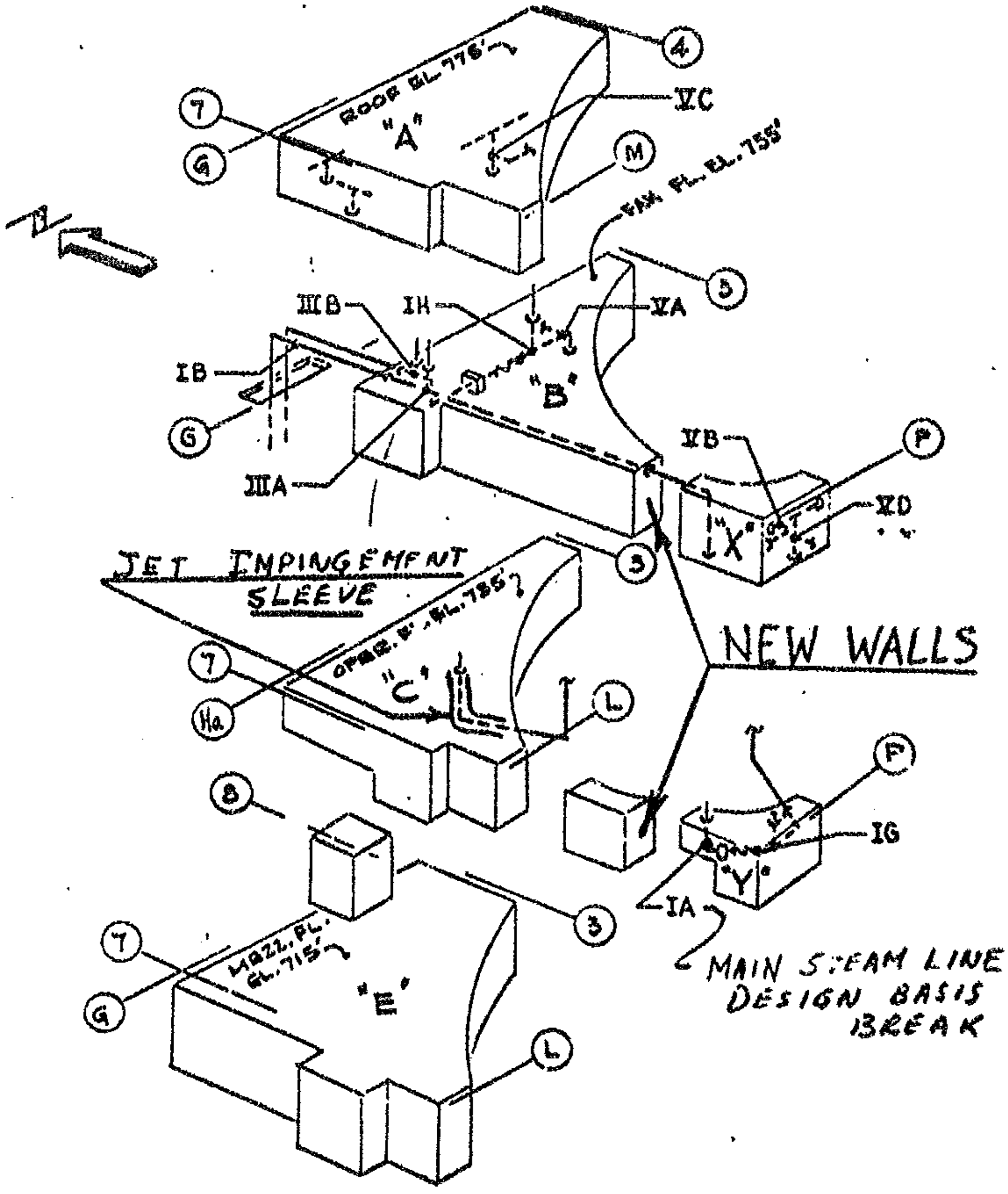


FIGURE 6.5-1
MAIN STEAM PIPING
NSP-PRAIRIE ISLAND UNIT 1

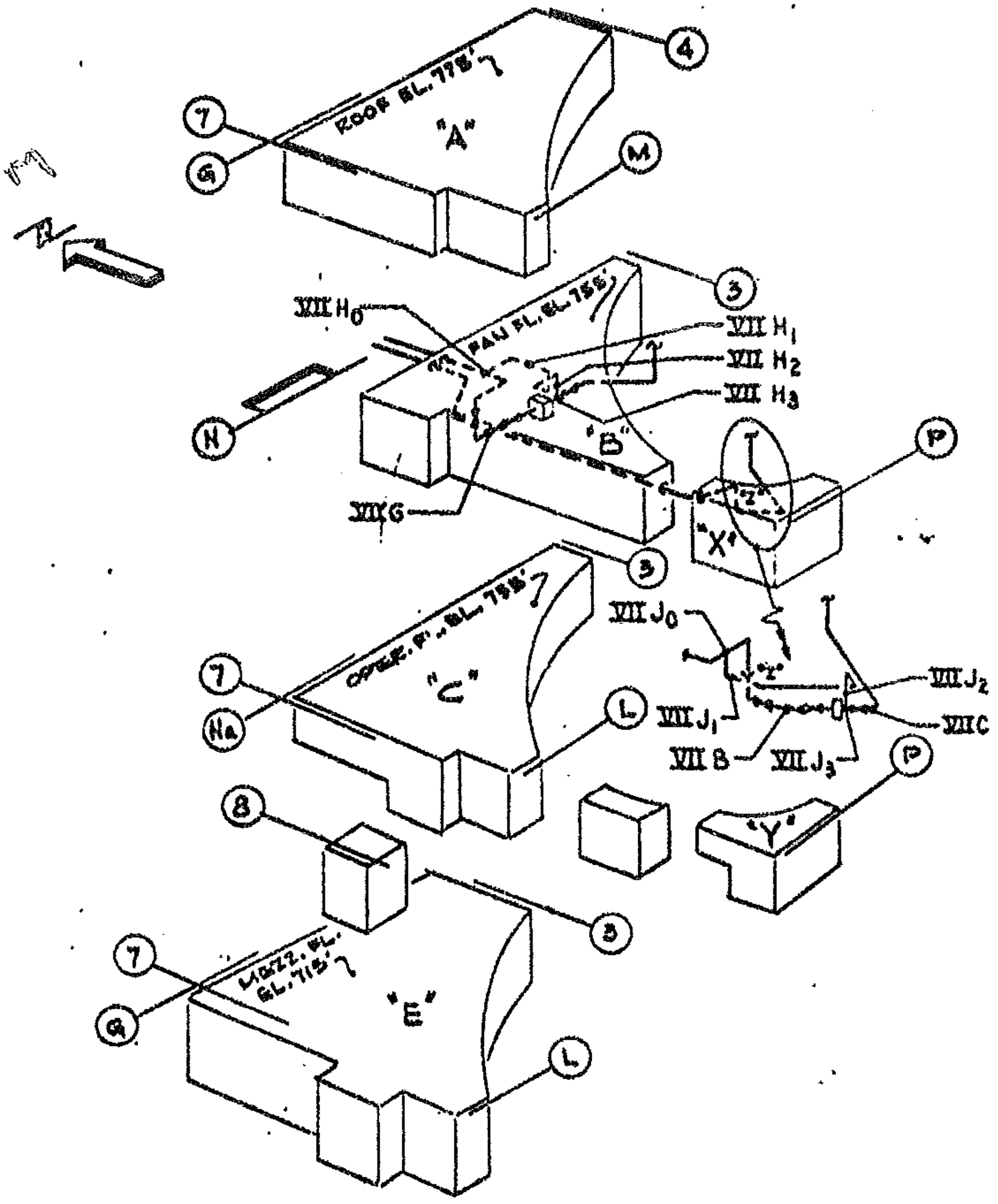


FIGURE 6.5-2
FEEDWATER PIPING
NSP-PRAIRIE ISLAND UNIT 1

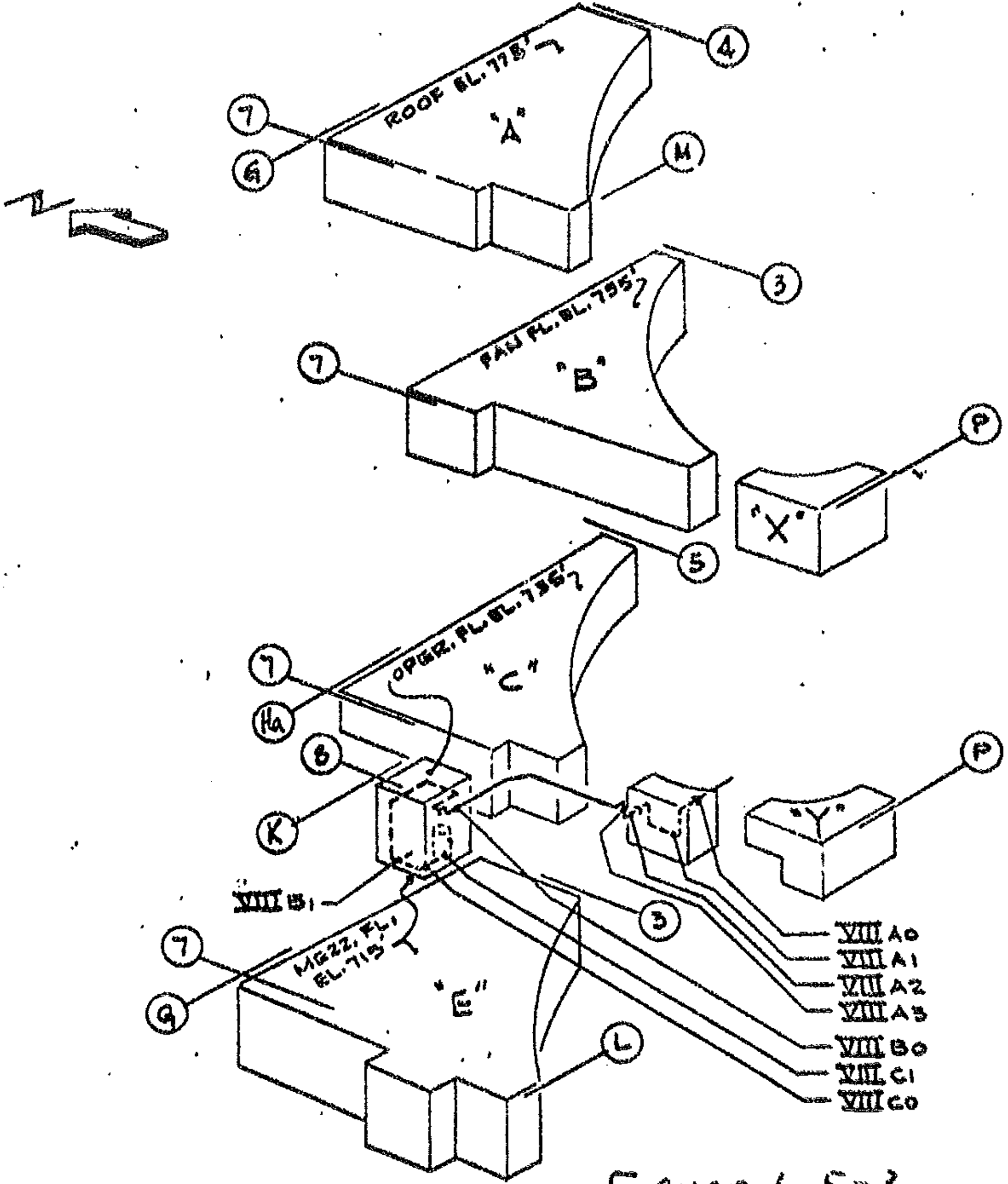


FIGURE 6.5-3
LETDOWN LINE
NSP - PRAIRIE ISLAND UNIT 1

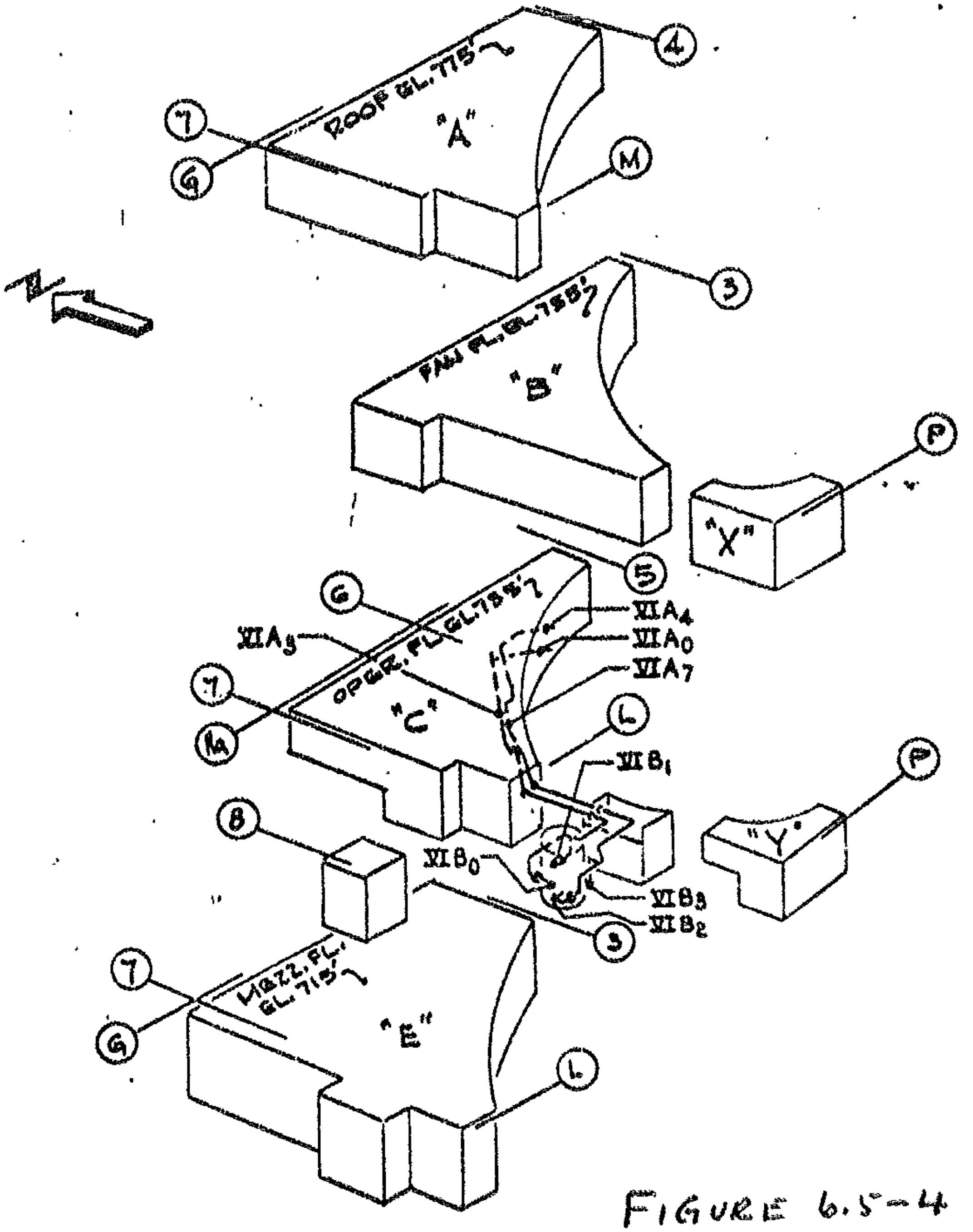


FIGURE 6.5-4

STEAM GENERATOR BLOWDOWN LINE
NSP-PRAIRIE ISLAND UNIT 1

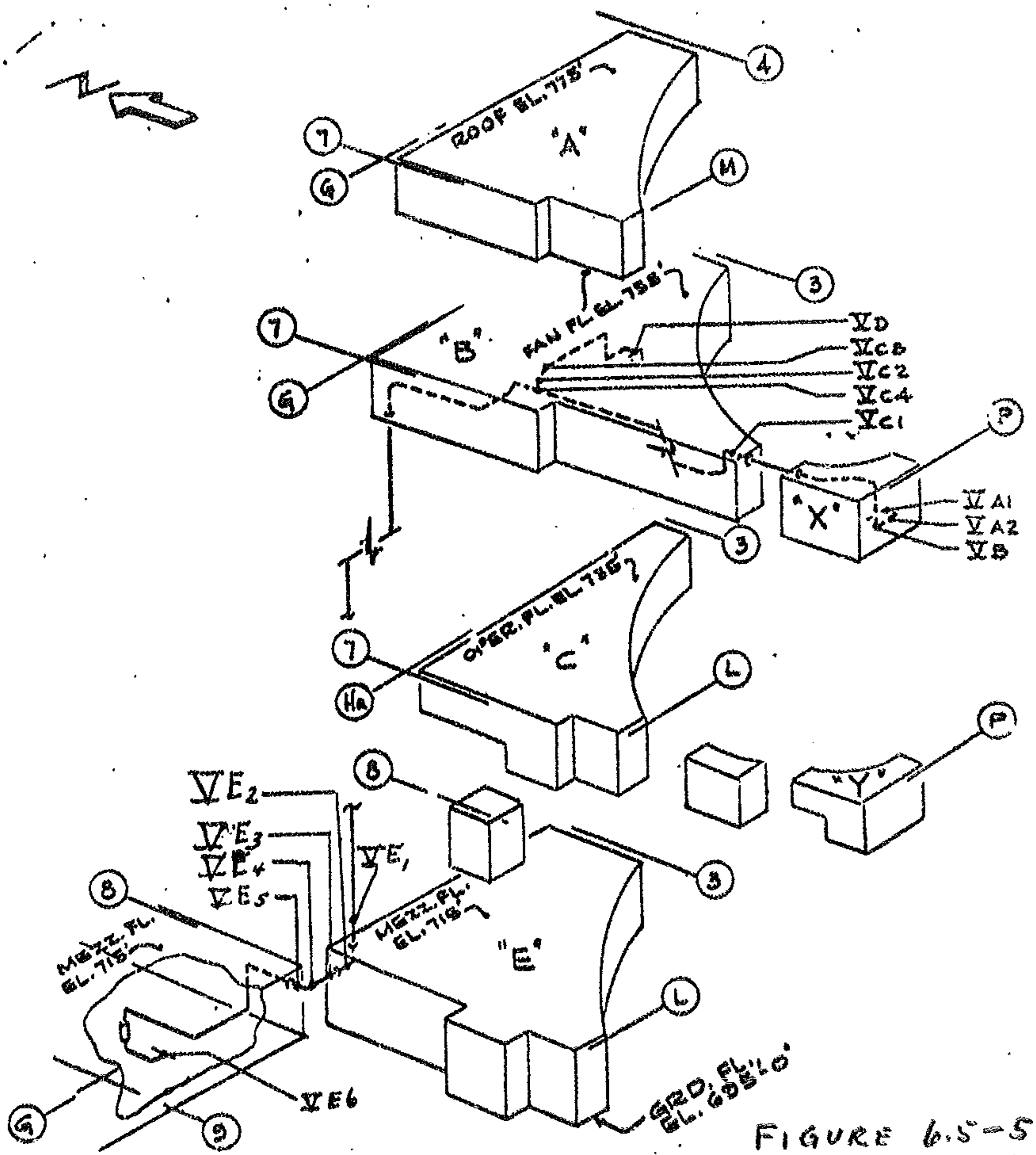


FIGURE 6.5-5
STEAM FOR AUX. F.W. PUMP
NSP - PRAIRIE ISLAND UNIT 1

Nuclear Services Corporation

ATTACHMENT I

EVALUATION OF PIPE RUPTURE - METHODS OF ANALYSES

Blowdown forces imposed on the piping system and the resultant effects of pipe whip are determined using the methods outlined below.

A. Blowdown Forces

Blowdown forces resulting from pipe rupture are determined using the computer program PRTHRUST (Reference 1). The program uses as a basis RELAP3 (Reference 2), the AEC's presently accepted loss of coolant accident program (as per Reference 3). PRTHRUST computes and plots the force-time history curve of the reaction loads resulting from a circumferential or longitudinal pipe break for subcooled liquid, flashing liquid and steam systems.

The system of interest is modeled as an assembly of volumes connected by flow paths. In a flow path there can be inserted a valve, check valve, or pump. The program solves the transient energy, momentum, and state equations for the volumes and flow paths. The program has the capability to solve the state equation for subcooled water, two phase steam-water mixtures, and superheated steam. The ASME Steam Tables (Reference 4) are tabulated within the program, and a table lookup is used to determine the state within each volume. An optional bubble separation model can be used to represent a vapor phase above a liquid phase (e.g., steam generator). The program calculates the flow in each junction using both an inertial model (no-choking) and Moody's critical flow model (choking) (Reference 5). The lower of the flows calculated from the two above models is limiting and thus taken as the actual flow.

The program will allow for the modeling of special component characteristics as applicable in the different systems. Leaks can be opened instantaneously or as a function of time. Pumps can continue to operate or can coastdown. Valves can be opened or closed, and check valves follow a prescribed pressure loss, flow characteristic.

The break force is calculated using the one dimensional momentum equation. The force on the piping consists of the following three forces:

Pressure force $= (P_t - P_e)A$ (zero for nonchoking flow)

Momentum force $= \frac{\rho A V^2}{g}$

Momentum change $= \frac{\Delta(mV)}{\Delta t\, g}$ (zero for steady state)

where: P_t = throat pressure
P_e = exit pressure
A = break area
ρ = density
V = velocity
g = 32.2 lbm ft/lbf sec^2
m = mass
t = time

The algebraic sum of the three forces is the resultant force on the broken piping.

B. Effects of Pipe Whip

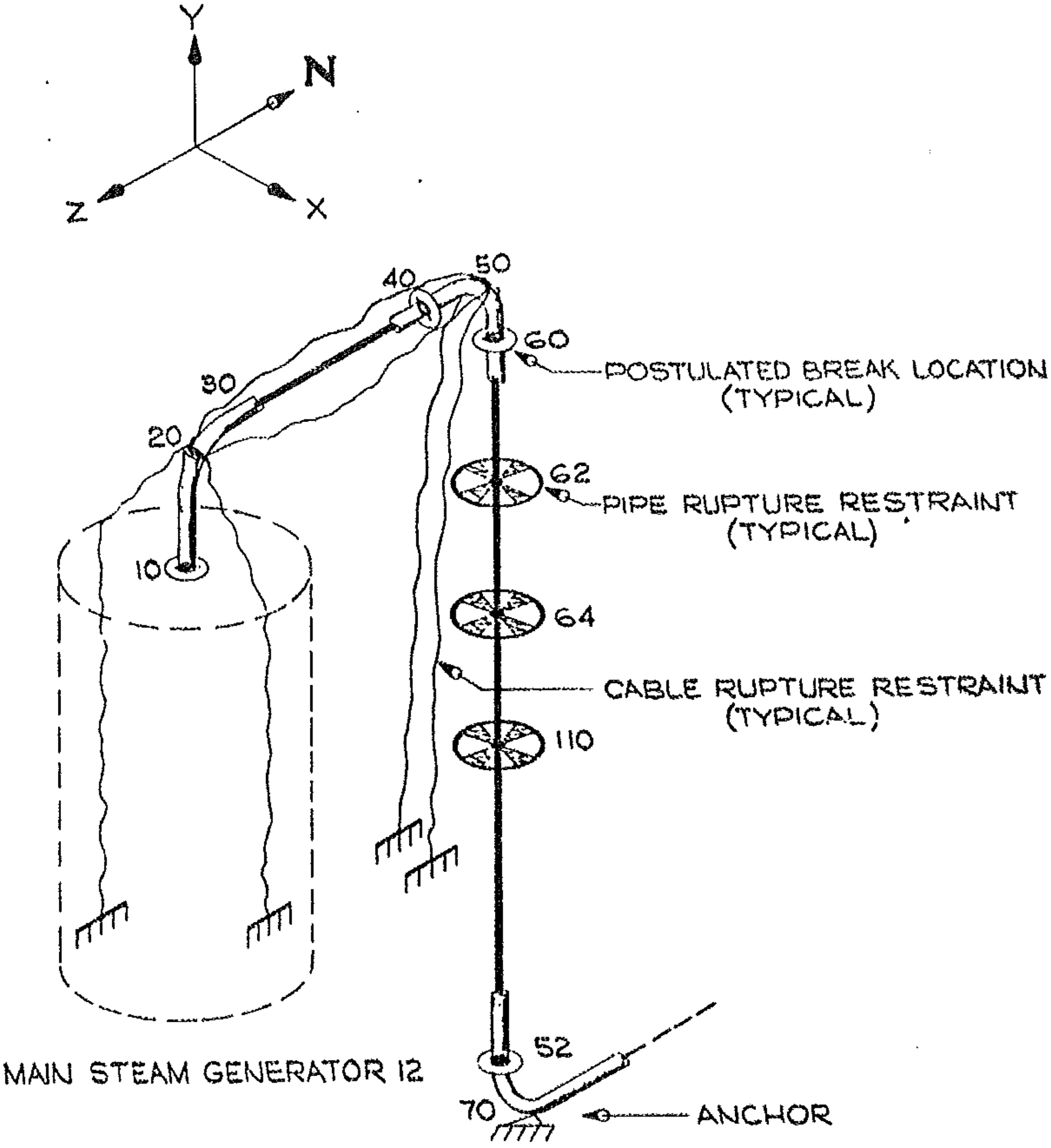
Computation of piping system response to pipe rupture forces are determined using the computer program PIPERUP (Reference 6). PIPERUP is an adaptation of the finite element method to the specific requirements of pipe rupture analysis. A dynamic response - time history of the piping system is determined which includes elastic-plastic pipe behavior and non-linear effects of pipe rupture restraints.

The piping system is modeled as an assemblage of straight and curved beams (elbows) connecting discrete nodal points. Weight of the piping system (including offset weights of valve motor operators) is "lumped" at selected nodal points. The blowdown force vs. time history as developed by PRTHRUST (Reference 1) is then applied as an excitation force to the appropriate piping node point. Dynamic response of the piping system is computed at iterative increments of time and includes forces, moments, deflections and rotations at each node. The resulting bending and torsional moments at each node are used to predict both initial yielding (at which time the elastic modulus at the affected point is replaced by the strain hardening modulus) and ultimate load (i.e., formation of a plastic hinge; after which the modulus is set to a very low value). In situations where stress reversal occurs, an isotropic strain hardening model is used. The strain in plastic hinges and deflections of node points are used to identify pipe trajectory.

Pipe rupture restraints are modeled in PIPERUP with an initial gap, and then both elastic and plastic moduli. At each time step, the programs will thus determine gap closure, elastic or plastic deformation, and the resulting impact load.

References:

1. Nuclear Services Corporation, "PRTHRUST: Computer Code for Pipe Rupture Thrust Calculations", NSC-A4, dated February 2, 1973.
2. W. H. Rettig, et. al., "RELAP 3 - A Computer Program for Reactor Blowdown Analysis", IN-1321 (June 1970). Also supplement of June 1971.
3. "Interim Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Power Reactors", Interim Policy Statement, Atomic Energy Commission Federal Register, June 20, 1971, page 12217.
4. C. A. Meyer et. al., "1967 ASME Steam Tables - Thermodynamic and Transport Properties of Steam," New York: The American Society of Mechanical Engineers, 1967.
5. F. J. Moody, "Maximum Flow Rate Of a Single Component, Two-Phase Mixture", J. Heat Trans. ASME, 87 n 1 (February 1965) pp 134-142.
6. Nuclear Services Corporation, "PIPERUP: A Computer Program for Analysis of Piping Systems Subject to Pipe Rupture Loads", dated December 1972.

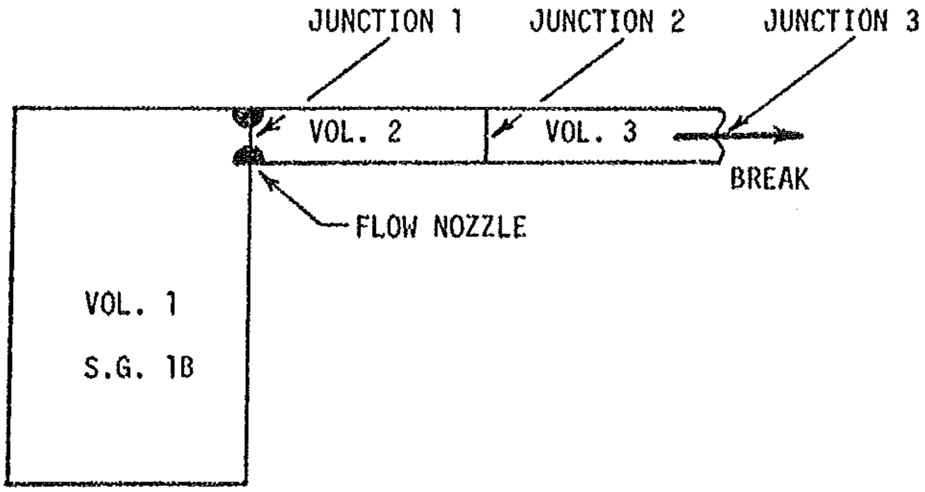


Nuclear Services Corporation
CAMPBELL, CALIFORNIA

FIGURE 1 - POSTULATED BREAK LOCATIONS FOR MAIN STEAM LINE 12 - INSIDE CONTAINMENT

FIGURE 2 ANALYTICAL MODEL FOR BLOWDOWN ANALYSIS - FULL LOAD CIRCUMFERENTIAL BREAK AT NODE 52 (SG1B SIDE)

A. CONFIGURATION



B. ASSUMPTIONS - FULL LOAD

1. P = 750 PSIA

2. T = 510.8°F

3. BREAK AREA = 4.264 FT^2

4. BREAK OPENS IN 0.001 SECS.

PRAIRIE ISLAND UNIT 1 MS - FULL LOAD

CIRCUMFERENTIAL BREAK AT NODE 52 (SG1B SIDE)

VOLUME DATA

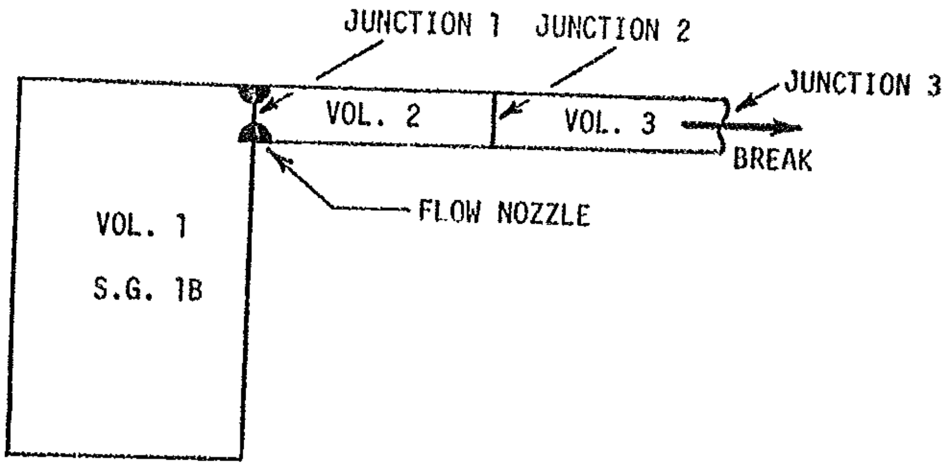
VOL. NO.	DESCRIPTION	VOLUME (FT^3)	PRESSURE (PSIA)	TEMP. (^oF)	QUALITY
1	STEAM GENERATOR 1B	5868 TOTAL 4148.3 STEAM	750	----	0.07
2	LINE FROM NODE 10 TO NODE 50	76	750	----	1.0
3	LINE FROM NODE 50 TO NODE 52	266.3	750	----	1.0

JUNCTION DATA

INLET VOL. NO.	OUTLET VOL. NO.	JUNCTION INERTIA L/A (FT^{-1})	MINIMUM FLOW AREA (FT^2)	INITIAL FLOW (LBM/SEC)	$\frac{fL}{D}$	COMMENTS
1	2	2.49	1.396	983.3	2.29	FLOW LIMITER
2	3	9.42	4.264	983.3	0.231	
3	0	7.32	4.264	983.3	1.0	BREAK

FIGURE 3 ANALYTICAL MODEL FOR BLOWDOWN ANALYSIS - HOT STANDBY CIRCUMFERENTIAL BREAK AT NODE 52 (SG1B SIDE)

A. CONFIGURATION



B. ASSUMPTIONS - HOT STANDBY

1. P = 1020 PSIA
2. T = 547°F
3. BREAK AREA = 4.264 FT^2
4. BREAK OPENS IN 0.001 SECS.

PRAIRIE ISLAND UNIT 1 MS - HOT STANDBY

CIRCUMFERENTIAL BREAK AT NODE 52 (SG1B SIDE)

VOLUME DATA

VOL. NO.	DESCRIPTION	VOLUME (FT^3)	PRESSURE (PSIA)	TEMP. (^oF)	QUALITY
1	STEAM GENERATOR 1B	5868 TOTAL 2357.4 STEAM	1020	----	0.0285
2	LINE FROM NODE 10 TO NODE 50	75	1020	----	1.0
3	LINE FROM NODE 50 TO NODE 52	266.3	1020	----	1.0

JUNCTION DATA

INLET VOL. NO.	OUTLET VOL. NO.	JUNCTION INERTIA L/A (FT^{-1})	MINIMUM FLOW AREA (FT^2)	INITIAL FLOW (LBM/SEC)	$\frac{fL}{D}$	COMMENTS
1	2	2.49	1.396	0.0	2.29	FLOW LIMITER
2	3	9.42	4.264	0.0	0.231	
3	0	7.32	4.264	0.0	1.0	BREAK

FIGURE 4

PI UNIT 1 MS BLOWDOWN FORCE - FULL LOAD

CIRCUM BREAK NODE 52 (SG1B SIDE)

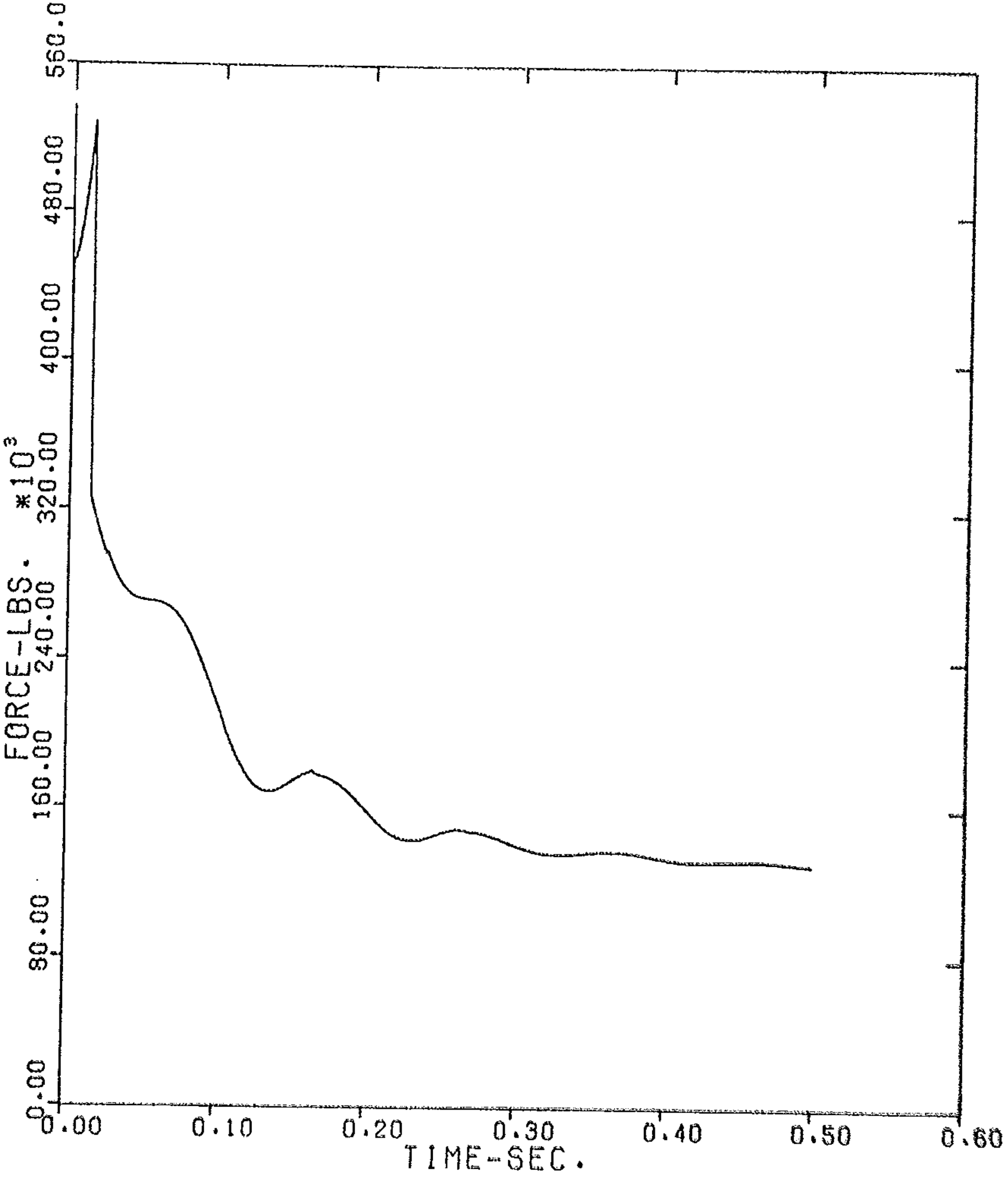
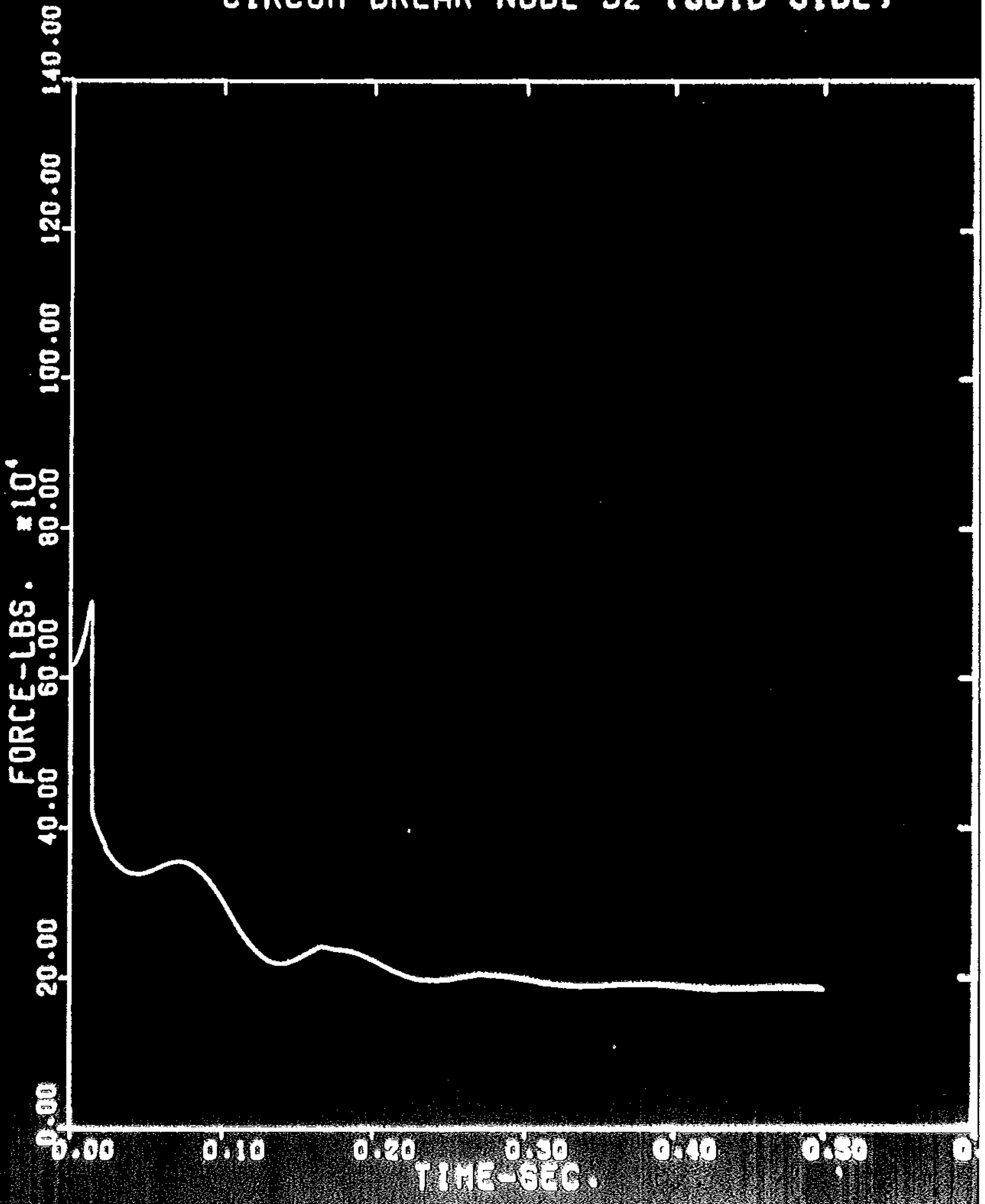


FIGURE 5

PI UNIT 1 MS BLOWDOWN FORCE-HOT STANDBY

CIRCUM BREAK NODE 52 (SG1B SIDE)



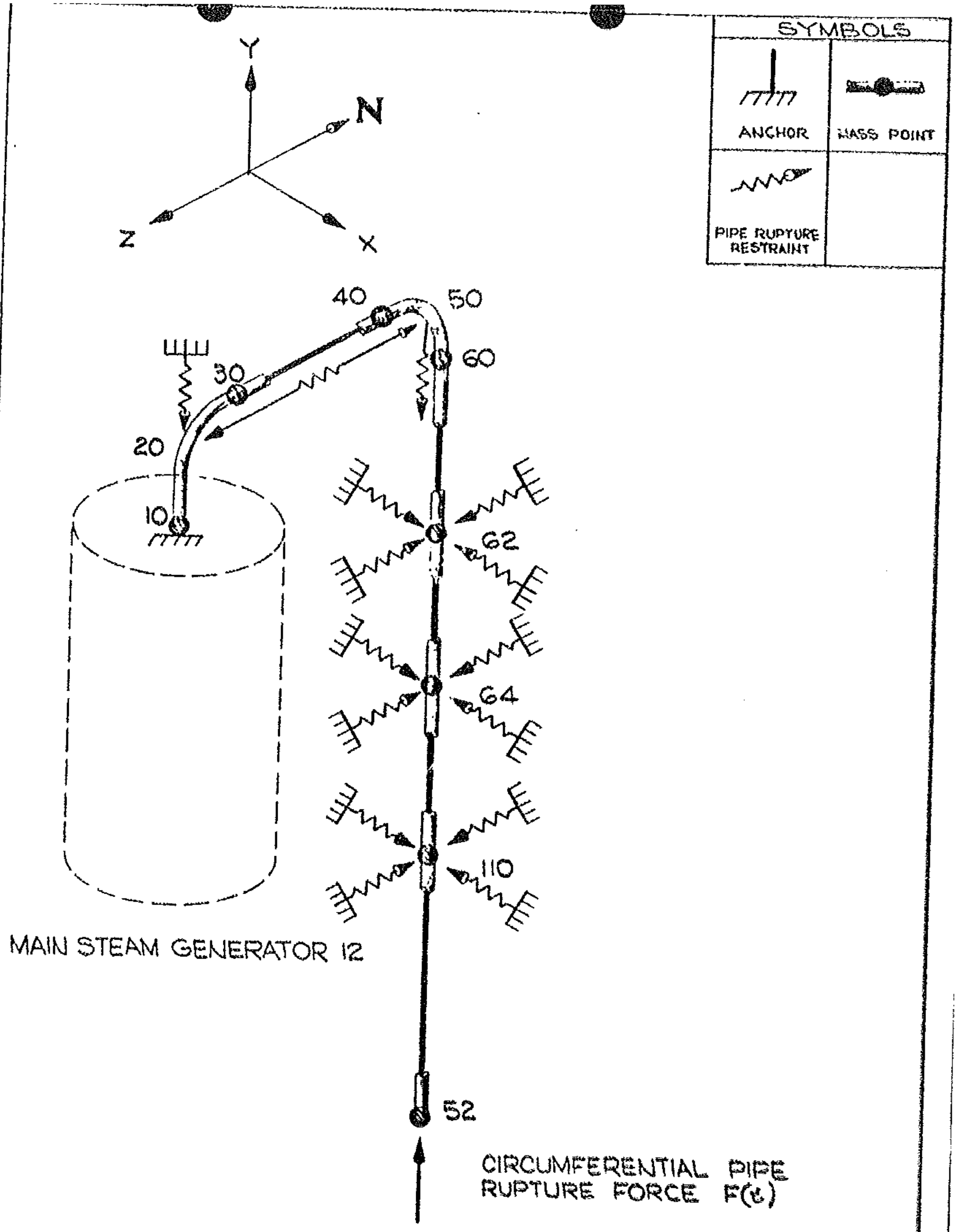


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FIGURE 6 - ANALYTICAL MODEL FOR PIPE WHIP ANALYSIS