RETURN TO

DAVE WATERS

POSTULATED PIPE FAILURE ANALYSIS

OUTSIDE OF CONTAINMENT

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PREFACE

POSTULATED PIPE FAILURE ANALYSIS OUTSIDE OF CONTAINMENT

In AEC-DOL's letter to CPL of December 5, 1972 (Mr. Giambusso to Mr. Jones) it was stated that the Regulatory Staff's continuing review of reactor power plant safety indicates that the consequences of postulated high energy pipe failures outside of the containment structure, including the rupture of a main steam or feedwater line, need to be adequately documented and analyzed by licensees and applicants, and evaluated by the staff as soon as possible. Additional detailed information was requested by AEC-DOL's letter to CP&L of March 29, 1973 (Mr. Schemel to Mr. Jones).

The information presented in this report is in response to the above requests for information.

1.0 DESCRIPTION OF HIGH ENERGY SYSTEMS

1.1 DEFINITION

High energy piping systems are defined as those which have a service temperature above 200°F and a design pressure above 275 psig. The plant operational conditions, under which this definition applies, includes normal reactor operation and the operational basis earthquake defined as an upset condition.

1.2 IDENTIFICATION OF SYSTEMS

The systems of the H. B. Robinson Unit No. 2 Generating Station which fall under the definition listed above are:

a.) Main Steam System

b.) Feedwater System

c.) Steam Generator Blowdown System

d.) Chemical and Volume Control System

e.) Steam Supply to Auxiliary Feedwater Pump Turbine

2.0 CRITERIA FOR PROTECTION AGAINST PIPE RUPTURE

2.1 DEFINITION OF A PIPING SYSTEM

A piping system is defined as having pressure-retaining components consisting of straight or curved pipe and pipe fittings such as elbows, tees, reducers and valves. The boundaries of a system are described in terms of a piping run. A piping run interconnects components such as pressure vessels, tanks, pumps and rigidly fixed valves or structural anchors that are designed to restrain pipe movement. A branch run differs from a main piping run only in that it originates at a piping intersection as a branch of the main pipe run.

2.2 DESIGN BASIS BREAKS AND CRACKS

Design Basis breaks and cracks in high-energy piping systems are defined in this section. The criteria for considering the effects of pipe whip, jet impingement, jet reaction differential pressure and elevated temperatures are considered in subsequent sections.

2.2.1 Break Location Based on High Stress Points

There is no ASME Section III Code Class 1 piping outside containment for the H. B. Robinson Unit No. 2 Nuclear Generating Station.

The criteria used to determine the design basis piping break locations are as follows:

ASME Section III Code Class 2 and 3 or ANSI B 31.1 piping breaks are postulated to occur at the following locations in each piping run or branch run:

1) The terminal ends

2) At intermediate locations where:

a) the circumferential or longitudinal stresses derived on an elastically calculated basis under the loadings associated with seismic events and operational plant conditions exceed 0.8 $(S_h + S_A)$* or the expansion stresses exceed 0.8 S_A or

b) a high-stress point has been calculated (minimum of 2 points).

Breaks are postulated only in systems whose operating pressure exceeds 275 psi <u>and</u> whose temperature exceeds 200°F.

2.2.2 Break Size and Orientation

Once a design basis break location has been established, as defined above, the break orientation and size depend upon the following additional conditions:

Longitudinal breaks in piping runs or branch runs will be examined for pipes of 4" nominal diameter and larger. A longitudinal break is parallel to the pipe axis and

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*S_h is the stress calculated by the rules of NC-3600 and ND-3600 for Class 2 and 3 components, respectively, of the ASME Code Section III Winter 1973 Addenda.

S_A is the allowable stress range for expansion stress calculated by the rules of NC-3600 of the ASME Code, Section III, or the ANSI Standard Code for Pressure Piping, ANSI B31.1.0-1967.

oriented at any point around the pipe circumference. The break area is equal to the effective cross-sectional flow area upstream of the break location with the length of the break equivalent to twice the inside pipe diameter. Dynamic forces resulting from such breaks are assumed to cause lateral pipe movements in the direction normal to the pipe axis.

Circumferential breaks must be considered in piping runs and branch runs exceeding a nominal 1-inch diameter. A circumferential break is perpendicular to the pipe axis, and the break area is equivalent to the cross-sectional area of the ruptured pipe. Forces resulting from such breaks are assumed to separate the piping axially one pipe diameter from its original position in any direction, thereby, allowing formation of a free jet unless otherwise physically restrained. In addition, whipping is postulated if a plastic hinge mechanism is formed as a result of the deformed geometry loading effects.

2.2.3 Design Basis Crack

A design basis crack is defined as a single open crack of a size of one-half the pipe diameter in length and one-half the pipe wall thickness in width oriented in any direction. Evaluation will be on the basis of jet impingement effects and differential pressure, temperature and humidity effects on structure and systems necessary for safe shutdown of the nuclear plant.

2.2.4 Crack Location

Where high-energy pipes are routed in the vicinity of structures and systems necessary for safe shutdown of the nuclear plant, a crack in the pipe system will be postulated. The criteria for evaluating the effects of jet impingement and resulting steam-air environment will be considered and are discussed in subsequent paragraphs.

Cracks are postulated only in systems whose operating pressure exceeds 275 psi <u>or</u> whose temperature exceeds 200°F.

2.3 CRITERIA FOR PIPE RUPTURE INDUCED LOADS

2.3.1 Pipe Whip

The reaction load resulting from pipe break will have the duration and initial conditions to adequately represent the jet stream and the system pressure characteristics.

The location of breaks for pipe-whip considerations are those defined in Section 2.2. The loads induced from pipe breaks will include the effects of any line restrictions; for example, flow limiters, between the pressure source and the break location.

The effect of the break-opening geometry on the magnitude of reaction loads are incorporated in the analysis. Unless otherwise determined by a more rigorous analysis, which includes development of specific forcing functions for the piping system being considered, pipe break loads are defined as equivalent static loads applied at the point of break.

The direction of load is perpendicular to the break plane with a magnitude equal to the system operating pressure times the postulated break area.

Pipe whip is assumed to occur when a plastic hinge mechanism is formed. The dynamic characteristics of the whipping pipe will be determined by the geometry of the postulated break and the unbalanced break forces causing plastic hinge mechanism rotation. Membrane equilibrium of a hinged piping system as a three hinge truss will be considered if such a system can be shown to be in equilibrium.

If a whipping pipe impacts adjacent pipes of equal or greater nominal pipe size and equal or heavier wall thickness, the adjacent pipe will be considered to be free from rupture. Protection from pipe-whip will not be provided if pipe rupture occurs in such a manner that the unrestrained pipe movement of either end of the ruptured pipe, in any possible direction about a plastic hinge formed at the nearest pipe-whip restraint, cannot impact any structure, system or component required for that incident.

Piping that is physically separated or isolated from structures, systems or components important to safety will be excluded from the analysis. The physical separation or isolation may take the form of protective barriers or pipe restraints designed specifically for pipe whip, such as concrete encasement.

2.4 CRITERIA FOR COMPARTMENT PRESSURE, TEMPERATURE AND HUMIDITY

Compartment differential pressure loadings and temperature and humidity effects from design basis breaks or cracks will be considered. The mass and energy flow rate will be computed on the basis of the flow characteristics of the pipe immediately adjacent to the break or crack. The mass and energy flow characteristics will be chosen such that a conservative calculation of compartment pressures and temperature can be made.

2.5 CRITERIA FOR JET IMPINGEMENT AND REACTION

2.5.1 Jet Impingement

Jet impingement loads of safety-related equipment, components and structures will be considered for the design basis cases defined in Section 2.3 in a high-energy system. The magnitude and area of influence of the jet will be determined for each break according to the break location, size and orientation criteria given in Section 2.2 and the procedures described in Sections 3.0 through 7.0 as applicable. The jet forces or loads at the point of rupture will be consistent with those used in the pipe whip analysis.

2.5.2 Jet Reaction

Jet reaction forces on the postulated broken system consistent with jet impingement loads will be evaluated to determine system geometry and loads on supporting structures.

2.6 DESIGN CRITERIA TO MITIGATE CONSEQUENCES OF PIPE RUPTURE

2.6.1 Pipe Restraints

The piping restraints are designed to accommodate the loading

induced by the reaction or whipping forces from design basis breaks.

2.6.2 Structural Components

Existing structures will be reviewed as required in consideration of postulated breaks for their adequacy. If required, these structures will be modified to the extent necessary to guarantee their integrity. Any new structures will be designed to withstand the effects of a postulated high energy system pipe failure. The analysis will consider the effects of pressure and temperature transients, static, thermal and dynamic reactions of the pipe in conjunction with applicable loads.

Failure of any structure, caused by the postulated break, will be reviewed to assure that it consequently will not cause failure of any other structure, system, or component in a manner to preclude the mitigation of the consequences of the postulated break and the capability to bring the plant to a safe shutdown condition.

Barriers will be provided if required to protect needed equipment from adverse jet-force loadings from all design basis events. The barriers provided will be shown to withstand the impingement loads as well as the normal working loads.

Criteria and method used to analyze the structural adequacy of existing structures and the design of new structures if required will be in accordance with FSAR commitments.

2.7 CRITERIA FOR PLANT OPERABILITY FOLLOWING PIPE RUPTURE

2.7.1 Control Room Habitability

The control room will be maintained habitable and its equipment functional for all design bases events. Thus, the capability to bring the reactor to a safe shutdown condition from the control room will be maintained.

2.7.2 Operation of Needed Equipment

Electrical equipment required to function following the accident will be capable of withstanding the steam-air environment resulting from the postulated pipe break or crack.

2.7.3 Redundancy

The capability to mitigate the consequences of an accident and bring the reactor to a safe shutdown condition will be assured. Loss of redundancy of equipment required for a particular accident will not be permitted in the protection system (as defined in IEEE-279) and Class II electric system (as defined in IEEE-308), or for engineered safety features equipment, cable penetrations and their interconnecting cables.

Environmentally induced failures caused by a crack or break which would not in itself result in protective action but may disable protective function will also be considered. In this regard, a loss of redundancy will be permitted but a loss of function will not be permitted. For such situations the capability for bringing the plant to safe shutdown will be assured.

2.7.4 Emergency Procedures

General emergency procedures will allow for evaluation of the specific incident and determination of appropriate actions to be taken to achieve a safe shutdown condition. Prompt achievement of hot shutdown will be assured automatically by adherence to the aforementioned criteria and maintenance of hot shutdown will be accomplished by adherence to general emergency procedures. These procedures will also allow for placing the reactor in a cold shutdown condition.

3.0 MAIN STEAM LINE ROUTING AND RUPTURE EVALUATION

3.1 ROUTING DESCRIPTION

The Main Steam lines from Steam Generators A, B and C penetrate the containment from the south at elevation 253'-6" and run outside of the containment, in a seismic Class I support tower, through power operated relief valves and main steam stop and check valves to the Steam Header. Two lines leave the header, turning south, and enter the Turbine Generator Building. These two lines run horizontally at elevation 258'-0", turn east and rise to elevation 274'-3'3/4" where they are welded to V1-1 valves through 32" x 30" reducers. The above routing is shown in isometric sketches Figures 3-1 and 3-2 and layout drawings Figures 3-3 and 3-4.

3.2 DESIGN BASIS BREAKS

3.2.1 Description of Break Locations

Postulated design basis break locations outside containment are as tabulated in Tables 3-1 and 3-2, with node numbers referring to Figures 3-1 and 3-2. These locations have been determined on the basis of calculated stress values and the criteria given in Section 2.2.1 for ANSI B31-1 p piping breaks. These consist of:

a. The terminal points of the main steam lines at the turbine stop valves in the Turbine Building, the containment wall anchor and the main steam header anchor.

b. The branch point connections in the main steam lines to reheaters and dump to condensers, the intermediate

points, the branch connection points and the terminal points on these branch lines.

c. The branch point connection is the main steam line to Steam Seal regulator.

d. Two additional points having the highest calculated stress values in each main steam and branch lines.

The stresses at the postulated break locations were obtained from the output of computer stress analyses for thermal expansion and seismic loads and a conservatively estimated stress for all components under dead weight load. The combined stress values due to thermal expansion, pressure, weight and seismic loading conditions have been computed and summarized in Tables 3-1 and 3-2. Only one of the points of the calculated stress values exceed 0.8 S_a or 0.8 (S_a + S_h). This location is at the branch point, of main steam to steam seal regulator.

The limiting break locations are the terminal ends of the 32-inch main steam piping at inlets to turbines. For a circumferential break postulated at the limiting break location, the main steam line is assumed to hinge at the main steam header and would permit the whipping 32-inch line to impact containment. Analyses were performed for these limiting cases, and the results are presented in Section 12.0 of this report.

3.2.2 Required Equipment

The equipment required for a design basis break in the main steam line is given in Table 8.0-1. Operability of

this equipment provides for reactor trip and the capability to maintain the reactor at hot shutdown after the break as well as ultimately achieving cold shutdown. Required equipment includes associated piping, cables and structures required for the equipment to perform its function.

3.2.3 Protection from Potential Pipe-Whip Damage

Pipe rupture restraints are provided and it has been determined that additional rupture restraints are not necessary in order to protect required equipment listed in Table 8.0-1.

A structural analysis was performed to determine the structural integrity of the steam header in the event of a postulated steamline break. In this analysis, it was assumed that the broken pipe would exert on the steam header a moment equal to the upper bound plastic hinge capacity of the broken pipe and an axial load equal to the thrust load determined from the broken line. The header was assumed restrained by the four other intact steam lines attached to it plus the support structure of the turbine building.

The turbine building structure was modeled as two interacting planar frames having the composite stiffners of the several turbine building bents in the immediate vicinity of the header support. Using the computer program STRUDL, a 6x6 diagonal stiffness matrix was determined

which represented the structural stiffness of that portion of the turbine building supporting the steam header. This building stiffness was used as an input boundary restraint to a mathematical model of the steam header and attached piping. This system was analyzed by the computer program WESDYNE for loads determined from the postulated pipe break to develop resultant loads in the attached pipe and forces in the restraint structure. The loads in the attached piping were evaluated and found in the limiting case to be about 15 percent of the lower bound plastic hinge moment of the attached piping. The loads in the support structure were reapportioned to the actual building frame bents in proportion to their relative stiffness and evaluated in accordance with AISC-69 Building Specification requirements with allowable stresses increased by the factor of 1.65 which is consistent with the Faulted Condition nature of the pipe break load.

Results of this analysis indicate the building structure will not fail as the results of postulated pipe break with loads in the turbine bent members averaging about 60 percent of their capacities with the maximum member load in the highly redundant reaching 102 percent of its capacity. Therefore, it can be concluded the steam header support system will not fail in the event of steamline break.

3.2.4 Protection from Jet Impingement

It has been determined that impingement barriers or guard-pipes are not necessary in order to protect the needed equipment listed in Table 8.0-1.

3.2.5 Protection from Adverse Environmental Conditions

Ventilation penetrations will be sealed when required to prevent Steam Environment from affecting required equipment located within the Control Room Air Conditioning Room, Control Room, Relay Room, Class 1 Area, Control Rod Drive Mechanism (CRDM) Room and Diesel Generator Rooms.

3.3 DESIGN BASIS CRACK

3.3.1 Description

The orientation and location of a design basis crack can be anywhere along the piping.

3.3.2 Required Equipment

The equipment required to place the reactor in a safe-shutdown condition is shown in Table 8.0-1. Required equipment includes associated piping, cabling and structures required for the equipment to perform its functions.

3.3.3 Protection from Potential Pipe-Whip Damage

Additional pipe restraints were determined not necessary in order to protect required equipment listed in Table 8.0-1.

3.3.4 Protection from Jet Impingement

It has been determined that impingement barriers or guard pipes are not necessary in order to protect the needed equipment listed in Table 8.0-1.

3.3.5 Protection from Environmental Conditions

Required equipment exposed to environmental conditions was evaluated. It was found that they are capable to perform their function.

TABLE 3-1

Stress Values -- Main Steam #1, #2 & #3

Containment to Main Steam Header

Allowable Stress Values:
(0.8 $S_{allowable}$)
Operational Plus Seismic Stresses < 27,400 psi

Thermal Stresses < 16,440 psi

Operational and Seismic Stresses

Node No.*	Pressure	Gravity	Thermal	Seismic (OBE)	Total
1	6,372	2,000**	13,745	3,063	25,180
11	6,372	2,000	14,953	417	23,742
12	6,372	2,000	15,148	1,986	25,506
18	6,372	2,000	4,667	217	13,256
20	6,372	2,000	12,710	3,989	25,071
31	6,372	2,000	14,371	533	23,276
32	6,372	2,000	14,490	2,264	25,126
39	6,372	2,000	779	235	7,386
40	6,372	2,000	12,365	4,856	25,593
52	6,372	2,000	15,782	668	24,822
53	6,372	2,000	15,049	2,584	26,005
61	6,372	2,000	6,052	458	14,882

*Node Numbers refer to figure 3-1

**Assumed maximum dead weight stress

TABLE 3-2

Stress Values -- Main Steam to Condenser "A" & "B"

Moisture Separator 1-A, 1-B, 2-A, 2-B

Allowable Stress Values:

Operational Plus Seismic Stresses < 27,400 psi

Thermal Stresses < 16,440 psi

Node No.*	Pressure	Gravity	Thermal	Seismic (OBE)	Total
	Operational and Seismic Stresses				
80	6,370	2,000**	5,098	0	13,468
84	5,894	2,000	2,404	1,685	11,983
86	3,091	2,000	19,341	10,065	34,497
91	6,370	2,000	11,128	283	19,781
97	6,370	2,000	9,963	2,034	20,367
98	6,370	2,000	9,346	857	18,573
99	6,370	2,000	5,146	646	14,162
107	6,370	2,000	12,712	274	21,356
109	5,894	2,000	8,834	800	17,528
113	6,370	2,000	10,466	1,556	20,392
114	6,370	2,000	9,250	794	18,414
115	5,894	2,000	13,423	985	22,302
116	5,894	2,000	16,145	983	25,022
123	5,694	2,000	8,286	1,009	16,989
127	5,427	2,000	2,607	2,577	12,611
128	5,427	2,000	1,884	623	9,934

*Node Numbers refer to figure 3-2

**Assumed maximum dead weight stress

TABLE 3-2 (Continued)

Stress Values -- Main Steam to Condenser "A" & "B"

Moisture Separator 1-A, 1-B, 2-A, 2-B

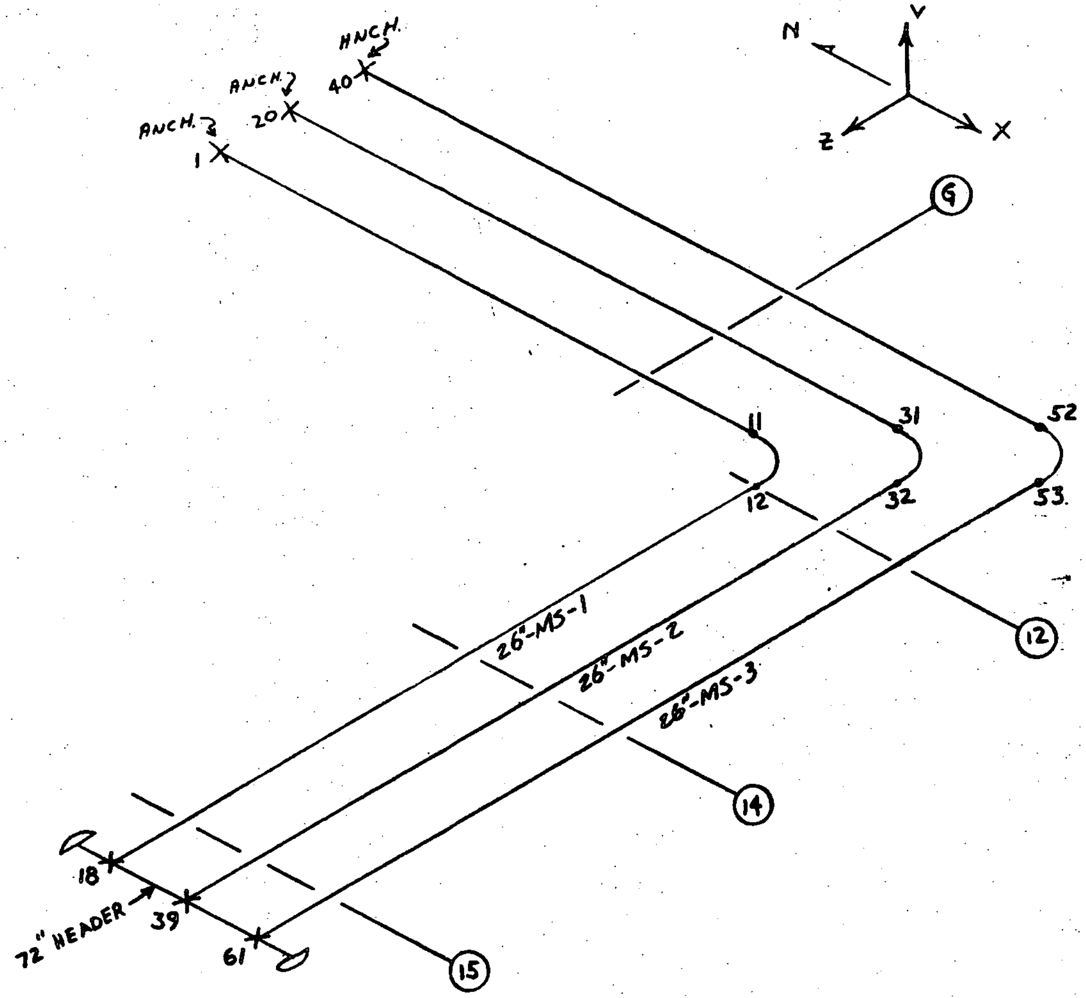
Node No.	Operational and Seismic Stresses				
	Pressure	Gravity	Thermal	Seismic (OBE)	Total
130	5,427	2,000	2,648	467	10,542
131	5,427	2,000	3,887	1,011	12,325
132	5,427	2,000	5,238	1,854	14,519
133	5,694	2,000	378	592	8,964
135	5,694	2,000	7,743	1,852	17,289
136	5,694	2,000	7,140	1,559	16,393
148	5,694	2,000	7,452	2,423	17,569
156	5,694	2,000	11,490	2,675	21,859
157	5,694	2,000	10,975	2,673	21,344
158	5,694	2,000	8,165	2,753	18,612
161	5,427	2,000	4,379	2,662	14,468
162	5,427	2,000	4,812	2,800	15,039
165	5,427	2,000	1,123	3,407	11,957
167	5,427	2,000	5,666	2,313	15,406
171	5,427	2,000	6,273	2,070	15,770
172	5,427	2,000	7,817	2,770	18,014
179	5,894	2,000	12,139	959	20,992
180	5,894	2,000	13,402	946	22,242
182	5,694	2,000	8,265	1,503	17,462
[illegible]6	5,427	2,000	3,779	1,973	13,179
187	5,427	2,000	2,010	670	10,107
189	5,427	2,000	3,337	378	11,142

TABLE 3-2 (Continued)

Stress Values -- Main Steam to Condenser "A" & "B"

Moisture Separator 1-A, 1-B, 2-A, 2-B

Node No.	Pressure	Gravity	Thermal	Seismic (OBE)	Total
	Operational and Seismic Stresses				
190	5,427	2,000	4,128	890	12,445
191	5,427	2,000	4,951	1,405	13,783
192	5,694	2,000	742	1,025	9,461
194	5,694	2,000	8,099	1,662	17,455
195	5,694	2,000	7,453	1,363	16,510
207	5,694	2,000	7,640	2,183	17,517
214	5,694	2,000	10,626	2,597	20,917
215	5,694	2,000	11,973	2,759	22,426
217	5,694	2,000	2,052	2,617	12,363
218	5,427	2,000	5,394	2,251	15,072
221	5,427	2,000	5,078	3,327	15,832
223	5,427	2,000	2,211	2,644	12,282
226	5,427	2,000	7,241	2,318	16,986
228	5,427	2,000	4,424	1,304	13,155
229	5,427	2,000	7,370	2,324	17,121
243	3,091	2,000	2,901	505	8,497
250	3,091	2,000	3,452	1,797	10,340
251	3,091	2,000	3,537	1,533	10,161



NOTES:

1. NODE POINTS SHOWN IN THIS FIGURE ARE THE BREAK LOCATIONS IDENTIFIED IN SECTION 3.2.1.
2. BREAK LOCATIONS HAVE BEEN DETERMINED BY THE CRITERIA OUTLINED IN SECTION 2.2.1
3. PIPE LINE DESIGNATIONS:

MS-1, 2 & 3 | MAIN STEAM - STEAM GENERATOR OUTLET

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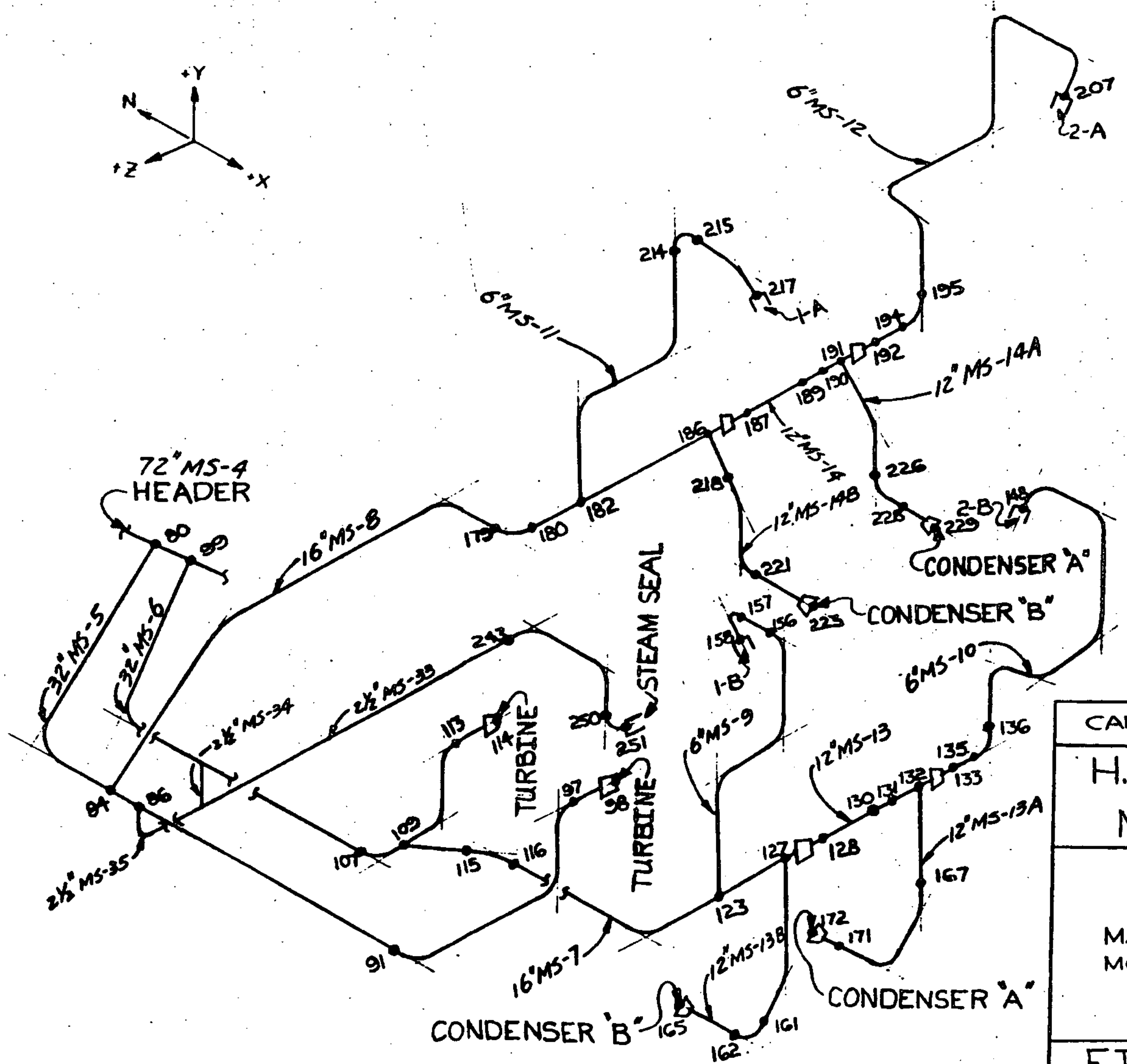
NUCLEAR POWER PLANT

ISOMETRIC

MAIN STEAM #1, #2 & #3
CONTAINMENT TO MAIN STEAM HEADER

FIGURE NO. 3-1



NOTES:

1. NODE POINTS SHOWN IN THIS FIGURE ARE THE BREAK LOCATIONS IDENTIFIED IN SECTION 3.2.1
2. BREAK LOCATIONS HAVE BEEN DETERMINED BY THE CRITERIA OUTLINED IN SECTION 2.2.1
3. PIPE LINE DESIGNATIONS:

MS-4	MAIN STEAM HEADER
MS-5 & 6	MAIN STEAM TO TURBINE
MS-7	MAIN STEAM TO REHEATERS & DUMP TO COND.
MS-8	MAIN STEAM TO REHEATERS & DUMP TO COND.
MS-9, 10, 11 & 12	MAIN STEAM TO REHEATERS
MS-13, 13A & 13B	MAIN STEAM DUMP TO CONDENSERS
MS-14, 14A & 14B	MAIN STEAM DUMP TO CONDENSERS
MS-33, 34 & 35	MAIN STEAM TO STEAM SEAL REG.

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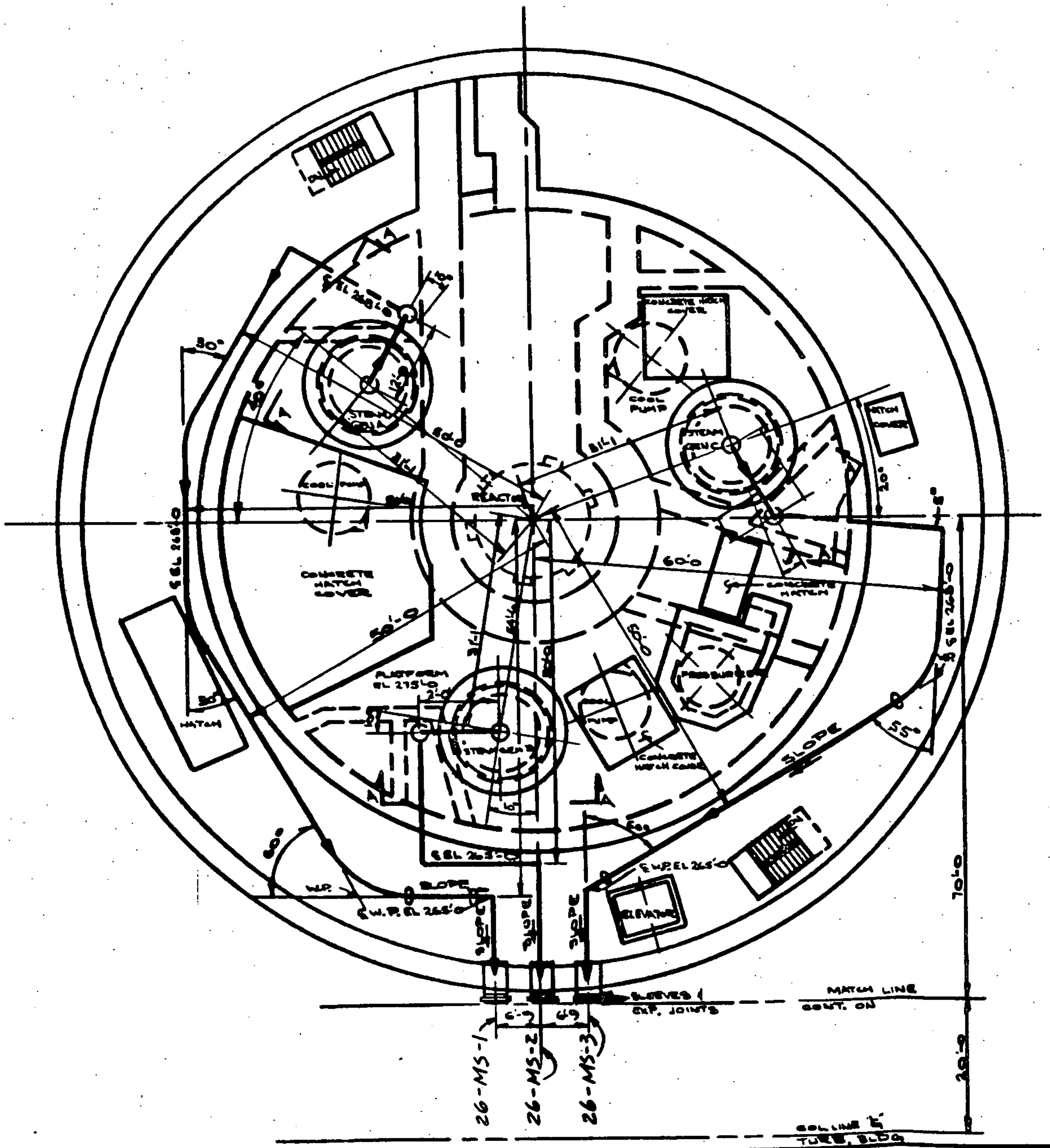
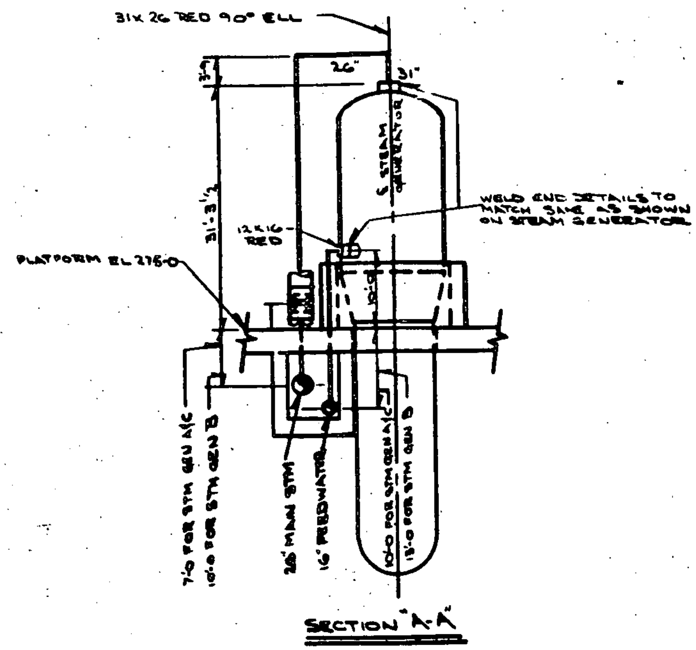
ISOMETRIC

MAIN STEAM TO CONDENSER "A" & "B"
MOISTURE SEPARATOR 1-A, 1-B, 2-A & 2-B

FIGURE NO. 3-2





SECTION "A-A"

CAROLINA POWER AND LIGHT COMPANY

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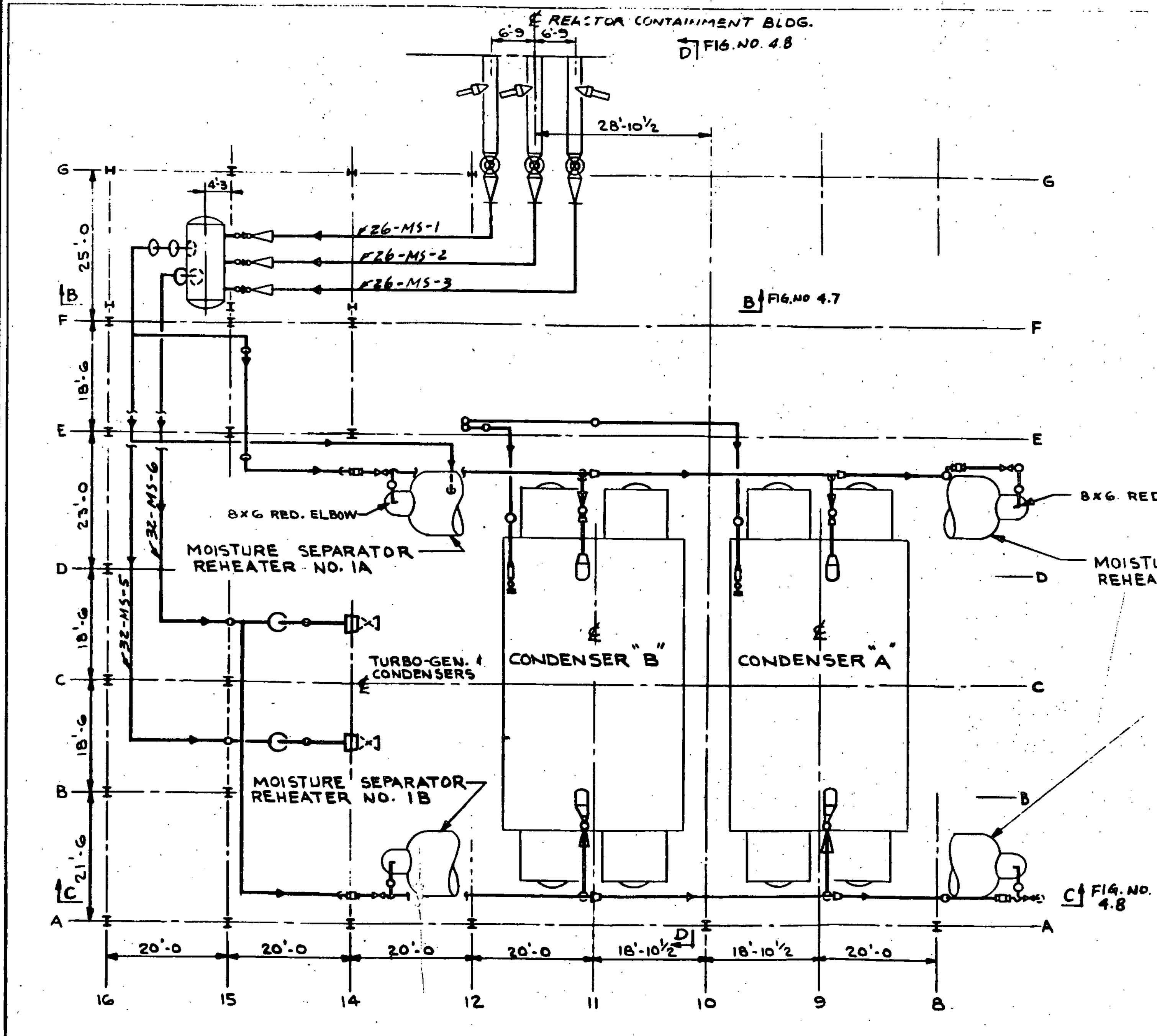
MAIN STEAM PIPING INSIDE CONTAINMENT & SECTION

FIGURE NO. 3.3



CAROLINA POWER AND LIGHT COMPANY

H. B. ROBINSON UNIT NO. 2
NUCLEAR POWER PLANT

STEAM PIPING
PLAN VIEW

FIGURE NO. 3-1

4.0 FEEDWATER ROUTING AND RUPTURE EVALUATION

4.1 ROUTING DESCRIPTION

The main feedwater lines are routed through the same area as the main steam lines. The discharge of each main feedwater pump is routed in the Turbine Building through a check valve and motor operated valve at the ground floor and high pressure heater at the mezzanine floor. Lines run parallel to the turbine building toward the north and three 16" main feedwater lines enter the support tower structure at elevations 244'-0", 249'-6" and 253'-6" respectively. The two lower elevated feedwater lines rise to the elevation 253'-6" and all three lines head north and penetrate the containment at the elevation from the south. Each line is provided with a flow nozzle, control valve and motor operated isolation valve before entering containment. These main feedwater lines, designated as 16-FW-9, 10 and 11 are routed toward Steam Generators A, B and C respectively after penetrating the containment. The above routing is shown in isometric sketches Figures 4-1, 4-2, 4-3 and 4-4 and layout drawings Figures 4-5, 4-6, 4-7, and 4-8. Feedwater piping from low pressure heaters are shown in layout drawings Figures 4-9 and 4-10.

4.2 DESIGN BASIS BREAKS

4.2.1 Description of Break Location

Postulated design basis break locations outside containment are as tabulated in Tables 4-1, 4-2, 4-3 and 4-4, with node numbers referring to Figures 4-1, 4-2, 4-3, 4-4. These locations have been determined on the basis of calculated

stress values and the criteria given in Section 2.2.1 for ANSI B31-1 piping breaks. These consist of:

a. The terminal points which are located at the high pressure feedwater heaters, feedwater pumps and the containment penetrations.

b. Branch point connections and terminal points for the feedwater piping from steam driven auxiliary feedwater pump.

c. Branch point connections and terminal points for the feedwater piping from motor driven auxiliary feedwater pumps.

d. Terminal points at typical structural anchors at elevations 227.50' and 244.00' for piping branches or runs.

e. Two intermediate points for each piping run or major branch run having the highest calculated stress values.

The stresses at the postulated break locations were obtained from the output of computer stress analyses for thermal expansion and seismic loads and a conservatively estimated stress for all components under dead weight load. The combined stress values due to thermal expansion, pressure, weight and seismic loading conditions have been computed and summarized in Tables 4-1, 4-2, 4-3 and 4-4. At no point do the calculated stress values exceed 0.8 S_a or 0.8 (S_h + S_a).

It has been determined that a plastic hinge formation at the outlet of the low pressure heater 5A would result in the 16-inch line whipping on the 18-inch thick concrete wall surrounding the safeguards switchgear room. The limiting break location was found at the bottom elbow of the 14'-7 7/16" riser. A detailed analysis of the equations of motion of the postulated broken line was performed and the results are presented in Section 9.0.

4.2.2 Required Equipment

The equipment needed for a design basis break in the feedwater line is given in Table 8.0-1. Operability of this equipment provides for reactor trip and the capability to place and maintain reactor in a safe shut-down condition. Required equipment includes associated piping, cables and structures required for the equipment to perform its function.

4.2.3 Protection from Potential Pipe Whip Damage

Pipe Rupture restraints are provided and it has been determined that additional rupture restraints are not necessary in order to protect required equipment listed in Table 8.0-1.

4.2.4 Protection from Jet Impingement

It has been determined that impingement barriers or guardpipes are not necessary in order to protect the required equipment listed in Table 8.0-1.

4.2.5 Protection from Adverse Environmental Conditions

Ventilation penetrations will be sealed as necessary to prevent the steam environment from affecting required equipment located within the Control Room Air Conditioning Room, Control Room, Relay Room, Class I Area CRDM Room and Diesel Generator Rooms.

4.3 DESIGN BASIS CRACKS

4.3.1 Description

The orientation and location of a design basis crack can be anywhere along the piping.

4.3.2 Required Equipment

The equipment required to place the reactor in a safe shutdown condition is given in Table 8.0-1. Required equipment includes associated piping, cabling and structures required for the equipment to perform its functions.

4.3.3 Protection from Potential Pipe Whip Damage

Additional pipe restraints were determined not necessary in order to protect the required equipment listed in Table 8.0-1.

4.3.4 Protection from Jet Impingement

An additional impingement barrier will be provided in order to protect the needed steam system transmitters listed in Table 8.0-1.

4.3.5 Protection from Environmental Conditions

Required equipment exposed to environmental conditions resulting from cracks was evaluated. It was found that they are capable to perform their function.

TABLE 4-1

Stress Values -- Feedwater Piping

From Auxiliary FDW Pumps "A" & "B"

Allowable Stress Values:
(0.8 S allowable)
Operational Plus Seismic Stresses < 30,000 psi

Thermal Stresses < 18,000 psi

Node No.*	Pressure	Gravity	Thermal	Seismic (OBE)	Total
	Operational and Seismic Stresses				
90	4,017	2,000**	9,934	3,394	19,345
92	4,017	2,000	9,296	5,991	21,304
93	4,017	2,000	8,034	4,720	18,771
[illegible]7	4,017	2,000	5,716	1,774	13,507
128	4,017	2,000	5,864	1,274	13,155
133	4,017	2,000	5,187	573	11,777
140	4,327	2,000	7,459	592	14,378
141	4,327	2,000	9,698	1,889	17,914
142	4,017	2,000	3,077	2,898	11,992
143	4,017	2,000	2,895	2,188	11,100
145	4,017	2,000	3,285	5,756	15,058
146	4,017	2,000	1,859	1,202	9,078
148	4,017	2,000	2,062	2,660	10,739
149	4,017	2,000	4,312	998	11,327
154	4,017	2,000	5,546	1,252	12,815
155	4,017	2,000	5,469	1,420	12,906

* Node Numbers refer to figure 4-1

** Assumed maximum dead weight stress

TABLE 4-1 (Continued)

Stress Values -- Feedwater Piping

From Auxiliary FDW Pumps "A" & "B"

Node No.*	Pressure	Gravity	Thermal	Seismic (OBE)	Total
	Operational and Seismic Stresses				
160	4,327	2,000	6,390	1,064	13,781
161	4,327	2,000	6,928	2,084	15,339
183	4,017	2,000	8,641	1,683	16,341
188	4,017	2,000	8,128	7,263	21,408
191	4,017	2,000	9,713	4,477	20,207
222	4,017	2,000	10,679	602	17,298
223	4,017	2,000	11,033	638	17,688
250	4,017	2,000	6,569	3,485	16,071

TABLE 4-2

Stress Values -- Feedwater Piping

From 4" Auxiliary FDW Pump Discharge to Anchors

Allowable Stress Values:
(0.8 S allowable)
Operational Plus Seismic Stresses < 30,000 psi

Thermal Stresses < 18,000 psi

Node No.*	Pressure	Gravity	Thermal	Seismic (OBE)	Total
	Operational and Seismic Stresses				
36	4,017	2,000**	2,994	5,559	14,570
73	4,017	2,000	2,995	5,391	14,403
109	4,017	2,000	2,811	5,633	14,461
[illegible]0	4,017	2,000	435	3,976	10,428
213	4,776	2,000	17,935	615	25,326
214	4,776	2,000	16,028	721	23,525
219	4,017	2,000	4,663	535	11,215
221	4,017	2,000	7,178	986	14,181
234	4,017	2,000	4,397	657	11,071
235	4,017	2,000	4,052	376	10,726
241	4,017	2,000	3,940	405	10,362
242	4,017	2,000	3,880	870	10,767

[illegible]Node Numbers refer to figure 4-2
** Assumed maximum dead weight stress

TABLE 4-3

Stress Values -- Turbine Bldg. FDW Piping

From Anchor EL. 227.50' to Anchor in Reactor Bldg.

Allowable Stress Values:
(0.8 S allowable)
Operational Plus Seismic Stresses < 30,000 psi

Thermal Stresses < 18,000 psi

Node No.*	Operational and Seismic Stresses				
	Pressure	Gravity	Thermal	Seismic (OBE)	Total
1	4,846	2,000	11,554	1,296	19,696
13	4,846	2,000	9,120	484	16,450
16	4,017	2,000	2,100	2,405	10,522
17	4,846	2,000	9,121	418	16,385
26	4,017	2,000	3,871	1,103	10,991
27	4,017	2,000	3,521	953	10,491
36	4,017	2,000	2,147	3,089	11,253
37	4,846	2,000	12,562	1,730	21,138
50	4,846	2,000	10,350	742	17,938
51	4,846	2,000	10,030	502	17,378
52	4,017	2,000	2,103	2,458	10,578
62	4,017	2,000	4,350	1,055	11,422
63	4,017	2,000	3,905	722	10,644

Node Numbers refer to figure 4-3

** Assumed maximum dead weight stress

TABLE 4-3 (Continued)

Stress Values -- Turbine Bldg. FDW Piping

From Anchor EL. 227.50' to Anchor in Reactor Bldg.

Node No.*	Pressure	Gravity	Thermal	Seismic (OBE)	Total
	Operational and Seismic Stresses				
73	4,017	2,000	2,603	3,061	11,681
74	4,846	2,000	11,580	3,268	21,694
88	4,846	2,000	10,112	381	17,339
90	4,017	2,000	3,007	2,201	11,225
91	4,846	2,000	10,600	186	17,632
102	4,017	2,000	5,703	810	12,530
[illegible]03	4,017	2,000	5,012	880	11,909
109	4,017	2,000	3,805	3,149	12,971
235	4,846	2,000	5	15	6,866
595	4,846	2,000	6	15	6,867
975	4,846	2,000	10	15	6,871

TABLE 4-4

Stress Values -- Turbine Bldg. FDW Pump

"A" & "B" Discharge to Heater 6A & 6B

And To Anchor at EL. 227.50' & 244.00'

Allowable Stress Values:
(0.8 S allowable)
Operational Plus Seismic Stresses < 30,000 psi

Thermal Stresses < 18,000 psi

Node No.*	Pressure	Gravity	Thermal	Seismic (OBE)	Total
	Operational and Seismic Stresses				
28	4,846	2,000**	9,238	960	17,044
36	4,846	2,000	1,255	1,115	9,216
65	4,846	2,000	7,131	805	14,782
73	4,846	2,000	446	1,139	8,431
100	4,846	2,000	7,933	965	15,744
109	4,846	2,000	643	1,084	8,573
150	4,846	2,000	2,503	344	9,693
153	4,846	2,000	9,048	372	16,266
154	4,846	2,000	8,250	264	15,360
156	4,907	2,000	15,015	1,747	23,669
157	4,846	2,000	2,648	668	10,162
158	4,846	2,000	3,705	587	11,138
159	4,882	2,000	8,882	1,932	17,696
164	4,882	2,000	2,549	250	9,681

*Node Numbers refer to figure 4-4

**Assumed maximum dead weight stress

TABLE 4-4 (Continued)

Stress Values -- Turbine Bldg. FDW Pump

"A" & "B" Discharge to Heater 6A & 6B

And To Anchor at EL. 227.50' & 244.00'

Node No.	Pressure	Operational and Seismic Stresses			
		Gravity	Thermal	Seismic (OBE)	Total
177	4,882	2,000	2,857	1,315	11,054
178	4,882	2,000	1,839	653	9,374
180	4,882	2,000	3,136	502	10,520
186	4,882	2,000	2,907	1,241	11,030
187	4,882	2,000	2,879	893	10,654
189	4,882	2,000	2,991	482	10,355
192	4,882	2,000	4,563	1,230	12,675
193	4,882	2,000	4,925	1,706	13,513
198	4,882	2,000	3,784	353	11,019
201	4,882	2,000	4,378	725	11,985
203	4,882	2,000	9,510	1,046	17,438
204	4,882	2,000	10,289	1,618	18,789
205	4,882	2,000	9,305	646	16,833
209	4,907	2,000	14,893	2,498	24,298
211	4,882	2,000	6,524	612	14,018
212	4,882	2,000	6,781	859	14,522
213	4,882	2,000	4,266	715	11,863
215	4,882	2,000	5,218	608	12,708
216	4,882	2,000	5,465	929	13,276
217	4,882	2,000	3,007	838	10,727
219	4,846	2,000	8,169	210	15,225

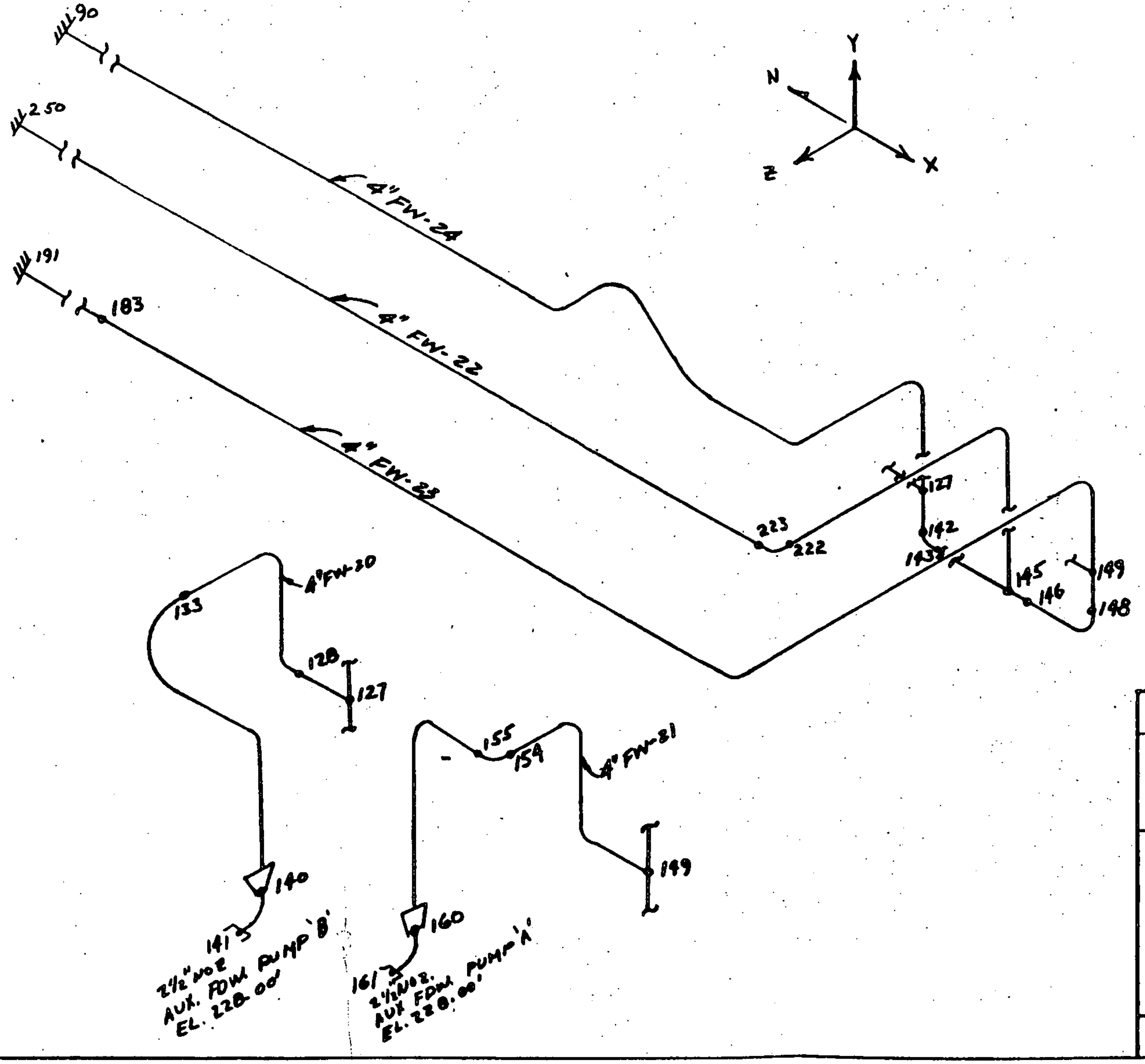
TABLE 4-4 (Continued)

Stress Values -- Turbine Bldg. FDW Pump

"A" & "B" Discharge to Heater 6A & 6B

And To Anchor at EL. 227.50' & 244.00'

Node No.			Operational and Seismic Stresses		
	Pressure	Gravity	Thermal	Seismic (OBE)	Total
220	4,846	2,000	8,323	334	15,503
223	4,846	2,000	1,521	329	8,696
229	4,846	2,000	7,866	273	14,985
230	4,846	2,000	8,725	391	15,962
233	4,846	2,000	1,758	280	8,884
[illegible]0	4,846	2,000	5,264	330	12,440
241	4,846	2,000	5,367	265	12,478
249	4,846	2,000	4,543	329	11,718
250	4,846	2,000	4,557	261	11,664
258	4,846	2,000	5,932	321	13,099
259	4,846	2,000	5,969	257	13,072



NOTES:

1. NODE POINTS SHOWN IN THIS FIGURE ARE THE BREAK LOCATIONS IDENTIFIED IN SECTION 4.2.1
2. BREAK LOCATIONS HAVE BEEN DETERMINED BY THE CRITERIA OUTLINED IN SECTION 2.2.1
3. PIPE LINE DESIGNATION:

   FW-20, FW-21 } MOTOR DRIVEN AUX. FEED PUMP DISCH.

   FW-22, FW-23, FW-24 } AUX. FDW. TO STEAM GENERATOR

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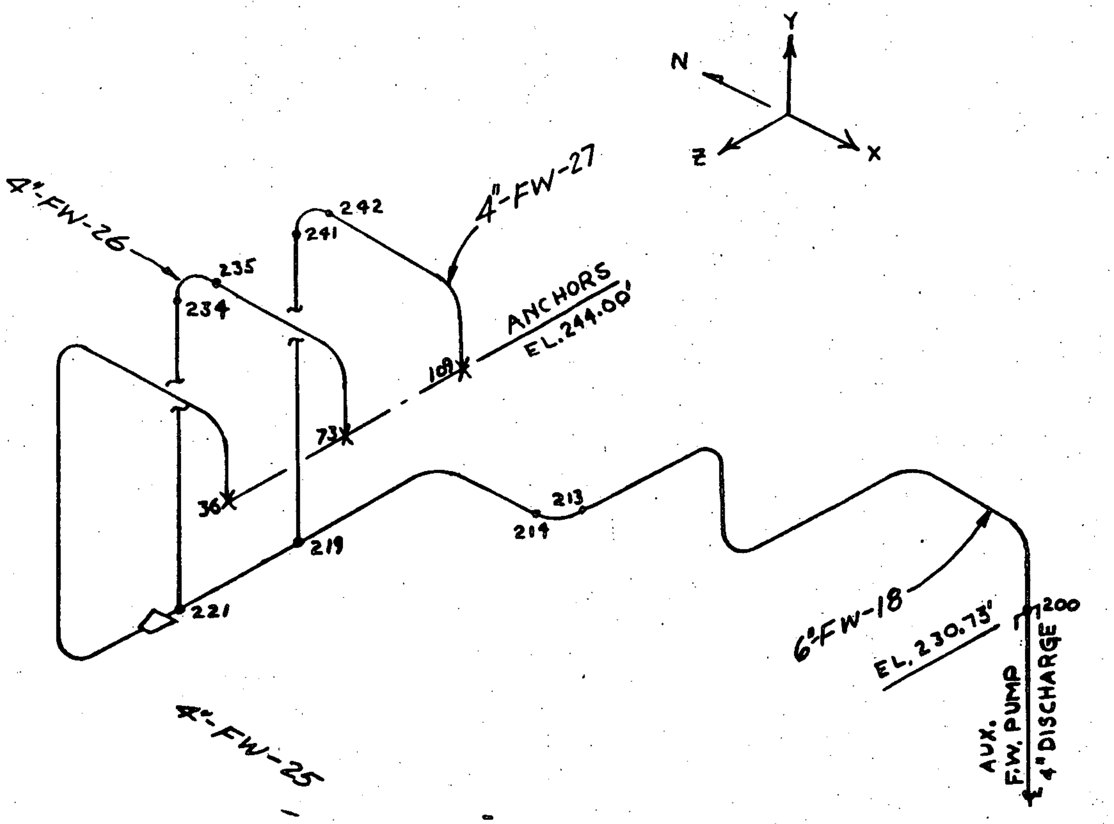
FEEDWATER PIPING FROM AUXILIARY FEEDWATER PUMPS "A" AND "B"

FIGURE NO. 4-1



NOTES:

1. NODE POINTS SHOWN IN THIS FIGURE ARE THE BREAK LOCATIONS IDENTIFIED IN SECTION 4.2.1
2. BREAK LOCATIONS HAVE BEEN DETERMINED BY THE CRITERIA OUTLINED IN SECTION 2.2.1
3. PIPE LINE DESIGNATIONS:

FW-18	AUX. FDW. PUMP DISCHARGE
FW-25	STEAM DRIVEN AUX. FDW. PUMP TO STEAM GEN.
FW-26	
FW-27	

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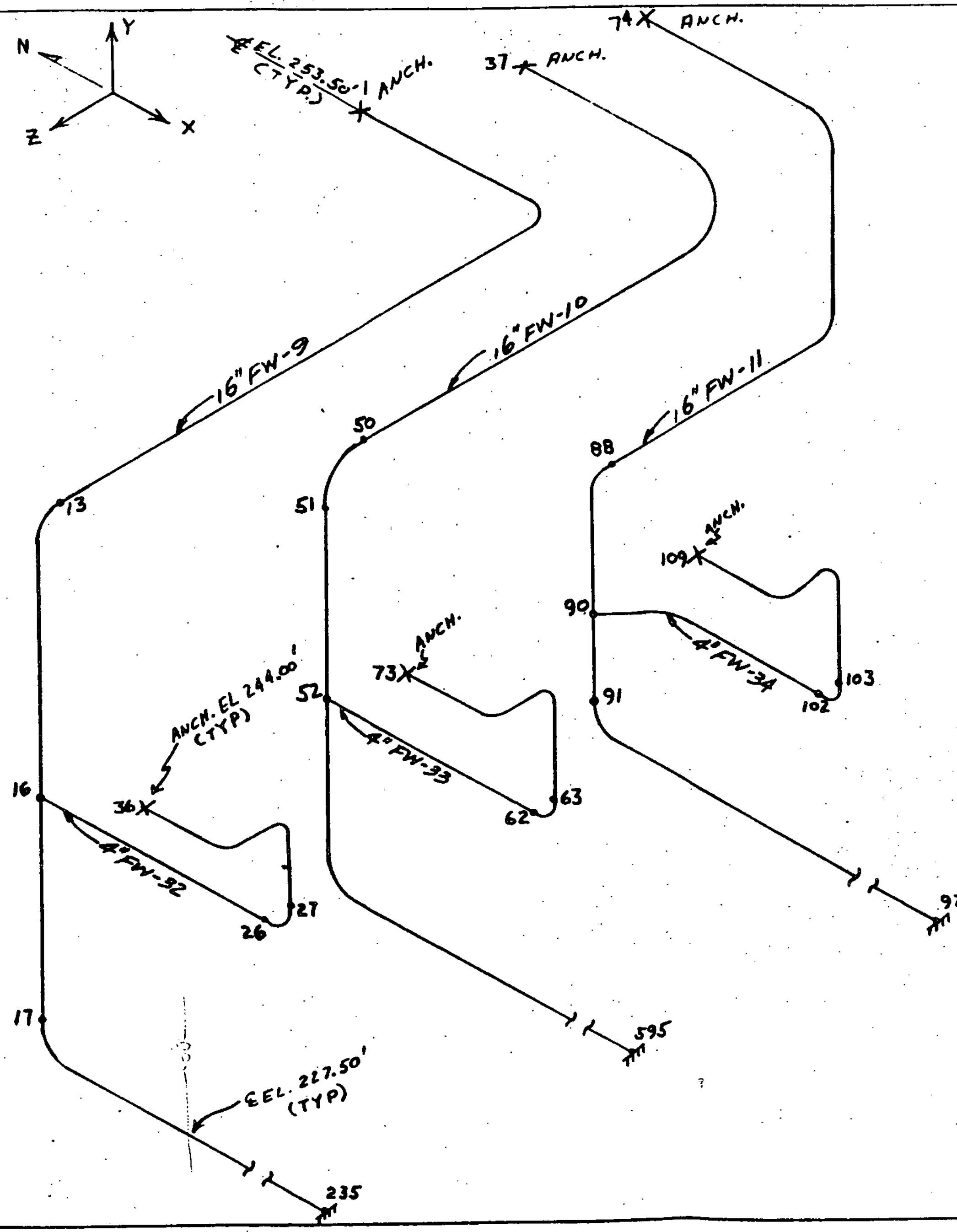
FEEDWATER PIPING FROM 4" AUXILIARY FEEDWATER PUMP DISCHARGE TO ANCHORS

FIGURE NO. 4-2



NOTES:

1. NODE POINTS SHOWN IN THIS FIGURE ARE THE BREAK LOCATIONS IDENTIFIED IN SECTION 4.2.1
2. BREAK LOCATIONS HAVE BEEN DETERMINED BY THE CRITERIA OUTLINED IN SECTION 2.2.1
3. PIPE LINE DESIGNATIONS :

FW-9, 10 & 11	FDW TO STEAM GENERATOR
FW-32, 33 & 34	FDW REGULATING VALVE BY-PASS

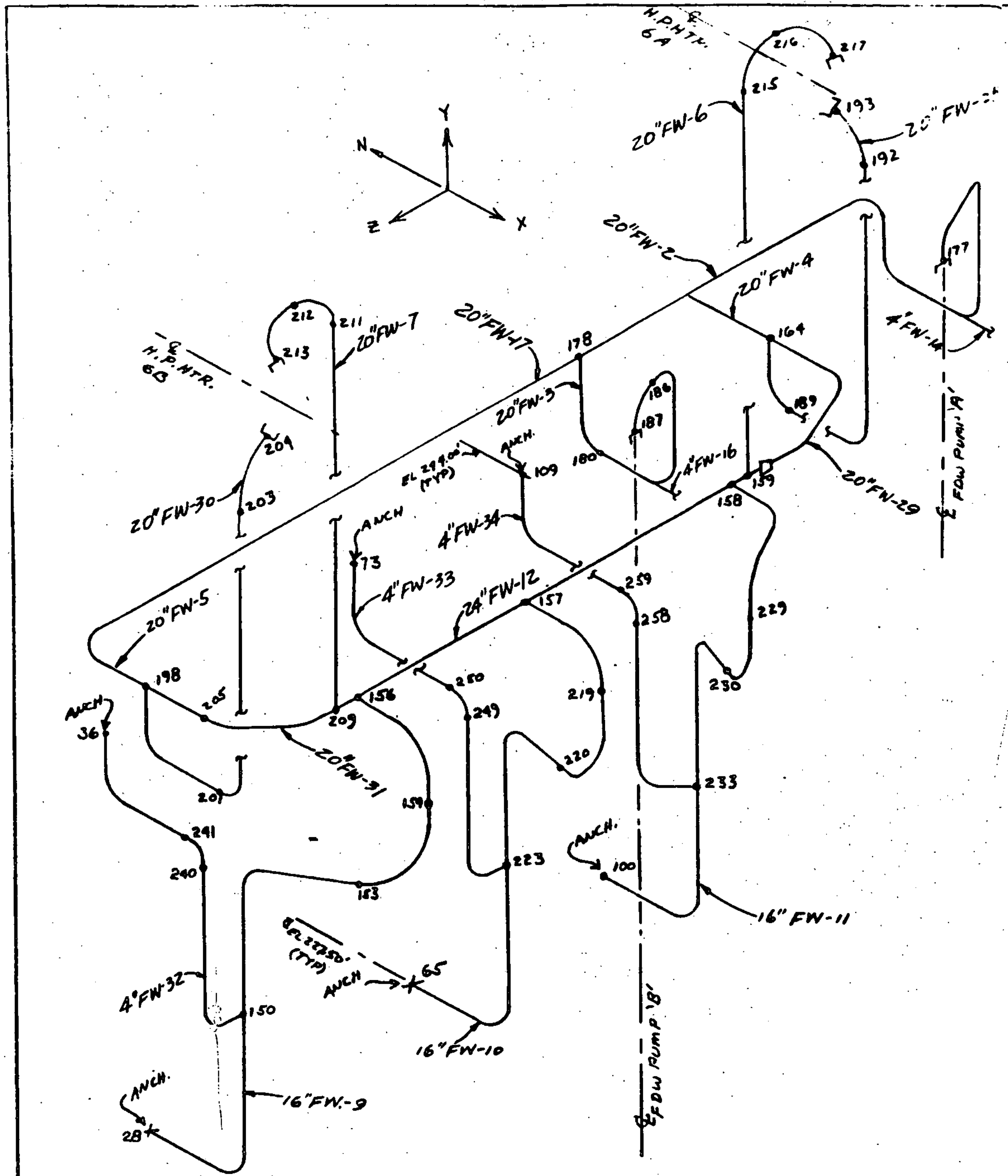
CAROLINA POWER AND LIGHT COMPANY

H. B. ROBINSON UNIT NO. 2
NUCLEAR POWER PLANT

ISOMETRIC

TURBINE BUILDING FEEDWATER PIPING FROM ANCHOR EL. 227.50' TO ANCHOR IN REACTOR BUILDING

FIGURE NO. 4-3



NOTES:

1. NODE POINTS SHOWN IN THE FIGURE ARE THE BREAK LOCATIONS IDENTIFIED IN SECTION 4.2.1
2. BREAK LOCATIONS HAVE BEEN DETERMINED BY THE CRITERIA OUTLINED IN SECTION 2.2.1
3. PIPE LINE DESIGNATIONS:

FW-2	FDW PUMP A DISCHARGE
FW-3	FDW PUMP B DISCHARGE
FW-4	FDW TO HP HEATER NO. 6A
FW-5	FDW TO HP HEATER NO. 6B
FW-6	FDW HP HEATER 6A OUTLET
FW-7	FDW HP HEATER 6B OUTLET
FW-9, FW-10, FW-11	FDW TO STEAM GENERATOR
FW-12	FDW. HEADER
FW-14, FW-16	FDW PUMP A/B RECIRCULATION TO CONDENSER A/B
FW-17	FDW PUMPS DISCHARGE HEADER
FW-28	FDW TO HP HEATER 6A
FW-29	HP HEATER 6A FDW BY-PASS
FW-30	FDW TO HP HEATER 6B
FW-31	HP HEATER 6B FDW BY-PASS
FW-32, FW-33, FW-34	FDW REGULATING VALVE BY-PASS

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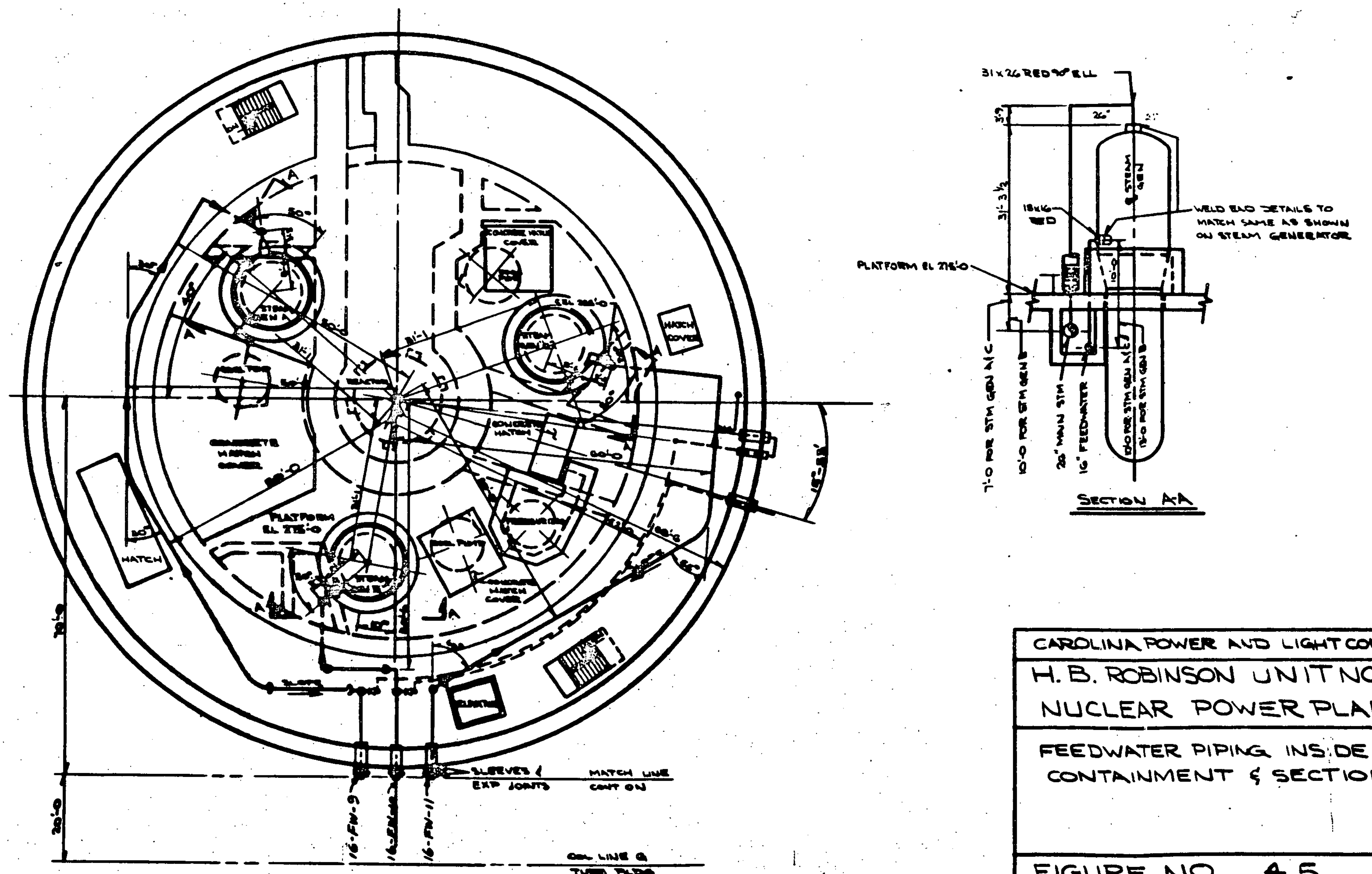
NUCLEAR POWER PLANT

ISOMETRIC

TURBINE BLDG. FDW PUMP "A" & "B" DISCHARGE TO HEATER 6A & 6B AND TO ANCHOR AT EL. 223.50' & 241.[illegible]

FIGURE NO. 4-4



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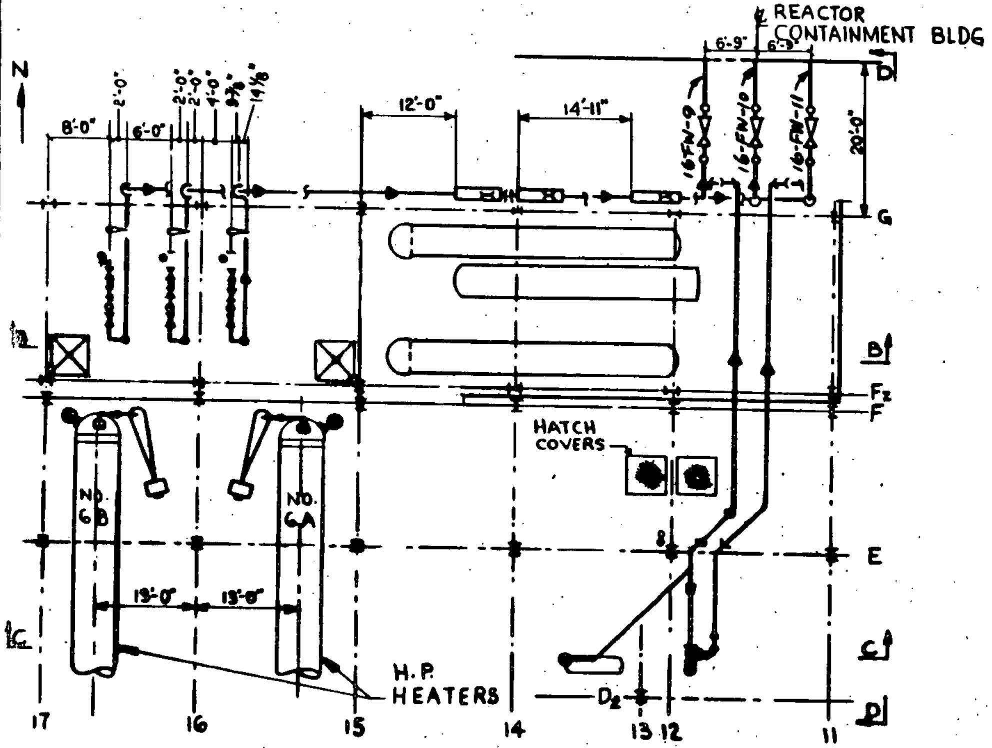
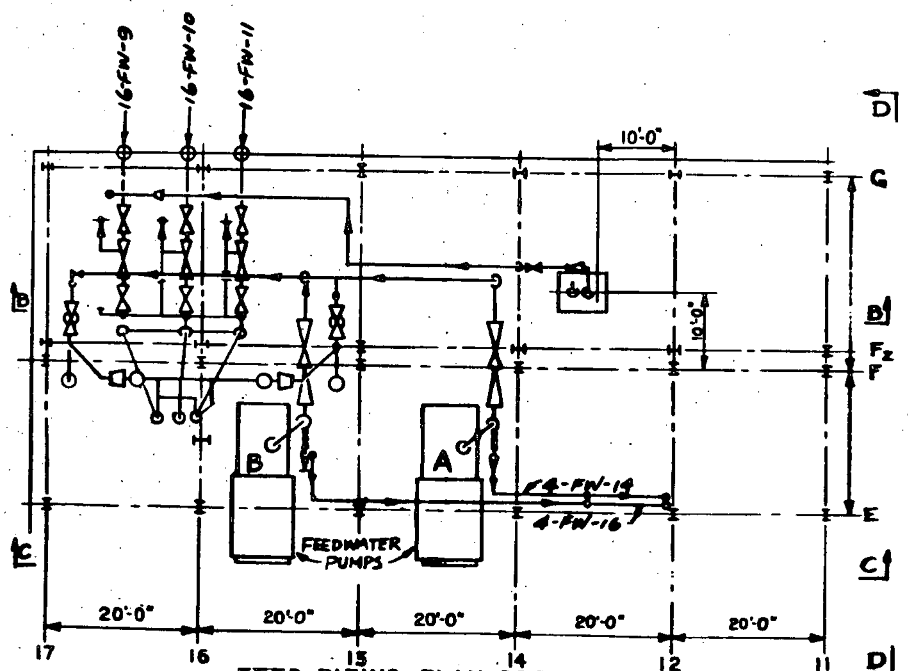
H. B. ROBINSON UNIT NO 2
NUCLEAR POWER PLANT

FEEDWATER PIPING INSIDE CONTAINMENT & SECTION

FIGURE NO 4.5

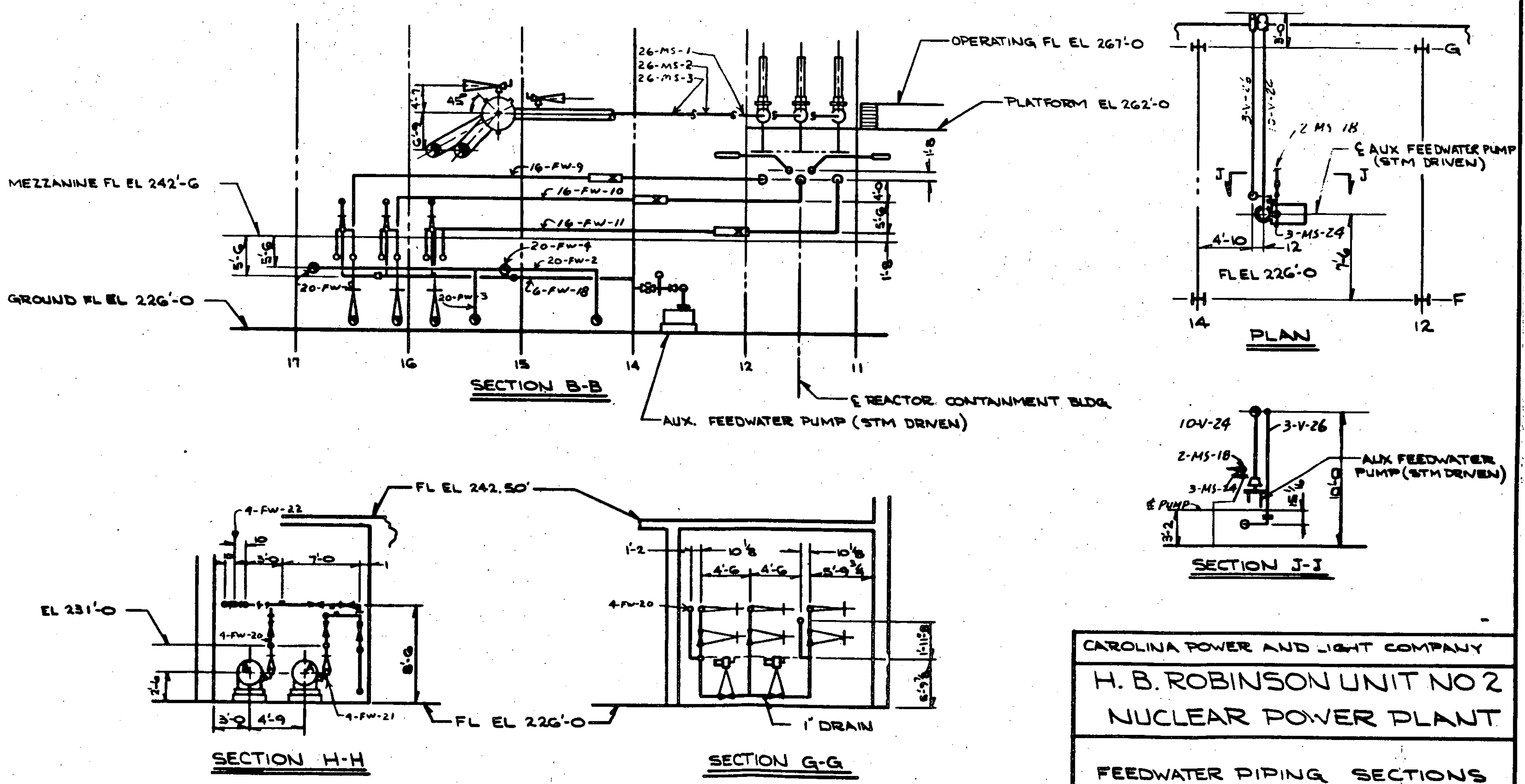


FEED PIPING PLAN- MEZZANINE & ABOVE



FEED PIPING - PLAN - GROUND FLOOR

CAROLINA POWER AND LIGHT COMPANY
H.B. ROBINSON UNIT NO. 2 NUCLEAR POWER PLANT
FEEDWATER PIPING MEZZANINE & GROUND FLOOR
FIGURE NO. 4-6

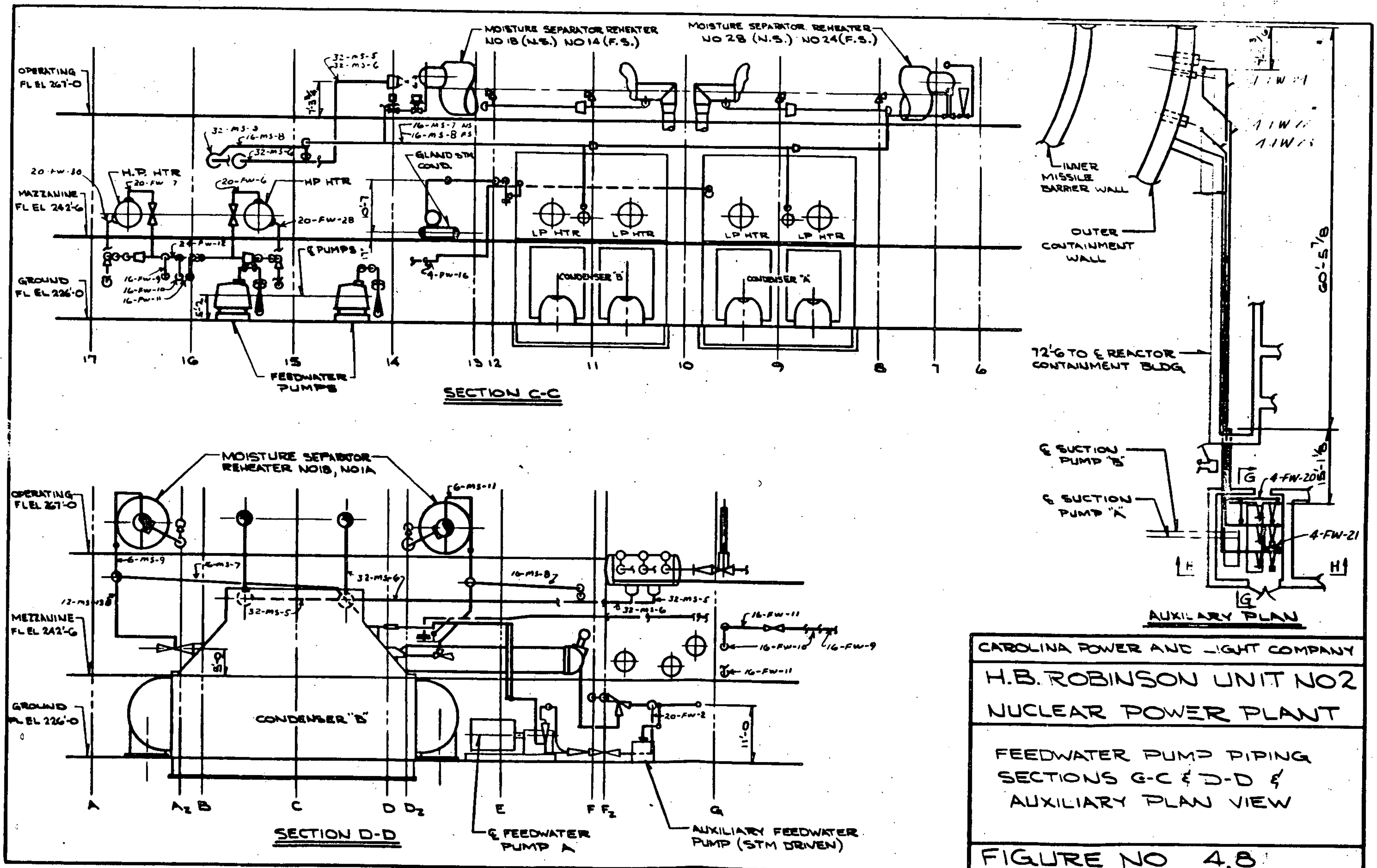


CAROLINA POWER AND LIGHT COMPANY

H. B. ROBINSON UNIT NO 2
NUCLEAR POWER PLANT

FEEDWATER PIPING SECTIONS "B-B", "H-H", "G-G", J-J & PLAN VIEW

FIGURE NO 4.7

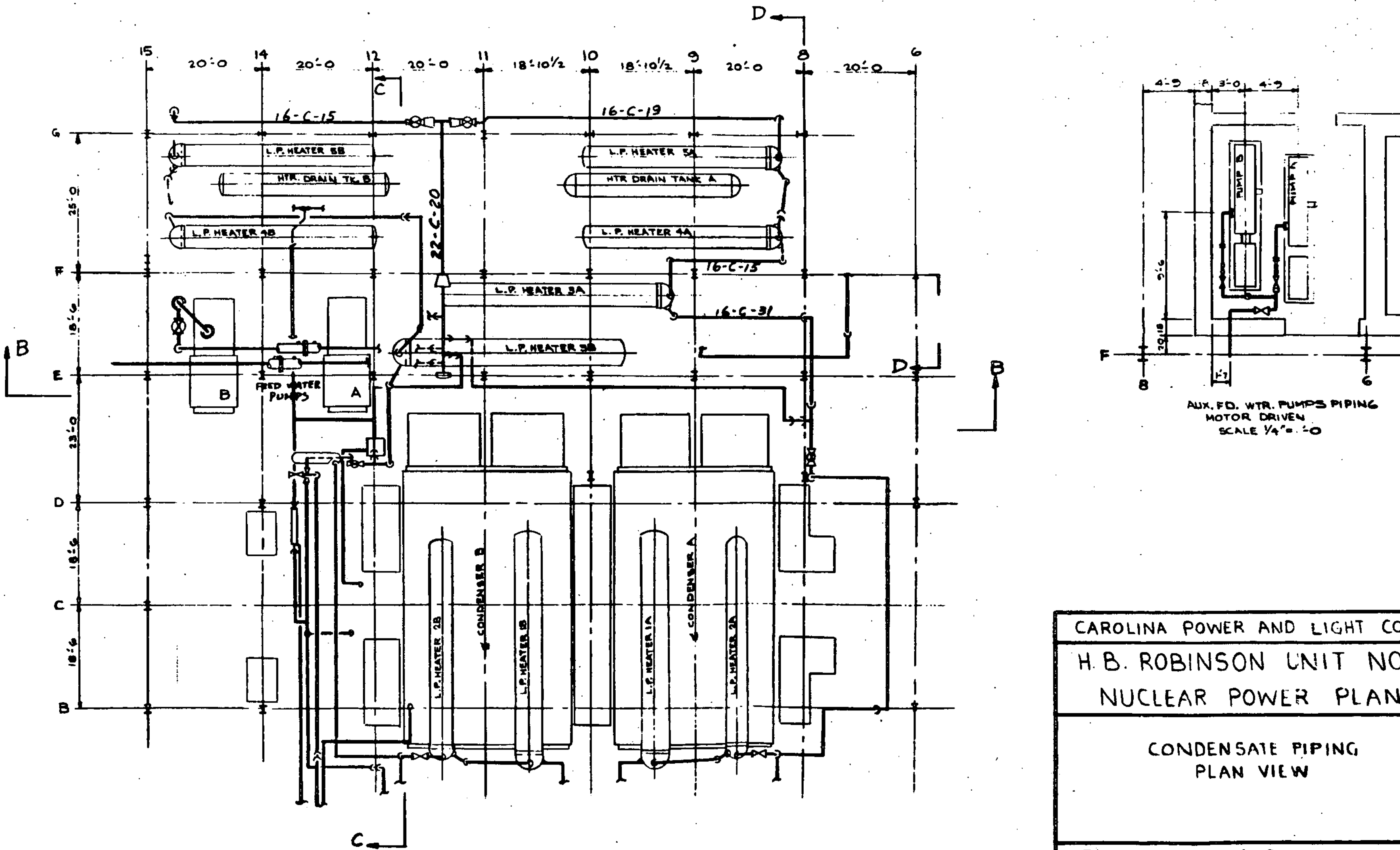


CAROLINA POWER AND LIGHT COMPANY

H.B. ROBINSON UNIT NO2
NUCLEAR POWER PLANT

FEEDWATER PUMP PIPING
SECTIONS C-C & D-D &
AUXILIARY PLAN VIEW

FIGURE NO 4.8

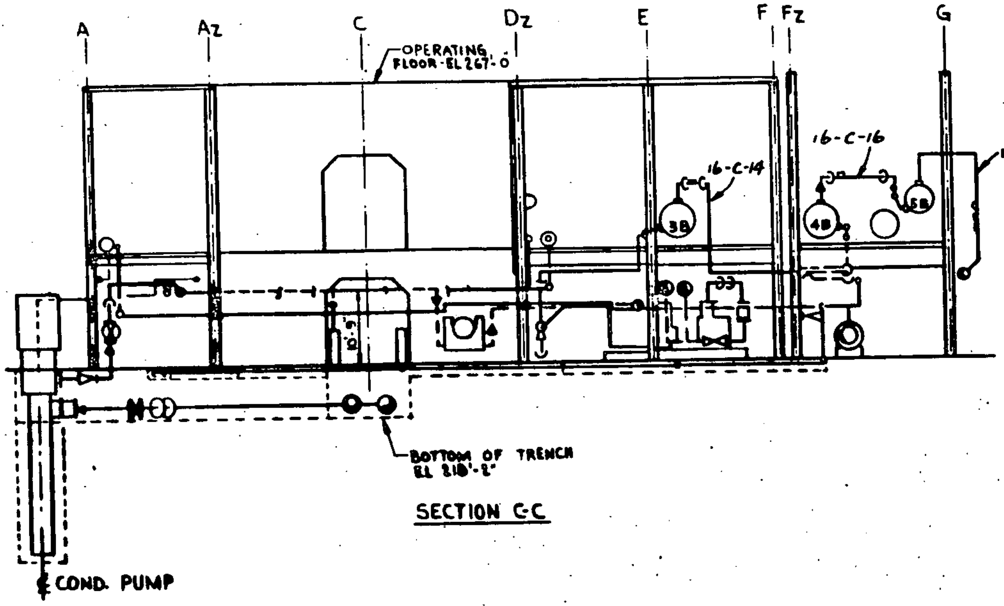


CAROLINA POWER AND LIGHT COMPANY

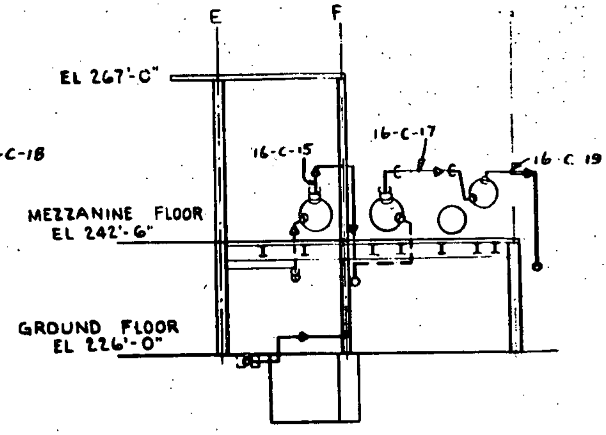
H. B. ROBINSON UNIT NO. 2
NUCLEAR POWER PLANT

CONDENSATE PIPING
PLAN VIEW

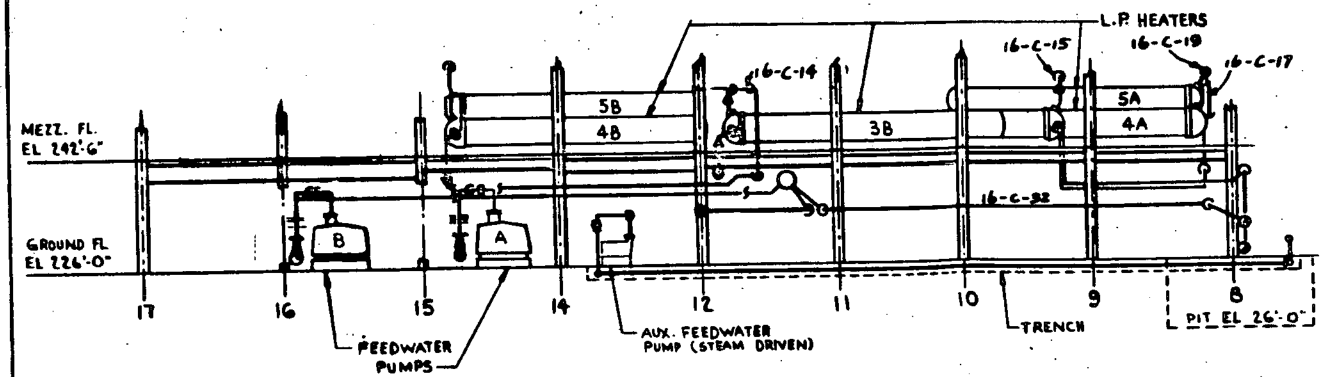
FIGURE NO 4-9



SECTION C-C



SECTION D-D



SECTION B-B

CAROLINA POWER AND LIGHT COMPANY

H. B. ROBINSON UNIT NO. 2
NUCLEAR POWER PLANT

CONDENSATE PIPING
SECTIONS

FIGURE NO. 4-10

5.0 CVCS LETDOWN AND CHARGING LINES ROUTING AND RUPTURE EVALUATION

5.1 ROUTING DESCRIPTION

The letdown line passes through the containment and then through the mechanical penetration area and into a pipe area to the non-regenerative heat exchanger which is next to the charging pump room at the ground floor. It then leaves the nonregenerative heat exchanger to PCV-145. The regenerative heat exchanger, the letdown orifice, and the isolation valves for the CVCS are located inside containment. Beyond PCV-145 the letdown line ceases to be a high energy line. In the penetration area it runs parallel and close to charging, seal water and steam generator blowdown lines.

The high energy portion of the charging line starts from the discharge of charging pumps A, B and C in the charging pump room located at the central portion of Auxiliary Building ground floor. Pump discharges are connected by the cross-connection lines before routing toward the pipe area. Then, the 3 inch charging line passes through the containment at the mechanical penetration area and leads toward the regenerative heat exchanger inside the containment. The above routing is shown in isometric sketch Figure 5-1 and layout drawings Figures 5-2, 5-3, 5-4 and 5-5.

5.2 DESIGN BASIS BREAKS

5.2.1 Description of Break Locations

Postulated design basis break locations for the letdown piping outside containment from penetration #23 to non-regenerative heat exchanger are as tabulated in Table 5-1. These locations have been determined on the basis

of calculated stress values and the criteria given in Section 2.2.1 for ANSI B31.1 piping breaks. These consist of:

a. Terminal points which are located at the nonregenerative heat exchanger, the containment penetration and the intersection with 12" AC-8.

b. Two intermediate points for each piping run and branch run having the highest calculated stress values.

There are no available stress analysis results for CVCS high energy piping other than the portion described in the above paragraph. Full area breaks were postulated to occur at any location and any orientation for the portion without stress results.

For the portion of CVCS piping with stress analyses results, the combined stress values at postulated break locations due to thermal expansion, pressure, weight and seismic loadings were computed and summarized in Table 5-1. At no point do the calculated stress values exceeded 0.8 S_a or 0.8 $(S_h + S_a)$.

5.2.2 Required Equipment

The equipment needed for a design basis break in the CVCS letdown and charging lines is given in Table 8.0-1. Operability of this equipment provides for reactor trip and the capability to place and maintain reactor in a cold shutdown condition. Required equipment includes associated piping, cables and structures required for the equipment to perform its function.

5.2.3 Protection from Potential Pipe Whip Damage and Jet Impingement

Rupture restraints and impingement barriers were found not necessary to protect the required equipment listed in Table 8.0-1.

5.2.4 Protection from Adverse Environmental Conditions

Required equipment exposed to environmental conditions were analyzed for their capability to perform their function and were found satisfactory in the anticipated environment as described in Section 10.0.

TABLE 5-1

Stress Values -- CVCS From Anchors @ EL. 237.50'

& EL. 241.25' to Non-Regenerative Heat Exchanger

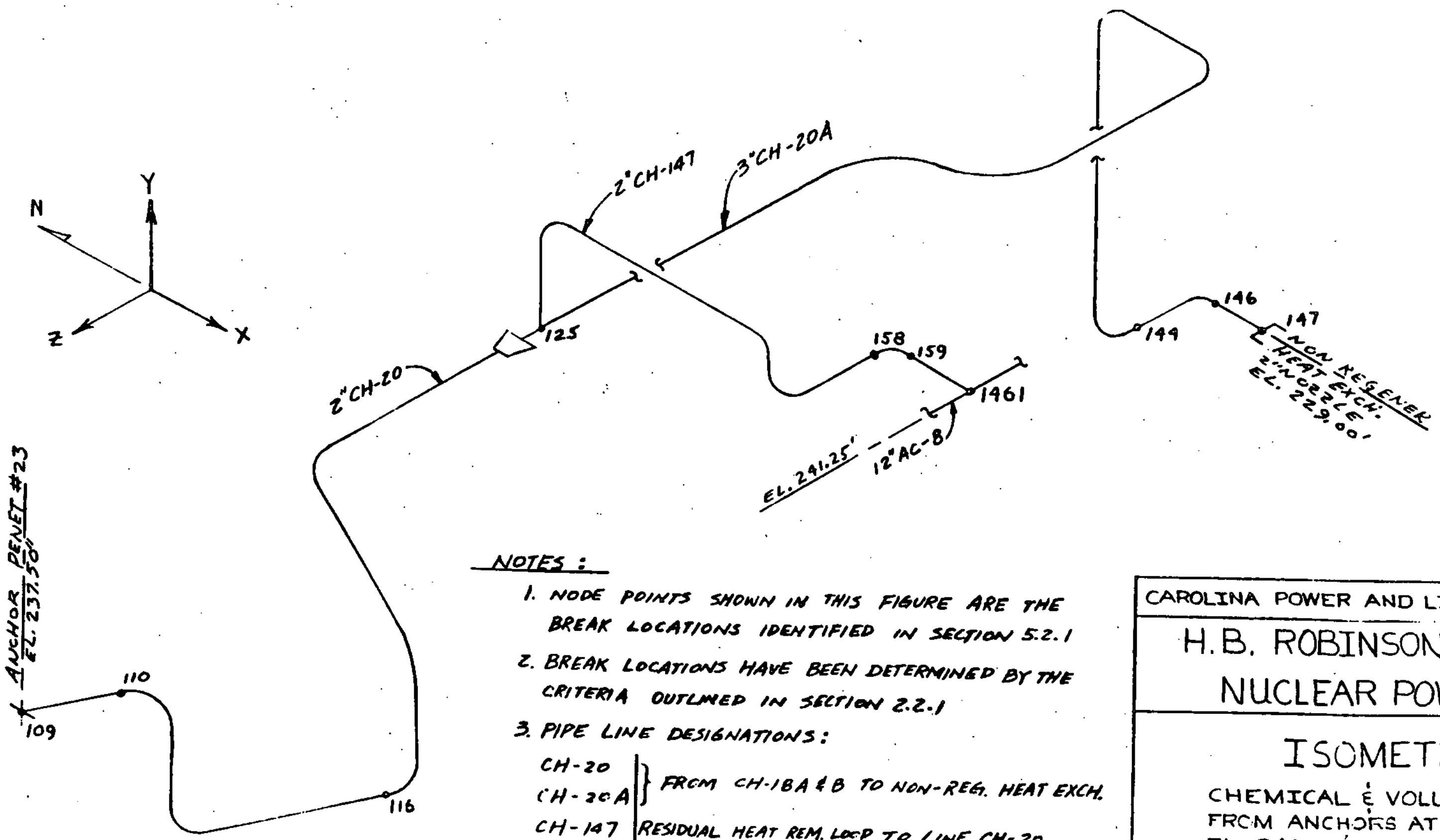
Allowable Stress Values:
(0.8 S allowable)
Operational Plus Seismic Stresses < 34,700 psi

Thermal Stresses < 21,980 psi

Node No.*	Pressure	Gravity	Thermal	Seismic (OBE)	Total
109	936	2,000**	7,369	2,166	12,471
110	936	2,000	11,757	1,828	16,521
116	936	2,000	10,166	2,527	15,629
125	994	2,000	2,970	1,695	7,659
144	1,139	2,000	7,598	1,351	12,088
146	1,139	2,000	11,858	2,741	17,738
147	1,139	2,000	11,828	3,333	18,300
158	994	2,000	19,609	7,219	29,822
159	994	2,000	20,480	5,463	28,937
1461	994	2,000	9,135	5,170	17,299

*Node Numbers refer to figure 5-1

**Assumed maximum dead weight stress



NOTES :

1. NODE POINTS SHOWN IN THIS FIGURE ARE THE BREAK LOCATIONS IDENTIFIED IN SECTION 5.2.1
2. BREAK LOCATIONS HAVE BEEN DETERMINED BY THE CRITERIA OUTLINED IN SECTION 2.2.1
3. PIPE LINE DESIGNATIONS:

CH-20 CH-20A	FROM CH-18A & B TO NON-REG. HEAT EXCH.
CH-147	RESIDUAL HEAT REM. LOOP TO LINE CH-20

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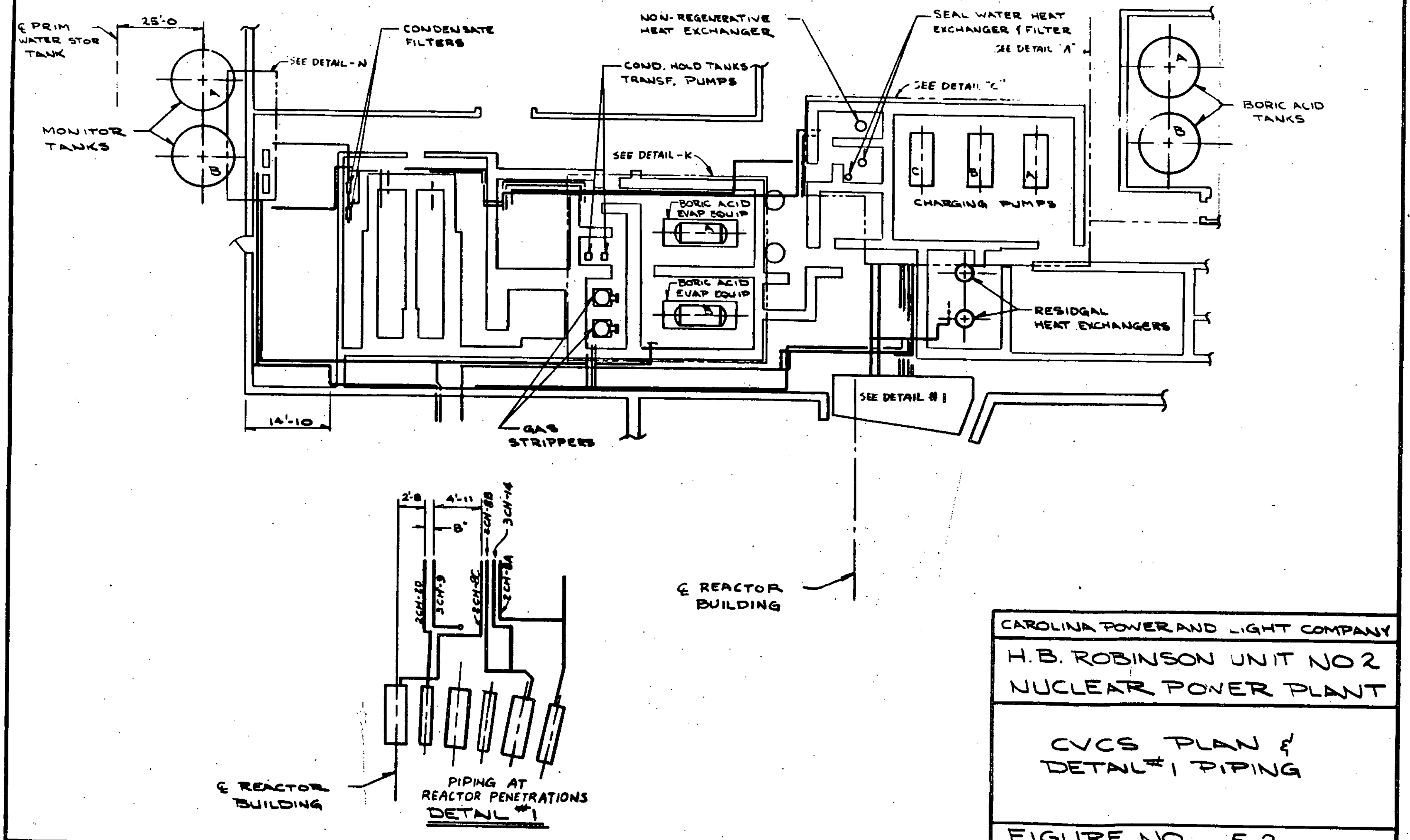
ISOMETRIC

CHEMICAL & VOLUME CONTROL FROM ANCHORS AT EL. 237.50' & EL. 241.25' TO NON REGENERATIVE HEAT EXCHANGE

FIGURE NO. 5-1



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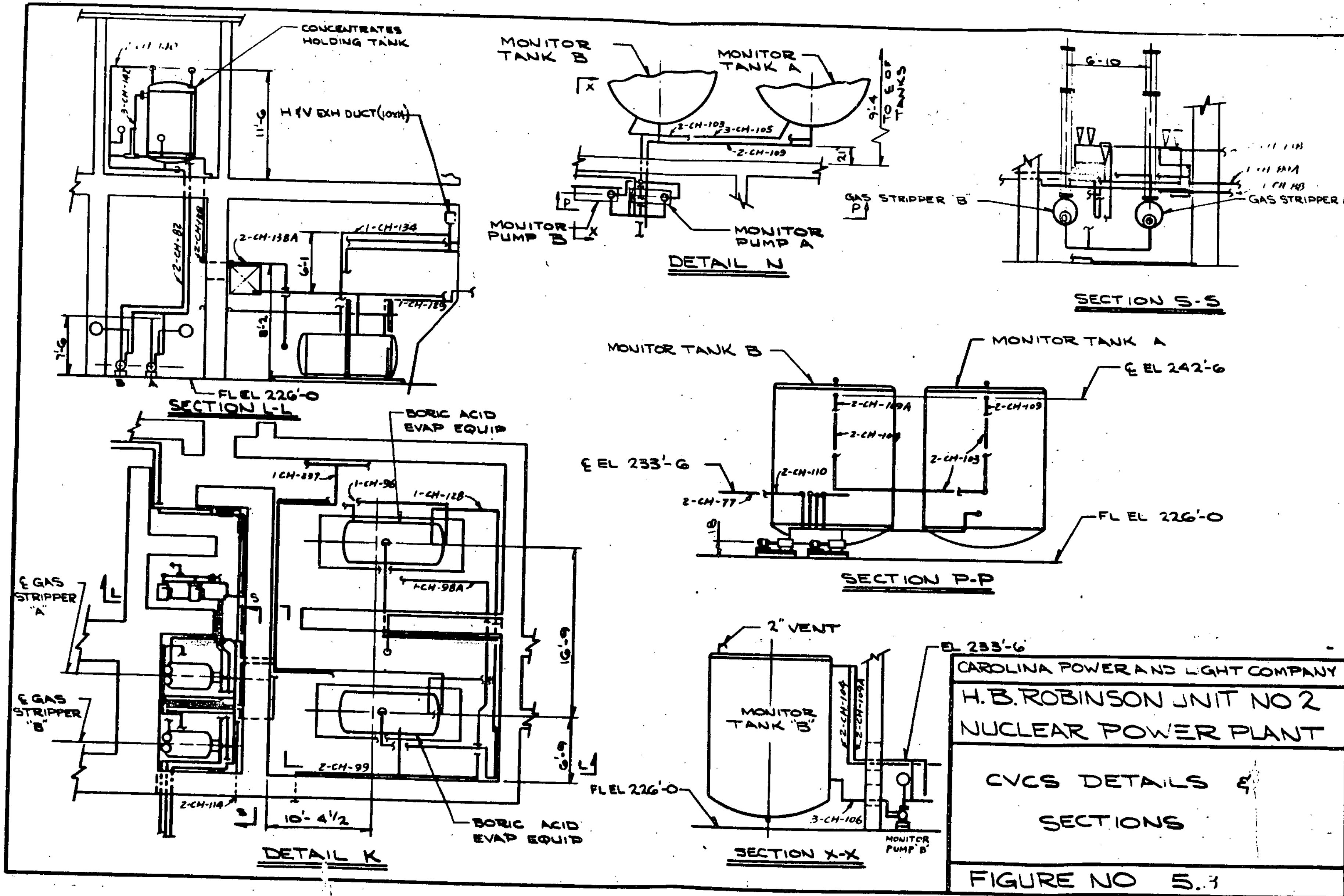
NUCLEAR POWER PLANT

CVCS PLAN & DETAIL #1 PIPING

FIGURE NO 5.2

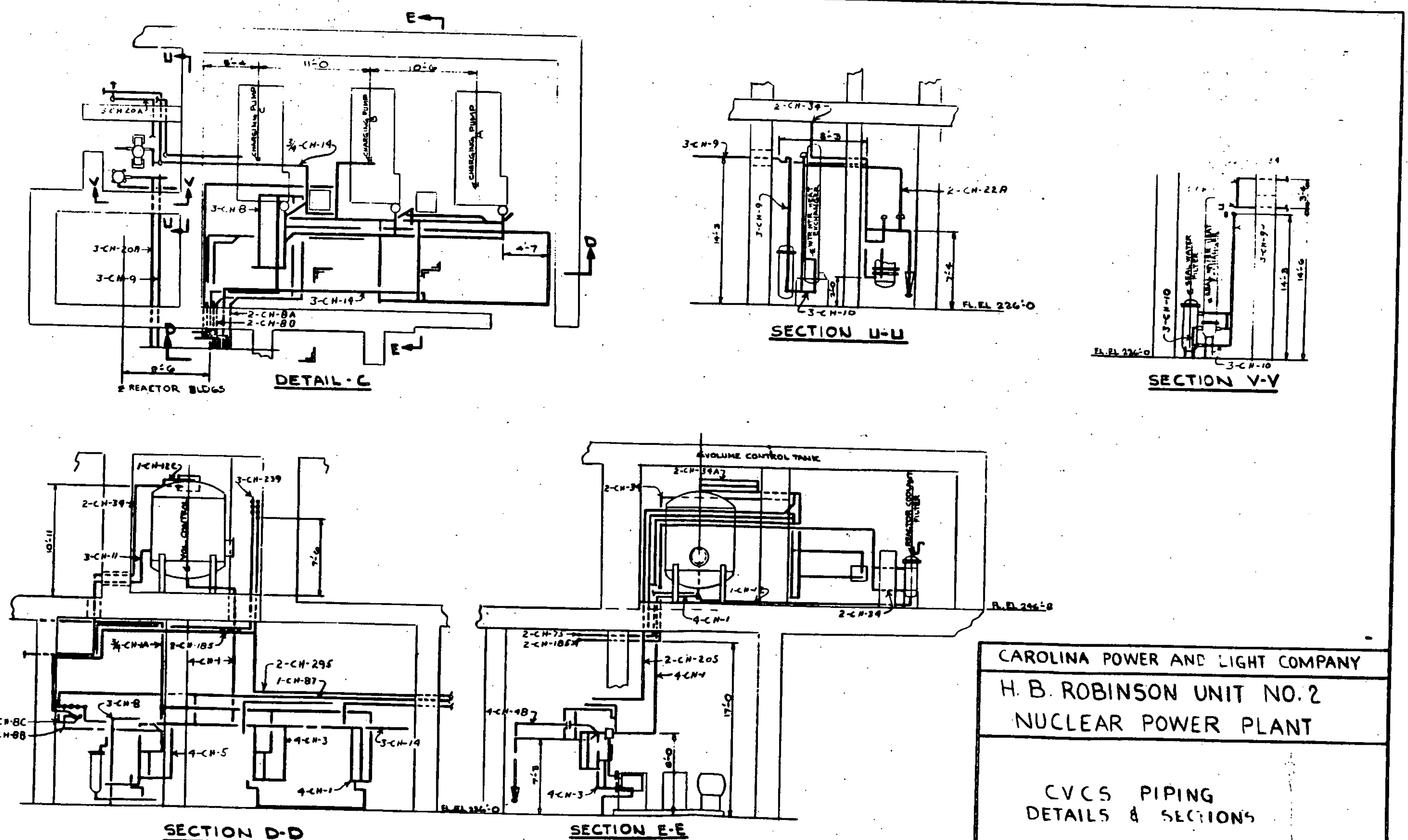


CAROLINA POWER AND LIGHT COMPANY

H. B. ROBINSON UNIT NO 2
NUCLEAR POWER PLANT

CVCS DETAILS & SECTIONS

FIGURE NO 5.3

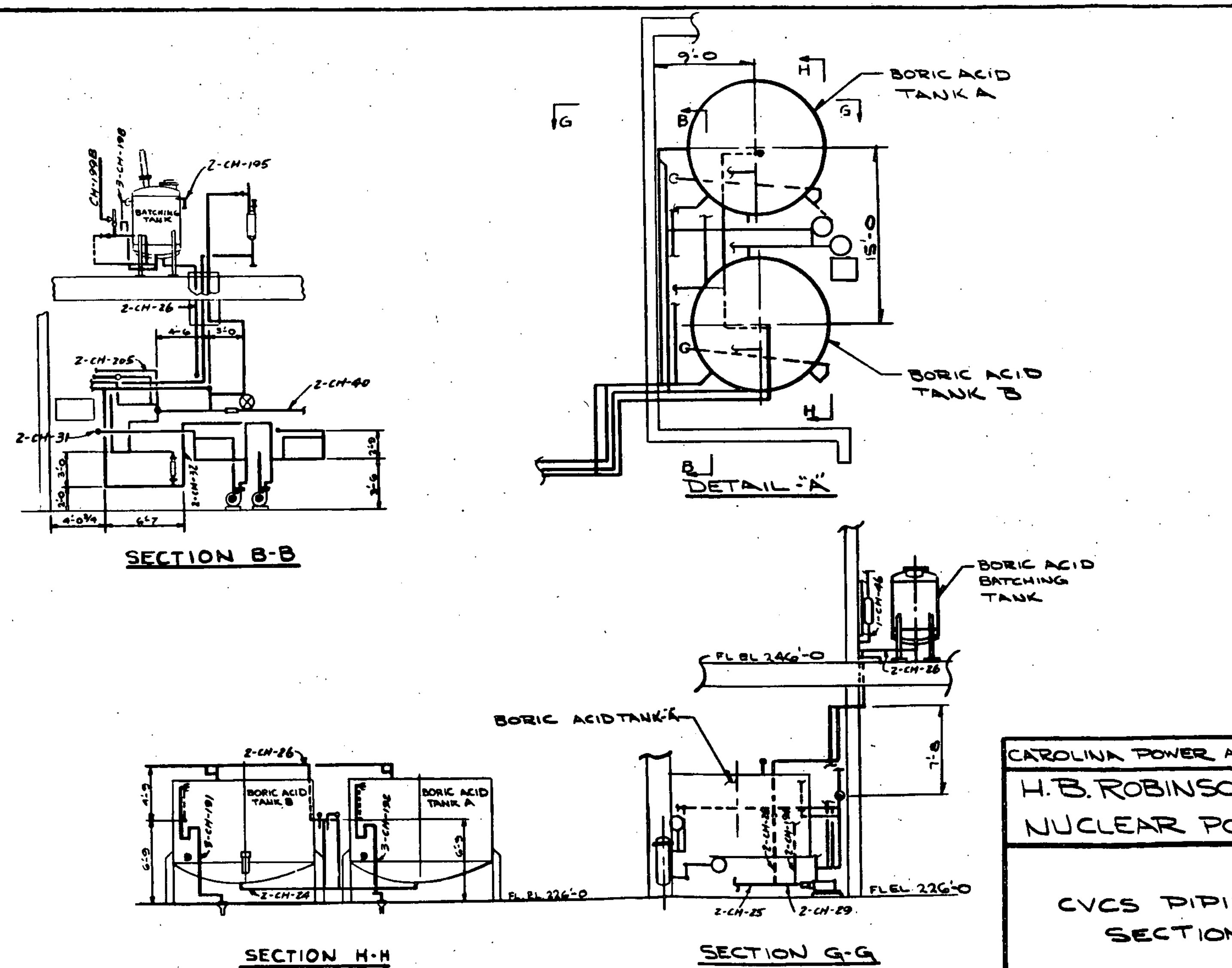


CAROLINA POWER AND LIGHT COMPANY

H. B. ROBINSON UNIT NO. 2
NUCLEAR POWER PLANT

CVCS PIPING
DETAILS & SECTIONS

FIGURE NO. 5.4



CAROLINA POWER AND LIGHT COMPANY

H. B. ROBINSON UNIT NO 2
NUCLEAR POWER PLANT

CVCS PIPING DETAIL & SECTIONS

FIGURE NO 5.5

6.0 STEAM GENERATOR BLOWDOWN LINE ROUTING AND RUPTURE EVALUATION

6.1 ROUTING DESCRIPTION

Blowdown lines for steam generators A, B and C leave the reactor building at the mechanical penetration area at elevations 234'-6" and 237'-6". Then all three lines drop to elevation 233'-8"; run parallel to each other toward the south and enter the turbine building between the ground and mezzanine floors to the steam generator drain tank. One 3/8 inch line branches out from each 2 inch blowdown line, passes through a steam generator blowdown sample heat exchanger and returns to the steam generator drain tank through the radiation monitor sample line. A 2 inch pipe with locked closed valves branches out from each blowdown line upstream of the tank inlet to the steam generator drain pump through a 4 inch suction pipe. Discharge of the drain pump is normally routed to the condenser discharge pipe. The above routing is shown in layout drawing Figure 6-1.

6.2 DESIGN BASIS BREAKS

6.2.1 Description of Break Locations

There are no stress analysis results available for the piping of this high energy system and, therefore, full area breaks were postulated to occur at any location and any orientation. Design basis cracks were not considered separately, since they are a less severe accident than the full area break.

6.2.2 Required Equipment

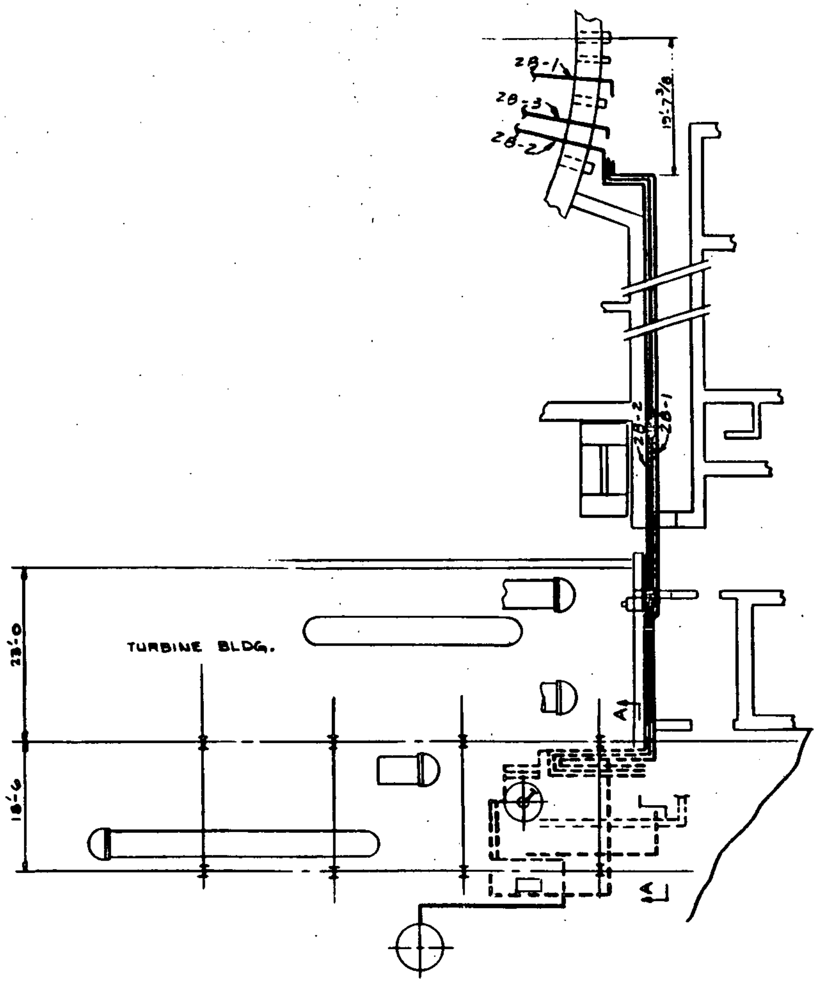
The equipment needed for a design basis break in the steam generator blowdown lines is given in Table 8.0-1. Operability of this equipment provides for reactor trip and the capability to place and maintain reactor in a safe shutdown condition. Required equipment includes associated piping, cables and structures required for the equipment to perform its function.

6.2.3 Protection from Potential Pipe Whip Damage and Jet Impingement

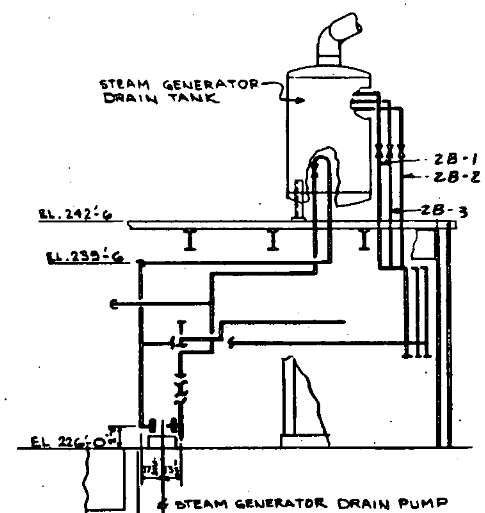
Rupture restraints and impingement barriers were found not necessary to protect the required equipment listed in Table 8.0-1.

6.2.4 Protection from Adverse Environmental Conditions

Required equipment exposed to environmental conditions were analyzed for their capability to perform their function and were found satisfactory in the anticipated environment as described in Section 10.0.



PLAN



SECTION A-A
SCALE 1/8=1'-0

CAROLINA POWER AND LIGHT COMPANY

H. B. ROBINSON UNIT NO. 2
NUCLEAR POWER PLANT

STEAM GENERATOR BLOWDOWN SYSTEM
PLAN & SECTION A A

FIGURE NO. 6-1

7.0 STEAM SUPPLY TO AUXILIARY FEEDWATER PUMP TURBINE LINES ROUTING AND RUPTURE EVALUATION

7.1 ROUTING DESCRIPTION

The 2 inch steam supply lines to the auxiliary feedwater pump turbine from steam generators A, B and C originate at main steam line branch connections upstream of main steam isolation valves at the south of the containment building. These field run lines route through the support tower to the turbine building and join at the 3 inch auxiliary feedwater pump steam inlet piping to the south-west corner of the auxiliary building. The above routing is shown in layout drawings Figures 4-7 and 4.8.

7.2 DESIGN BASIS BREAKS

7.2.1 Description of Break Locations

Full area breaks were postulated to occur at any location and any orientation. Design basis cracks were not considered separately, since they are a less severe accident than the full area break.

7.2.2 Required Equipment

The equipment needed for a design basis break in the steam supply to auxiliary feedwater pump turbine lines is given in Table 8.0-1. Operability of this equipment provides for reactor trip and the capability to place and maintain reactor in a safe shutdown condition. Required equipment includes associated piping, cables and structures required for the equipment to perform its function.

7.2.3 Protection from Potential Pipe Whip Damage and Jet Impingement

Rupture restraints and impingement barriers were found not necessary to protect the required equipment listed in Table 8.0-1.

7.2.4 Protection from Adverse Environmental Conditions

Required equipment exposed to environmental conditions were analyzed for their capability to perform their functions and were found satisfactory in the anticipated environment since these lines are located in the open turbine building.

8.0 PLANT OPERATION FOLLOWING PIPE RUPTURE

8.1 CONTROL ROOM HABITABILITY

The control room will be maintained habitable and its essential equipment functional for design basis events. The capability to bring the reactor to a cold shutdown condition from the control room will be maintained.

8.2 REDUNDANCY

The capability to mitigate the consequences of an accident and bring the reactor to a cold shutdown condition will be assured. Loss of redundancy of equipment (as a consequence of the postulated break) required to mitigate the consequences of an accident and obtain a safe hot shutdown will not be permitted. Environmentally induced failures caused by leak or rupture which would not in itself result in protective action, but may disable protective equipment will also be considered. In this regard a loss of redundancy will be permitted but a loss of function will not be permitted. For such situations the capability for bringing the plant to a safe cold shutdown will be assured.

8.3 OPERATION OF NEEDED EQUIPMENT

As part of the high energy pipe break review, equipment required for a safe cold shutdown was reviewed in conjunction with postulated pipe ruptures or design basis cracks. This review considered required safeguards equipment and equipment used for obtaining a safe cold shutdown assuming the complete loss of offsite power. In the event offsite power is lost, however, the preferred mode of operation would be to hold the plant in a safe hot shutdown

condition until such a time that offsite power could be restored (in the unlikely event it was lost) and then cool the plant down to a safe cold shutdown in a normal manner.

Equipment required to mitigate the consequences of the high energy line rupture and obtain cold shutdown conditions is presented in Table 8.0-1.

8.4 EMERGENCY PROCEDURES

Appendix 8B presents the general plant emergency procedures. It is deemed appropriate to allow for assessment of the incident prior to ultimately bringing the reactor to a cold shutdown condition.

8.5 PLANT ANALYSES

Initial review of postulated high energy line breaks with regard to equipment required for a safe shutdown indicate that only two areas require further investigation.

One area is in the mechanical penetration area where some electrical cable trays pass near the vicinity of some of the blowdown lines. Further investigation is proceeding to determine the environmental effect on the cabling in these trays required to mitigate the consequences of a blowdown line break.

The other area being investigated further is the area on the turbine deck where some of the steam line pressure transmitters are located in the vicinity of the feed lines. This is discussed further in Section II.

TABLE 8.0-1

REQUIRED SAFEGUARDS AND EQUIPMENT
REQUIRED FOR SAFE SHUTDOWN

A. Pumps

1. Safety Injection Pumps
2. Residual Heat Removal Pumps
3. Service Water Pumps
4. Component Cooling Water Pumps
5. Auxiliary Feedwater Pumps

B. Tanks

1. Refueling Water Storage Tank
2. Boron Injection Tank
3. Component Cooling Water Surge Tank
4. Condensate Storage Tank (or other appropriate source for auxiliary feed)

C. Heat Exchangers

1. Residual Heat Removal Heat Exchangers
2. Component Cooling Water Heat Exchanger

D. Valves

1. Refueling Water Suction to S.I. Pumps
2. Refueling Water Suction to Residual Heat Removal Pumps
3. Boron Injection Tank Isolation
4. Boron Injection Tank to Boric Acid Tank Isolation
5. Residual Heat Removal Flow Control
6. Residual Heat Removal Suction
7. Pressurizer Safety

***8. Main Steam Stop (Trip closed feature)

9. Main Steam Safety

10. Main Steam Power Operated Relief (for manual operation)

**11. Main Feedwater Control (trip closed feature)

12. Essential Service Water

13. Essential Component Cooling Water

14. Auxiliary Feed Pump Suction and Discharge

15. Auxiliary Feed Control

**16. Main Feedwater Isolation Valves (trip closed feature)

**17. Bypass Feedwater Control (trip closed feature)

***18. Main Steam Line Stop Valve Bypass (trip closed feature)

***19. Steam Generator Blowdown Isolation (automatic closure feature)

20. Letdown Isolation

**Required for steam line and steam generator blowdown line break only.

***Required for steam line, feed line, and steam generator blowdown line break only.

REQUIRED SAFEGUARDS AND EQUIPMENT
REQUIRED FOR SAFE SHUTDOWN

E. Instrumentation and Indications Available to the Operator

1. Wide range reactor coolant system pressure
2. Pressurizer Level
3. Wide Range Reactor Coolant System Temperature (Preferable T_{hot} for each loop)
4. Containment Pressure
5. Each Steam Generators Pressure
6. Each Steam Generators Level (Wide and narrow range)
7. Component Cooling Water Heat Exchangers Pressure or Flow
8. Service Water Heat Exchangers Flow or Pressure
9. Heat Tracing (as required for items on this list)
10. RHR Interlocks
11. Reactor trip and safeguards actuation channels including sensors, circuitry, and processing equipment (the protection circuits used to trip the reactor on undervoltage underfrequency, and turbine trip may be excluded).
12. Suction and Discharge Pressure of all Required Pumps

F. Electrical

1. Rx Trip Breakers
2. Batteries
3. Safeguards Busses
4. Diesel Generators and Associated Auxiliary Equipment
5. Containment Safeguards Fan Coolers
6. Emergency Lighting
7. Circuits and/or Equipment Required to Trip the Main Feedwater Pumps
8. Controls for Defeating Safety Injection Actuation During a Cooldown and Depressurization
9. Manual control of required pumps available to the operator

G. Miscellaneous

1. Control room equipment must not be damaged to an extent where any equipment will be spuriously actuated or any of the equipment contained elsewhere in this list cannot be operated.
2. Control Room Ventilation.
3. Capability for obtaining a reactor coolant system sample
4. Support systems for the above equipment such as long term diesel fuel storage, battery chargers, and a long term water supply for the auxiliary feedwater system must be available.
5. Equipment includes required structures, associated piping and valves (e.g. miniflow) and instrument actuation channels including sensors, processing equipment and associated circuitry for the equipment to perform its function.

9.0 DESCRIPTION OF PIPE WHIP ANALYSIS

Restraints are provided to prevent pipe whip where there is any possibility that whip following a pipe rupture would damage systems, components or structures that are needed to mitigate the consequences of that pipe rupture. These restraints are designed using an equivalent static load analysis based on system operating pressure and the postulated break area as given in Section 2.

Dynamic response and energy absorption capability of the system is implicitly provided for by the inherent ductility of the pipe and the support. Stresses in pipe whip restraints based on these analytical assumptions are conservatively limited to 90 percent of yield for membrane plus bending.

Pipe whip restraints are so spaced so as to preclude the formation of plastic hinge mechanism in the piping system.

Detailed evaluation of the potential for pipe whip of high energy lines outside containment on plant safety not previously considered in design indicate the potential for a pipe whip in the feedwater line between heaters 5 and 6 while could impact on that portion of the auxiliary building structure which houses the safeguards switchgear room.

Analysis of the motion the feedwater line in the event of postulated rupture indicate it would reach a normal impact velocity on the 18 inch thick switchgear room reinforced concrete wall of 190 ft./sec. With an impact area of 1.396 sq. ft., the depth of penetration is determined as ~1.3 inches. The equivalent static impact load is 2.1 kips. The capacity of the section is approximately 10 kips. It can, therefore, be concluded the impact of the feedwater line would not effect the safeguards switchgear.

10.0 DESCRIPTION OF COMPARTMENT ENVIRONMENTAL EFFECTS ANALYSIS

The Transient Mass Distribution (TMD) code is used to determine the local pressure on building walls and floor, and compartment temperatures. This code calculates the pressure and temperature transients, in a group of interconnected compartments. The blowdown rate and duration depend on the break location and size.

In the compartment where pipe rupture occurs, the mass and energy input rates are based on the blowdown for the specific break. The initial conditions of all compartments are identical. A sequence of calculations is performed to determine the mass and energy transferred between compartments for a short time interval.

The solution is obtained in the following manner. The control volume technique is used to represent the physical geometry of the system. The technique allows specification of local volumes, lengths, areas and flow resistances.

The only enclosed volume subject to pressure and temperature buildup which would potentially violate the criteria for plant operation following rupture is the pipe penetration gallery. The results of the TMD Code Analysis of the enclosed pipe penetration gallery for the pipe ruptures postulated in the gallery indicate a calculated pressure buildup of 0.2 psi and a maximum temperature of 2.4°F. This consequence is the result of the limiting postulated steam generator blowdown line rupture having a blowdown mass and energy release rate as shown in Table 10-1. The operating condition of the various lines considered are summarized in Table 10-2. The pipe penetration gallery contains a volume of 40,700

cu. ft. and possessing a blowdown area of approximately 108 sq. ft. as shown in Figures 10-1 to 10-5. These limiting environmental conditions as determined from the limiting postulated break will have no effect on the structural adequacy of the auxiliary building or on plant operation.

The effects of flooding as the result of a postulated crack in the piping in the charging pump room and in the motor driven auxiliary feedwater pump room have been evaluated. Both of the systems in the effected rooms operate at temperatures below 130°F so that pressure and temperature effects are not a consideration.

The crack discharge flow rate for the charging pump line is 107 lbs/sec. The pump room is serviced by two 4 inch floor drains discharging into a 6 inch main. Assuming a steady state rate of discharge the depth of water developed in the pump room is 7.0 inches.

The crack discharge rate for the motor driven auxiliary feedwater line is 70 lbs/sec. This feedwater pump room is serviced by one 4 inch drain. Again based on steady state rate of discharge the computed head is 3.1 feet. However, this room is located on grade and communicates directly to the outside through double steel doors. It is anticipated therefore the water level in the room would not exceed 1.0 feet. It is therefore concluded plant operability criteria as presented in Section 8.0 would not be violated in the event of either a charging or motor driven auxiliary feedwater line rupture.

TABLE 10-1

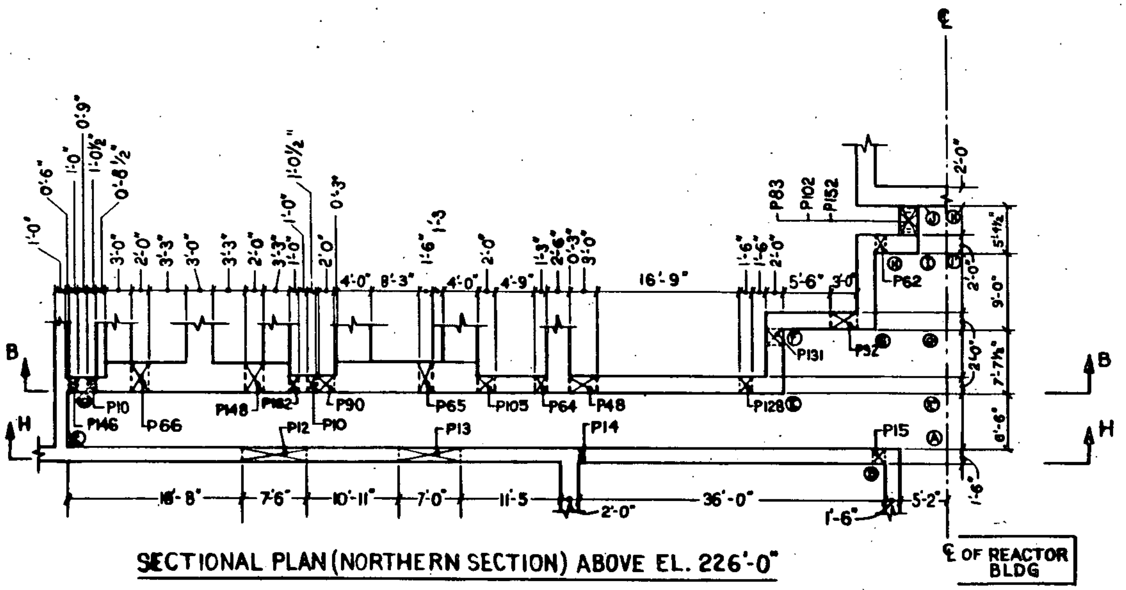
MASS AND ENERGY BLOWDOWN RESULTING
FROM POSTULATED RUPTURE OF STEAM GENERATOR
BLOWDOWN LINE

Time	Mass Release Rate (lbs/sec)	Energy Release Rate (BTU/Sec)
$0 < t \leq 1$	484	125210
$1 < t \leq 4$	269.6	77934
$4 < t$	27.6	15329

TABLE 10-2

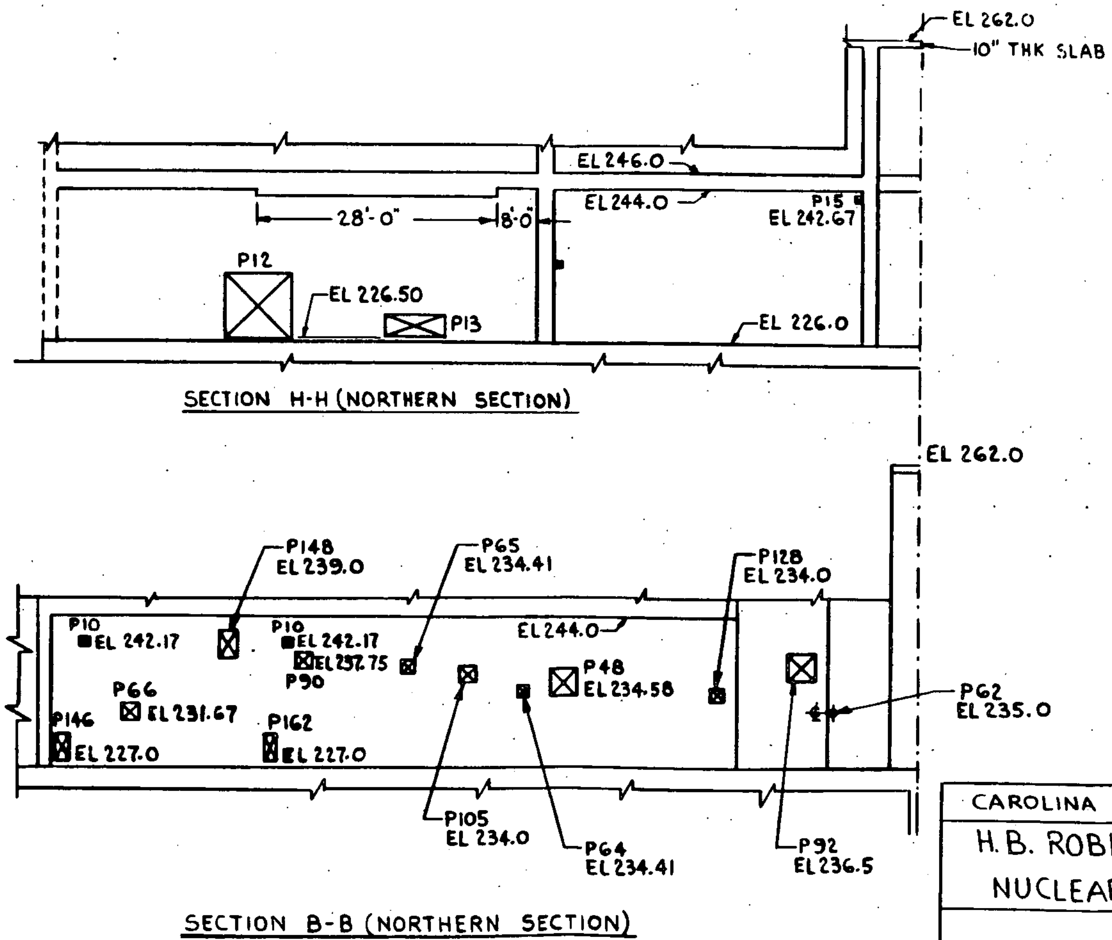
SUMMARY OF OPERATING PARAMETERS
ASSOCIATED WITH ASSUMED PIPE RUPTURE

Line	Size	Temperature °F	Pressure psig
1. Steam Generator Blowdown Line	2"	450	1000
2. Reactor Letdown Line		380	285
3. Charging Line	3"	130	2560
4. Motor Driven Auxiliary Feedwater Line		70	



SECTIONAL PLAN (NORTHERN SECTION) ABOVE EL. 226'-0"

CAROLINA POWER AND LIGHT COMPANY
H. B. ROBINSON UNIT NO. 2 NUCLEAR POWER PLANT
SECTIONAL PLAN NORTHERN SECTION (ABOVE EL 226'-0")
FIGURE NO. 10 1

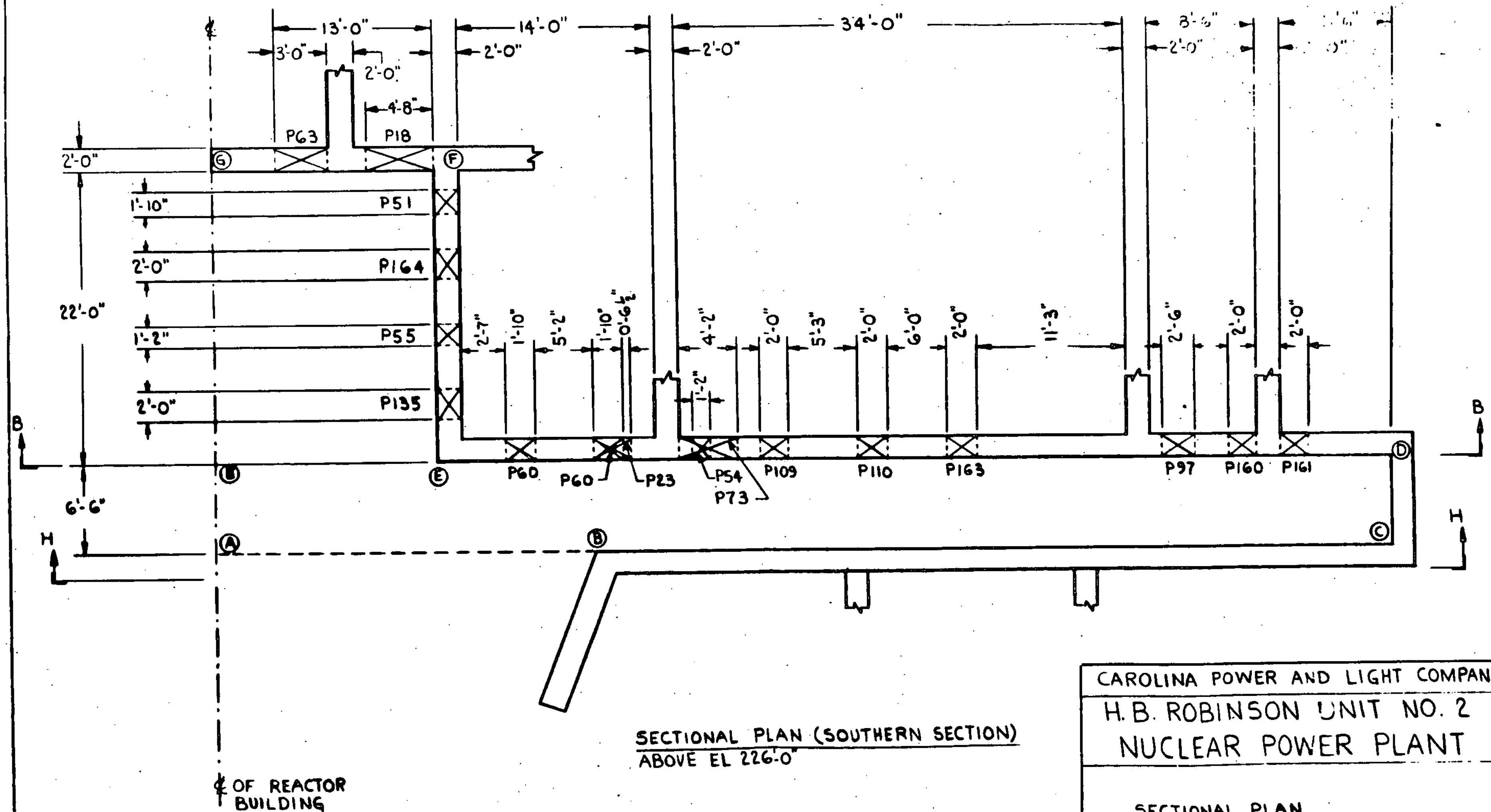


CAROLINA POWER AND LIGHT COMPANY

H.B. ROBINSON UNIT NO. 2
NUCLEAR POWER PLANT

SECTION H-H
SECTION B-B
(NORTHERN SECTIONS)

FIGURE NO. 10-2



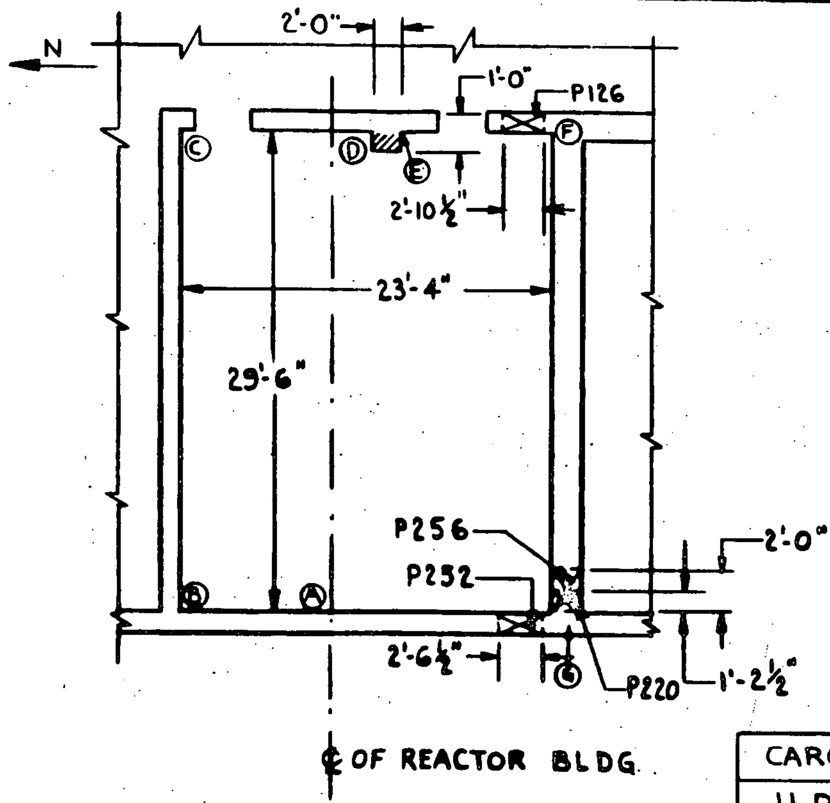
SECTIONAL PLAN (SOUTHERN SECTION)
ABOVE EL 226'-0"

CAROLINA POWER AND LIGHT COMPANY

H. B. ROBINSON UNIT NO. 2
NUCLEAR POWER PLANT

SECTIONAL PLAN
SOUTHERN SECTION
(ABOVE EL 226'-0")

FIGURE NO. 10-3



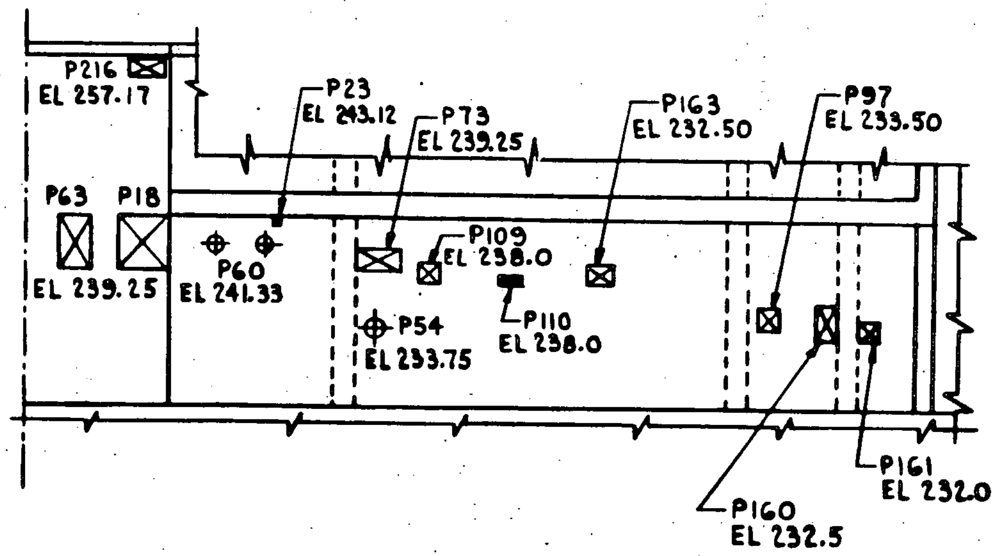
SECOND FLOOR PLAN

CAROLINA POWER AND LIGHT COMPANY

H.B. ROBINSON UNIT NO. 2
NUCLEAR POWER PLANT

SECOND FLOOR PLAN - PARTIAL

FIGURE NO. 10-4

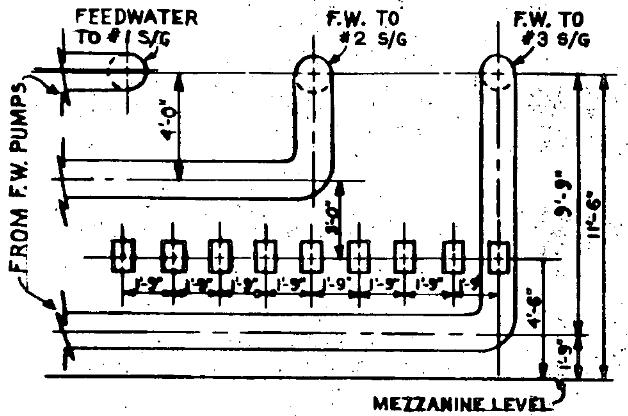


CAROLINA POWER AND LIGHT COMPANY

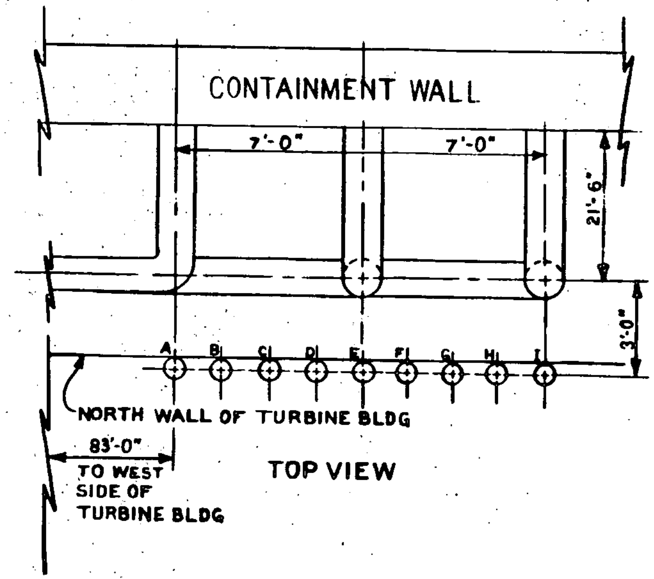
H. B. ROBINSON UNIT NO. 2
NUCLEAR POWER PLANT

PARTIAL PLAN-SECOND FLOOR

FIGURE NO. 10-5



VIEW FROM TURBINE BLDG



TOP VIEW

CAROLINA POWER AND LIGHT COMPANY

H. B. ROBINSON UNIT NO. 2
NUCLEAR POWER PLANT

LOCATION OF STEAM PRESSURE TRANSMITTER RELATIVE TO MAIN FEED LINES

FIGURE NO. 10-6

11.0 DESCRIPTION OF JET IMPINGEMENT LOAD ANALYSIS

The jet impingement load is defined as the load on a component (piping, equipment or structure) of the undeflected jet from an instantaneous circumferential or longitudinal break in a high energy pipe. Unless more rigorously determined at the point of rupture, the jet pressure p is assumed equal to system operating pressure, and the jet area is assumed to be the effective break area (A_o). The jet flow away from the point of rupture is assumed to diverge at an included angle ($\emptyset$) of 20^o.

$$A_i = [L_1/2 + (L_3) \text{ Tan } (\emptyset/2)] \, [L_2/2 + (L_3) \text{ Tan } (\emptyset/2)] \, \pi \qquad (1.1)$$

where:

L_1 = width of break

L_2 = length of break

L_3 = distance to target

$\emptyset$ = divergence angle = 20°

The total force exerted by the jet is assumed constant. Jet impingement pressure on a barrier as a function of perpendicular distance from the plane of the assumed break is determined by equation 1.2.

$$p_i = p_o \frac{A_o}{A_i} \qquad (1.2)$$

Results of the jet impingement load evaluation indicate that an impingement shield should be installed to protect steam pressure transmitters located adjacent to the feedwater line as shown in Figure 11.1 in the event of a feedwater line crack occurrence. The shield is necessary to insure the insulation protecting the

transmitters would not be blown off by the postulated jet with a resultant rise in transmitter temperature above 212°F. This temperature limitation is necessary to assure the proper function of the electronic portion of the transmitter. The shield in accordance with the criteria presented in Section 2.2.3 and the physical proximity of the transmitters to the feedwater line, will be designed for a line load of .745 K/in over 8 inches of length. Either a horizontal or a vertical line load will be considered. Structural design criteria for the shield is the same as that used for pipe whip restraints.

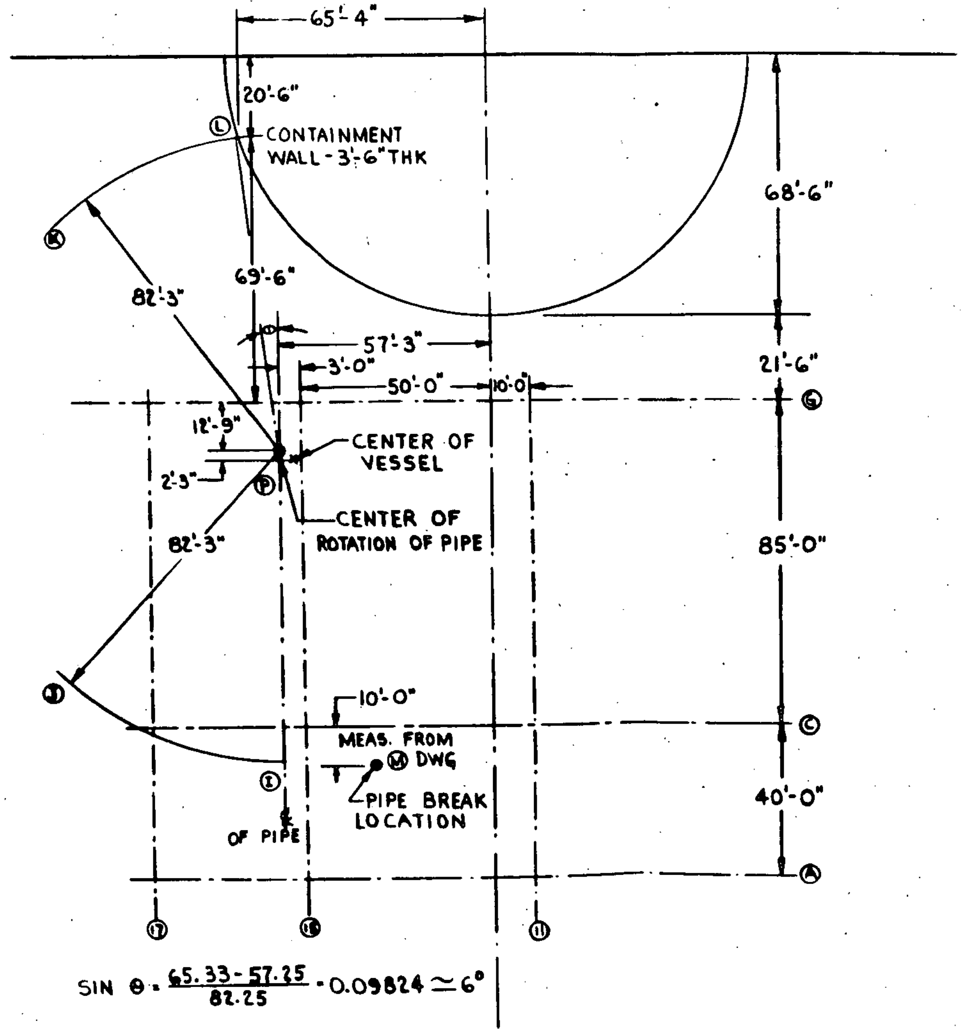
12.0 CONTAINMENT INTEGRITY

The present anchorage and pipe whip restraint system design will preclude any damage to the Containment Vessel, due to any of the postulated breaks except in the case associated with the postulated rupture of a steam line downstream from the 72 inch diameter steam header as the limiting case.

The limiting break location is at the terminal end inlet to the turbine. For such a postulated circumferencial break, the steam line is assumed to hinge at the main steam header. As shown in Figure 12.1, such a postulated plastic rupture would permit the rotating steam line to impact containment. Calculations for a resultant pipe whip velocity of 569 ft./sec. and an effect mass of 25.04 lb-sec.^{2}/in. and contact area of 17.54 sq. ft. results in a containment penetration of about 8.0 inches. The equivalent static impact load is determined as 1175 kips.

No detailed structural analysis of the containment shell was performed for this missile effect as an impulsive load, but a comparison was made with the equivalent static load as determined by the forcing function developed by Riera[1] for the analysis of the 3 Mile Island containment structure for aircraft impact. The effective load determined for the steam line break is approximately 20 percent of the load used in the successful evaluation of 3 Mile Island containment. It is, therefore, concluded that the containment will maintain its integrity in the event of a postulated steam line break.

[1] Riera, J. D., "On the Stress Analysis of Structures Subjected to Aircraft Impact Forces", Nuclear Engineering and Design, North Holland Publishing Co. Amsterdam, 1968, pp 415-426.



CAROLINA POWER AND LIGHT COMPANY

H.B. ROBINSON UNIT NO. 2
NUCLEAR POWER PLANT

CONTAINMENT VESSEL & MAIN STEAM HEADER

FIGURE NO. 12-1

13.0 CONCLUSIONS AND RECOMMENDATIONS

The results of the study regarding pipe rupture outside containment for the H. B. Robinson Unit No. 2 indicate the plant is adequately protected against such postulated ruptures with the single additional requirement that a jet impingement shield is required to protect the steam system pressure transmitters from a postulated crack in the feedwater line as described in Section 11.0.