

EVALUATION OF THE EFFECTS OF POSTULATED
PIPE FAILURES OUTSIDE CONTAINMENT
FOR
MCGUIRE NUCLEAR STATION

May 2, 1977

REPORT NO. MDS/PDG-77-1

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TABLE OF CONTENTS

	<u>Page</u>
FOREWORD	i
1.0 INTRODUCTION	1-1
2.0 PLANT DESCRIPTION	2-1
2.1 General	2-1
2.2 Piping Layout	2-1
3.0 CRITERIA FOR POSTULATING RUPTURES	3-1
3.1 Piping Systems Subject to Postulated Pipe Ruptures	3-1
3.1.1 High Energy Piping Systems	3-1
3.1.2 Moderate Energy Piping Systems	3-2
3.2 Break Location and Configuration	3-2
3.2.1 Break Locations in Duke Class B, C, & F Piping Runs	3-2
3.2.2 Break Locations in Other Piping Runs	3-3
3.2.3 Break Configuration	3-4
3.3 High Energy Piping Outside Containment	3-6
3.4 Moderate Energy Piping Outside Containment	3-6
4.0 SYSTEMS AND EQUIPMENT EVALUATION	4-1
4.1 General	4-1
4.2 Protection From Rupture of High Energy Piping Systems	4-2
4.2.1 High Energy Piping Systems Physically Separated From Essential Structures and Systems	4-2
4.2.2 Piping Systems Enclosed Within Structures	4-4
4.2.3 Other High Energy Piping Systems	4-4
4.3 Protection From Through-Wall Cracks in Moderate Energy Piping	4-5
4.4 Protection Criteria and Assumptions	4-5
4.5 Essential Systems	4-8
4.5.1 Normal Shutdown	4-9
4.5.2 Emergency Shutdown	4-10
5.0 DAMAGE EVALUATION	5-1
5.1 Pipe Whip/Jet Impingement	5-1
5.1.1 Break Criteria	5-1
5.1.2 Establish Plant Arrangement and Layout	5-1
5.1.3 Preliminary Interaction Matrices	5-1
5.1.4 Final Interaction Matrices	5-2
5.1.5 Pipe Whip Model	5-3
5.1.6 Jet Model	5-5

5.2	Compartment Pressurization	5-5
5.2.1	TMD Code-Short Term Analysis	5-5
5.2.2	RELAP Code	5-7
5.2.3	Hand Calculations	5-10
5.3	Flooding Effects	5-11
5.3.1	Discharge Rates	5-11
5.3.2	Moderate Energy Piping	5-11
5.3.3	High Energy Piping	5-12
5.4	Environmental Consequences	5-12
6.0	SUMMARY AND CONCLUSIONS	6-1
6.1	Pipe Whip and Jet Impingement Effects	6-1
6.1.1	Main Steam and Feedwater Systems	6-1
6.1.2	Steam Generator Blowdown System	6-2
6.1.3	Auxiliary Feedwater System	6-3
6.1.4	Safety Injection System (Upper Head)	6-3
6.1.5	Boron Thermal Regeneration System	6-4
6.1.6	Chemical and Volume Control System	6-4
6.1.7	Main Steam Supply to Auxiliary Equipment	6-5
6.2	Compartment Pressurization Effects	6-5
6.2.1	Main Steam and Feedwater	6-5
6.2.2	Upper Head Injection	6-6
6.2.3	Steam Generator Blowdown	6-8
6.2.4	Letdown Line, CVCS System	6-8
6.3	Flooding Effects	6-8
6.3.1	Auxiliary Feedwater Pump Room	6-9
6.3.2	Containment Spray/RHR Pump Rooms	6-9
6.4	Environmental Effects	6-10
6.4.1	Water Spray	6-10
6.4.2	Temperatures	6-10
APPENDIX A	Mitigation Devices - General Design Considerations	A-1
APPENDIX B	Blowdown Forces and Methods of Calculations	B-1

LIST OF FIGURES

<u>Figure Number</u>	<u>Title</u>
2-1	McGuire Nuclear Station, Key Plan
2-2	Compartment Locations, Elevations 695 and 736
2-3	Compartment Locations, Elevation 716
2-4	Compartment Locations, Elevation 733
2-5	Compartment Locations, Elevation 750
2-6	Compartment Locations, Elevation 767
3-1	Flow Diagrammatic, Steam Generator Blowdown System - BB
3-2	Flow Diagrammatic, Auxiliary Feedwater System - CA
3-3	Flow Diagrammatic, Feedwater System - CF
3-4	Flow Diagrammatic, Safety Injection System - NI
3-5	Flow Diagrammatic, Boron Thermal Regeneration System - NR
3-6	Flow Diagrammatic, Chemical & Volume Control System - NV
3-7	Flow Diagrammatic, Main Steam Supply to Auxiliary Equipment - SA
3-8	Flow Diagrammatic, Main Steam System - SM
3-10	Piping Layout, Steam Generator Blowdown System
3-11	Piping Layout, Auxiliary Feedwater System (See Figure 3-12 also)
3-12	Piping Layout, Feedwater System
3-13	Piping Layout, Safety Injection System (Upper Head Portion)
3-14	Piping Layout, Boron Thermal Regeneration System
3-15	Piping Layout, Chemical & Volume Control System
3-16	Piping Layout, Main Steam System and Main Steam to Auxiliary Equipment
4-1	Logic Symbols
4-2	Main Turbine Trip Logic Diagram
4-3	Start/Trip Logic Diagram for Main and Auxiliary Feedwater Pumps
4-4	Reactor Trip Logic Diagram
4-5	Reactor Coolant Pump Start/Trip and Main Steam Isolation Valve Logic Diagram
4-6	Safeguards Actuation, Feedwater Isolation, and Makeup Control Logic Diagram

- 6-1 Main Steam System - Isometric
- 6-2 Main Steam System - Guard Pipe/Process Pipe Details
- 6-3 Main Steam System - Pipe Rupture Restraints
- 6-4 Main Steam System - Pipe Rupture Restraints
- 6-5 Doghouse Model - Pressure Rise Analysis
- 6-6 Elevation - Peak Temperatures
- 6-7 Typical Temperature Response Curves for BB Line Break
- B-1 Pipe Rupture Reaction Thrust Force Transient
- B-2 Moody's Steady-State Separated Flow Thrust Coefficient Without Frictional Effects
- B-3 Friction Effect on Steady Blowdown Force
- B-4 Thrust Force as a Function of Stagnation Enthalpy, h_0 , and Pressure, P_0
- B-5 Subcooled Water Blowdown Thrust Force as a Function of Stagnation Enthalpy for Various Valves of Pipe Friction Parameter, f_l/D
- B-6 Water Blowdown Thrust Force as a Function of Pipe Friction Parameter, f_l/D , For Various Stagnation Enthalpies

LIST OF TABLES

<u>Table Number</u>	<u>Title</u>
3-1	Auxiliary Building, High Energy Piping Reviewed for Pipe Rupture
3-2	Turbine Building, High Energy Piping
3-3	Auxiliary Building, Moderate Energy Piping Systems
6-1	High Energy Pipe Break Locations
6-2	Unacceptable Pipe Whip/Jet Impingement Effects
6-3	Results of Damage Evaluation, Water Spray - Moderate Energy Piping
6-4	Doghouse Pressure Rise Analysis Model

FOREWORD

This report was prepared in response to Mr. A. Giambusso's December 19, 1972 letter to Duke Power Company in regard to the consequences of pipe rupture outside containment during the operation of the McGuire Nuclear Station. This report meets the intent of NRC Branch Technical Positions, MEB 3-1 and APCSB 3-1 with the exceptions identified in FSAR Table 3.6.1-3. This report has been prepared under the direction of Duke Power Company's Design Engineering Department, Mechanical and Nuclear Division.

- 3 | The first phase of this report includes the criteria and |
procedures used for evaluating the postulated failures and the results
of all evaluations for Unit 1. Subsequent revisions will include any
pertinent differences between Unit 1 and Unit 2. For resolutions
3 | to unacceptable consequences detailed in Chapter 6, alternate resolutions |
that may accomplish the same results may be used.

1.0 INTRODUCTION

This report presents the results of the evaluation of effects of postulated pipe failures outside containment for McGuire Nuclear Station. The report describes criteria and procedures used in the analysis as developed from the Nuclear Regulatory Commission Branch Technical Positions MEB 3-1 and APCS 3-1. The intent of this report is to comply with all licensing commitments as provided in FSAR Section 3.6.

Postulated rupture locations and types were determined in accordance with the criteria of Section 3.0. Systems and equipment essential to achieve plant shutdown and mitigate effects of piping failures were determined in accordance with Section 4.0. Sections 5.0 and 6.0 provide the methods used to evaluate possible damage and the results of the evaluation. All required mitigative devices are identified in Section 6.0.

2.0 PLANT DESCRIPTION

2.1 General

The McGuire Nuclear Station is located on Lake Norman in Mecklenburg County, North Carolina and consists of two 1180 MWe generating units. The Nuclear Steam Supply System for each unit is a pressurized water reactor with four coolant loops and is supplied by Westinghouse Electric Corporation. Each containment consists of a free standing cylindrical steel shield enclosed by a separate reinforced concrete Reactor Building.

Major plant structures consist of the Reactor Buildings, the Auxiliary Building, Service Building, and Turbine Buildings. The Reactor Building houses the containment vessel, reactor plant and ice condenser. The common Auxiliary Building contains the spent fuel storage facilities, diesel generators, safeguards system, waste disposal systems, reactor auxiliary systems, and control room. Non-safety shared equipment is housed in the Service Building. The Turbine Building contains the turbine generators and associated secondary system pumps and heat exchangers. These structures and their relationship are shown in Figure 2-1. Figures 2-2 through Figure 2-6 show the various compartments of each Auxiliary Building elevation. Compartment locations are given by building elevation and compartment number.

2.2 Piping Layout

Main steam and feedwater piping represent the greatest damage potential from pipe ruptures outside containment due to their large size and high operating pressure. These lines have been routed outside the Auxiliary Building, which contains the bulk of the safety related equipment. The main steam and feedwater piping penetrate the Reactor Building in the doghouses (room 750-23, -24, -25, -26, Figure 2-5) and pass over the Auxiliary Building roof into the Turbine Building. While the possibility for structural damage resulting from main steam or feedwater ruptures outside containment does exist, potential consequences are minimized by this layout.

Separation of high energy piping from essential components is the most desirable method of precluding unacceptable pipe break damage. Careful attention has been given to location of essential components and the routing of piping systems. In every possible instance, equipment susceptible to pipe break damage is located remote from the piping postulated to fail. The extensive compartmentalization in the Auxiliary Building provides separation of safety related fluid system components from redundant equipment and other high and moderate energy piping systems.

The Turbine Buildings are located directly adjacent to the Auxiliary Building as shown in Figure 2-1. This plant arrangement dictates that Turbine Building piping be routed nearby the Auxiliary Building. However, efforts have been made to separate Turbine Building high energy piping from the Auxiliary Building to the greatest extent practical. Although the potential for some damage to the Auxiliary Building wall is

possible due to Turbine Building piping failures, the interferences between a broken pipe and Turbine Building structural members and piping should prevent unacceptable consequences.

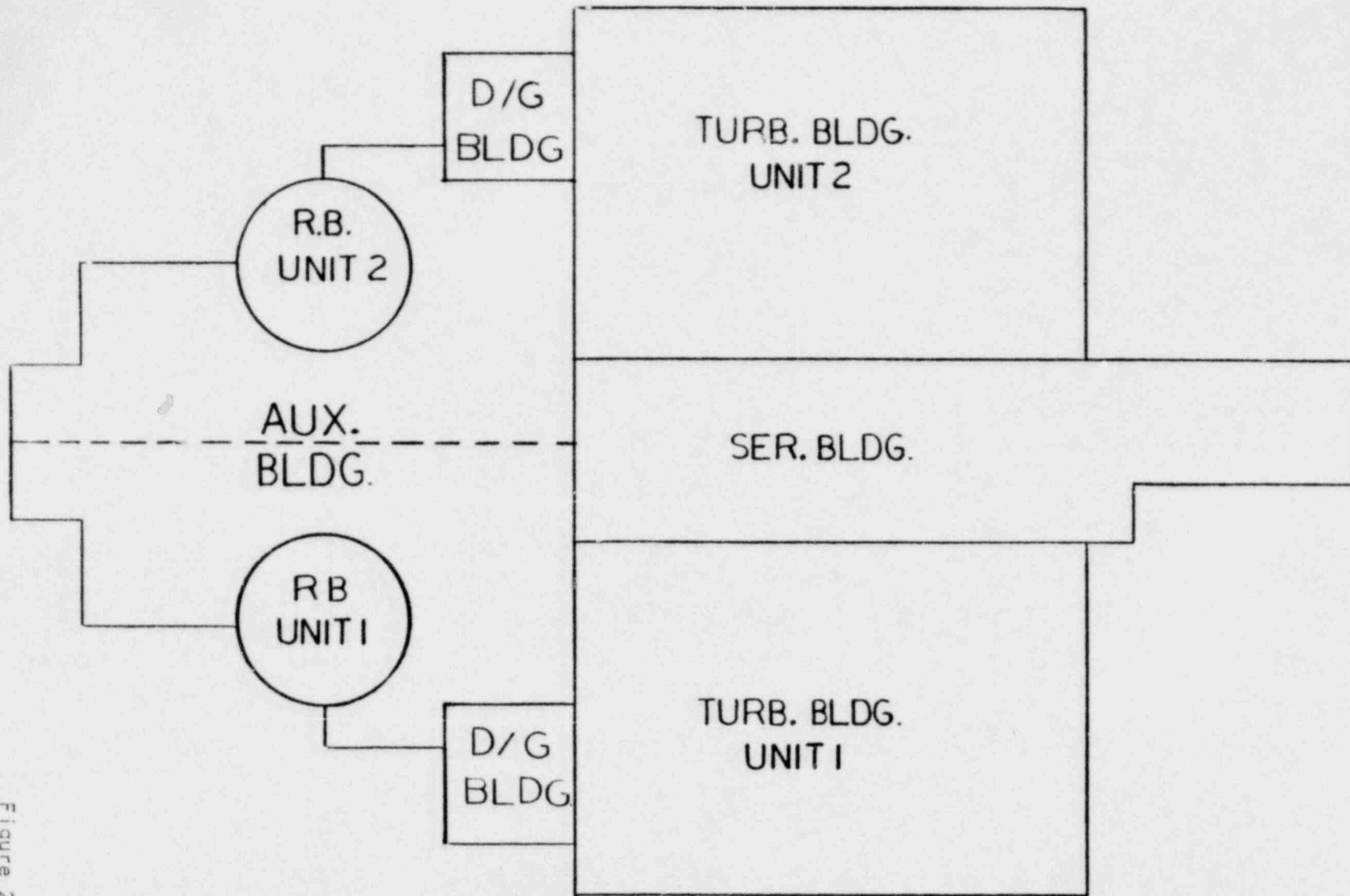
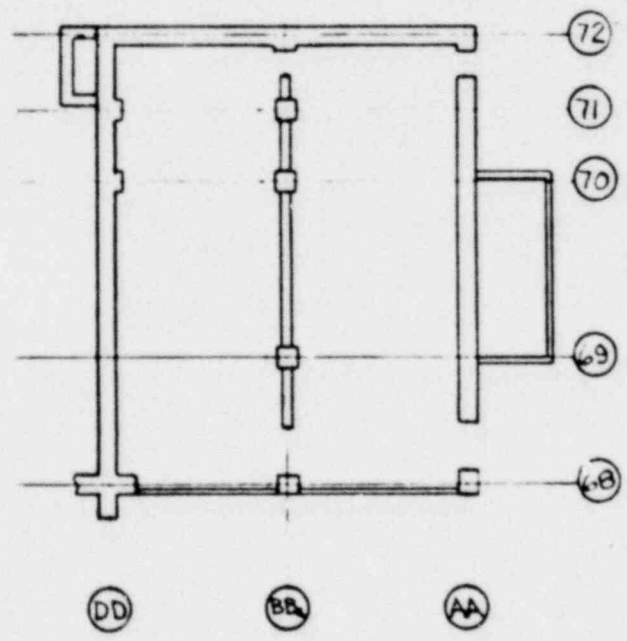
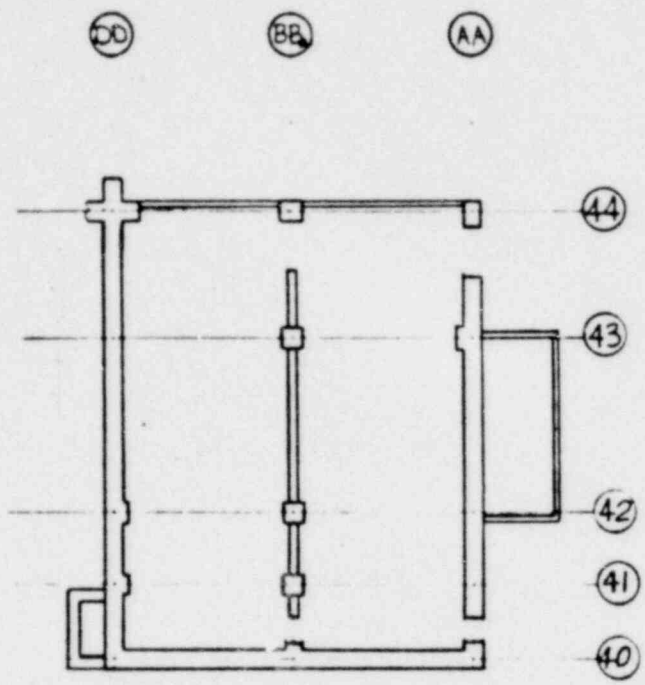


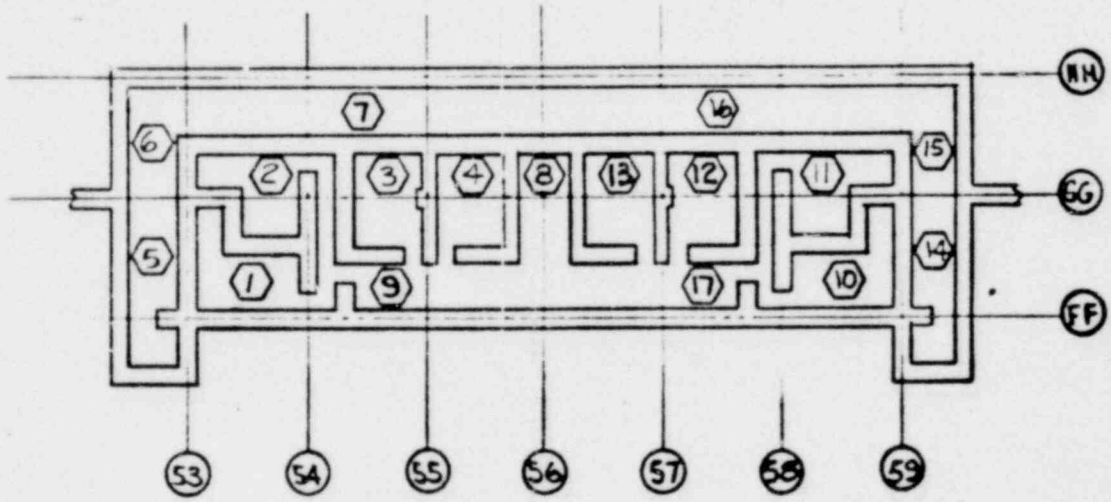
Figure 2-1

MC GUIRE NUCLEAR STATION

KEY PLAN

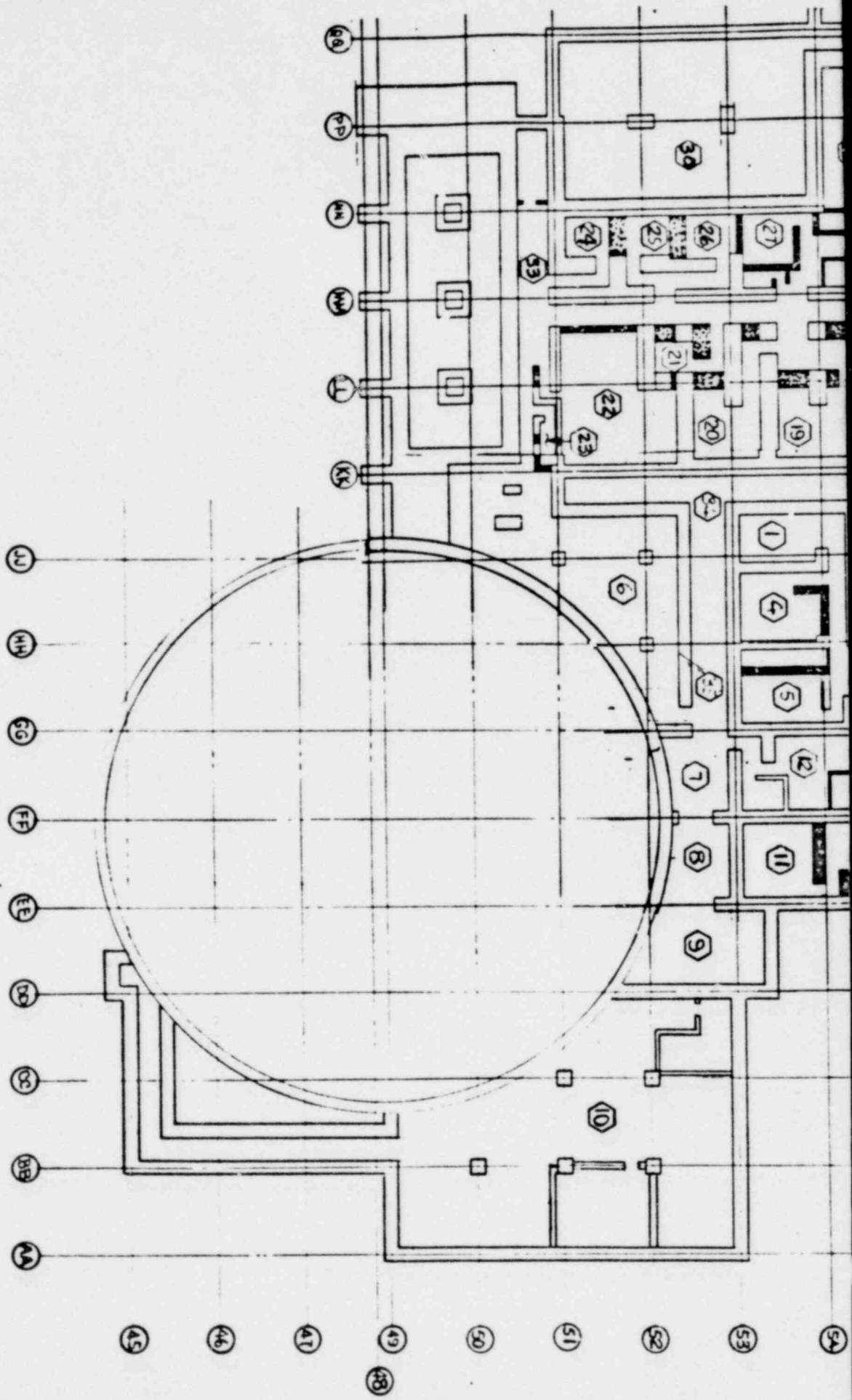


Elevation 736



Elevation 695

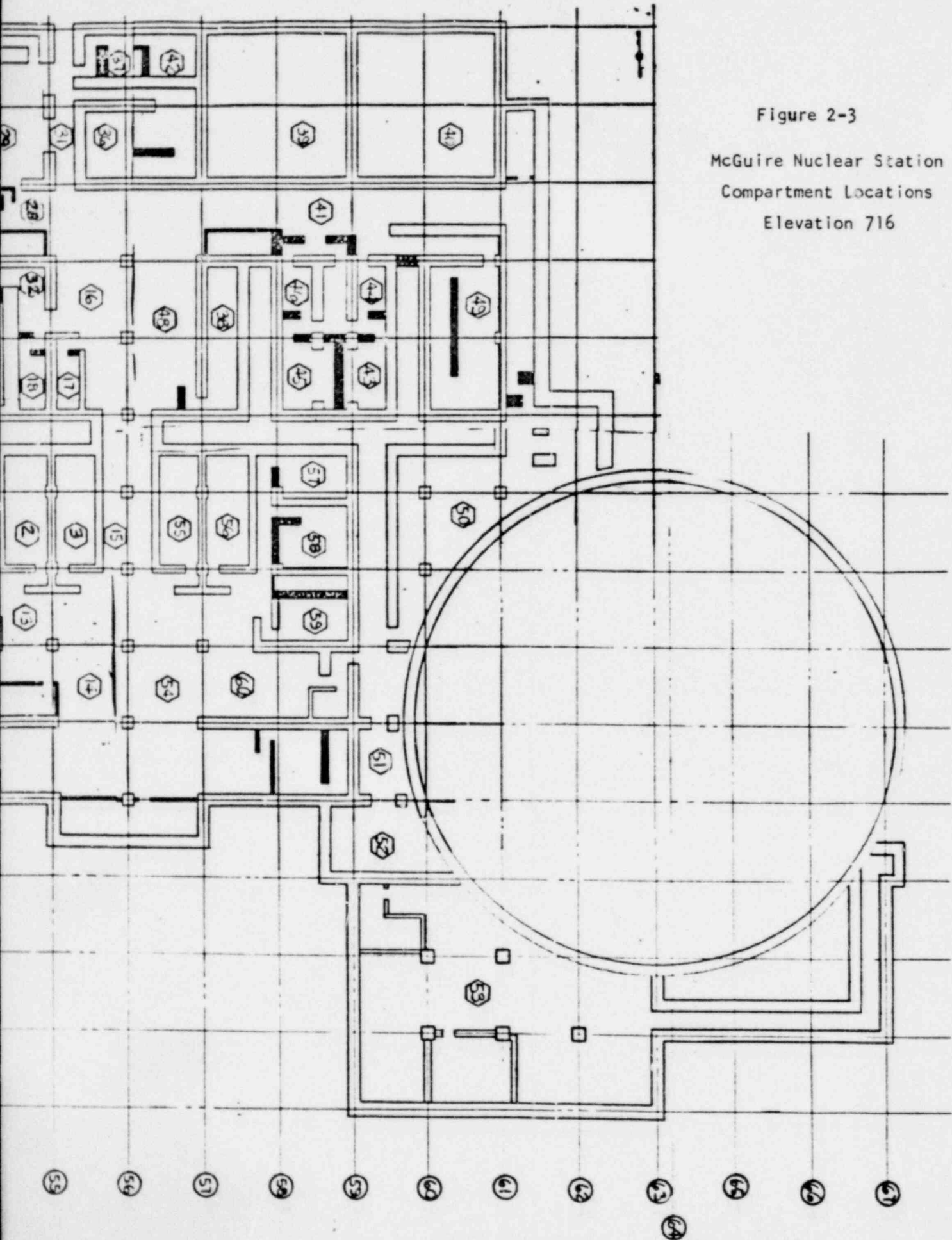
Figure 2-2
 McGuire Nuclear Station
 Compartment Locations

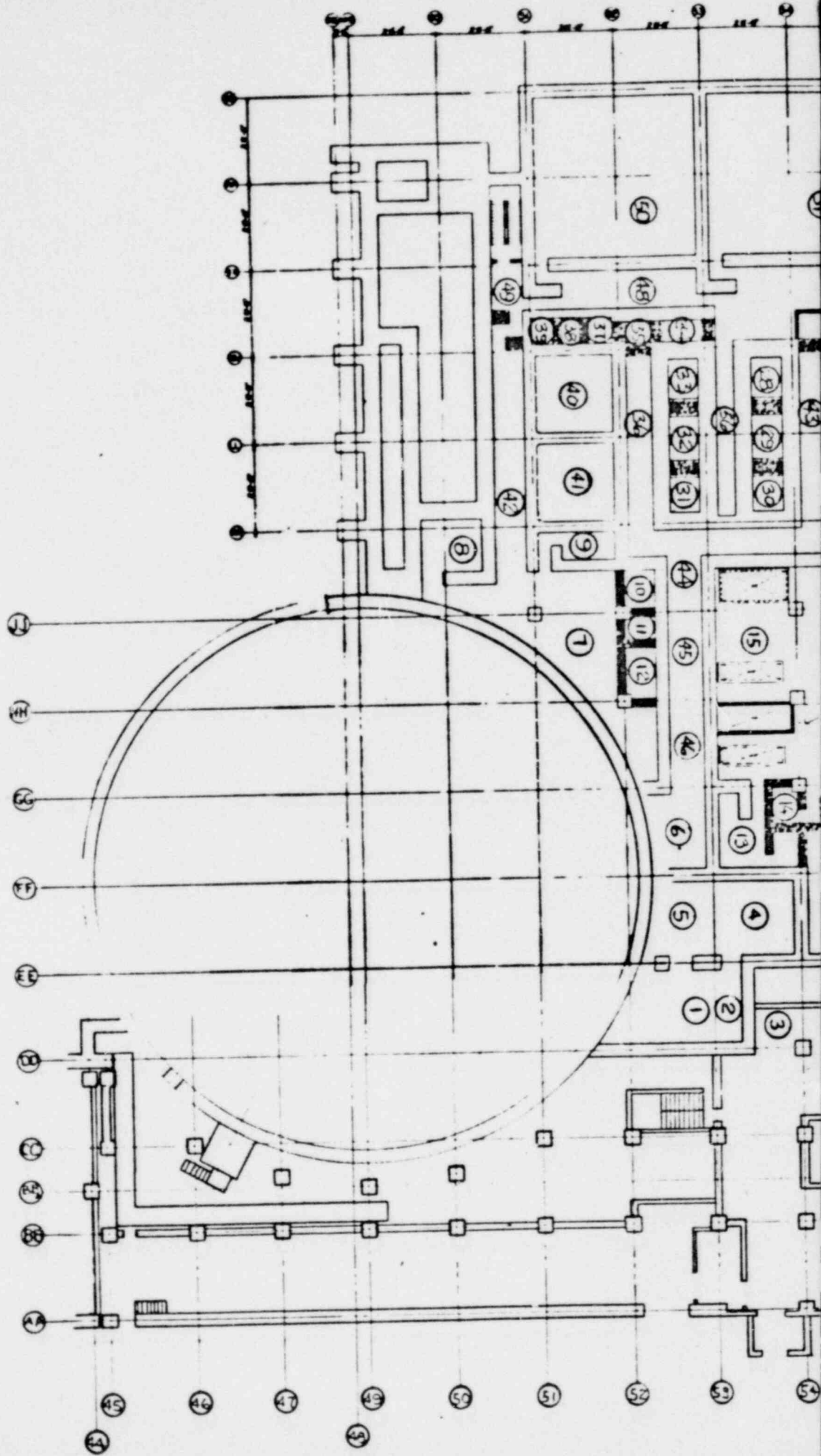


E 116-0"

Figure 2-3

McGuire Nuclear Station
Compartment Locations
Elevation 716





EC 11-4-0

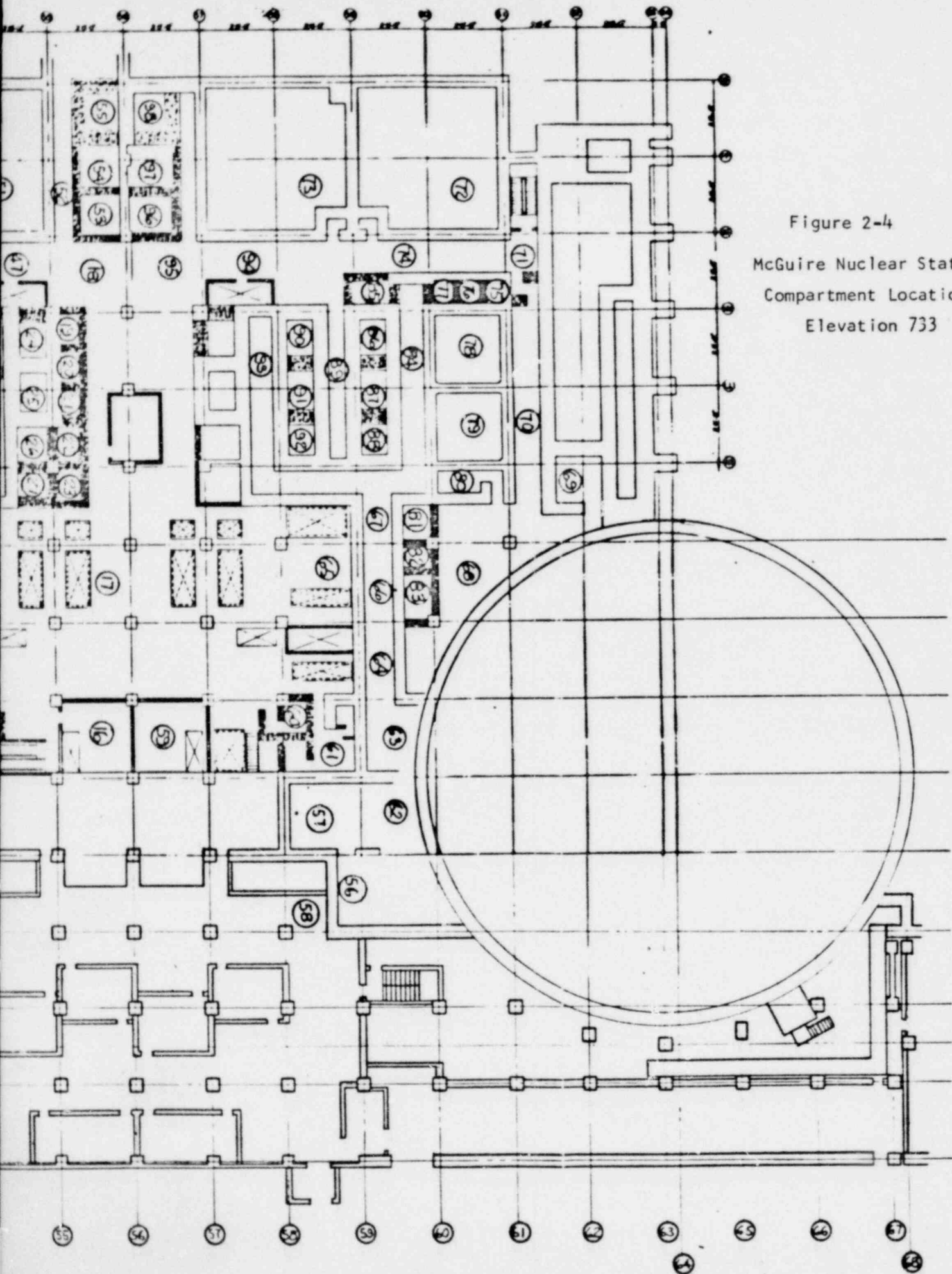
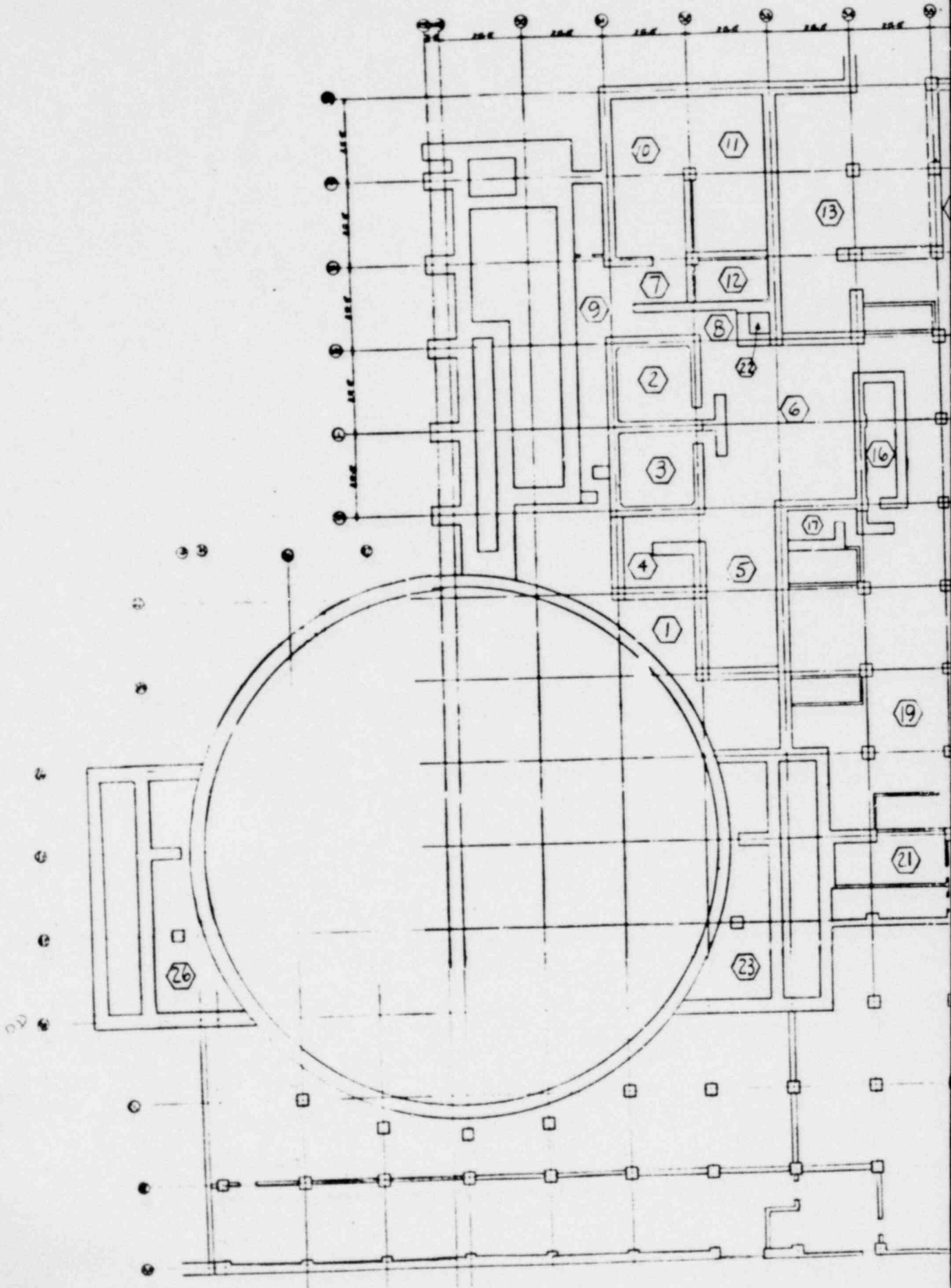


Figure 2-4
McGuire Nuclear Station
Compartment Locations
Elevation 733



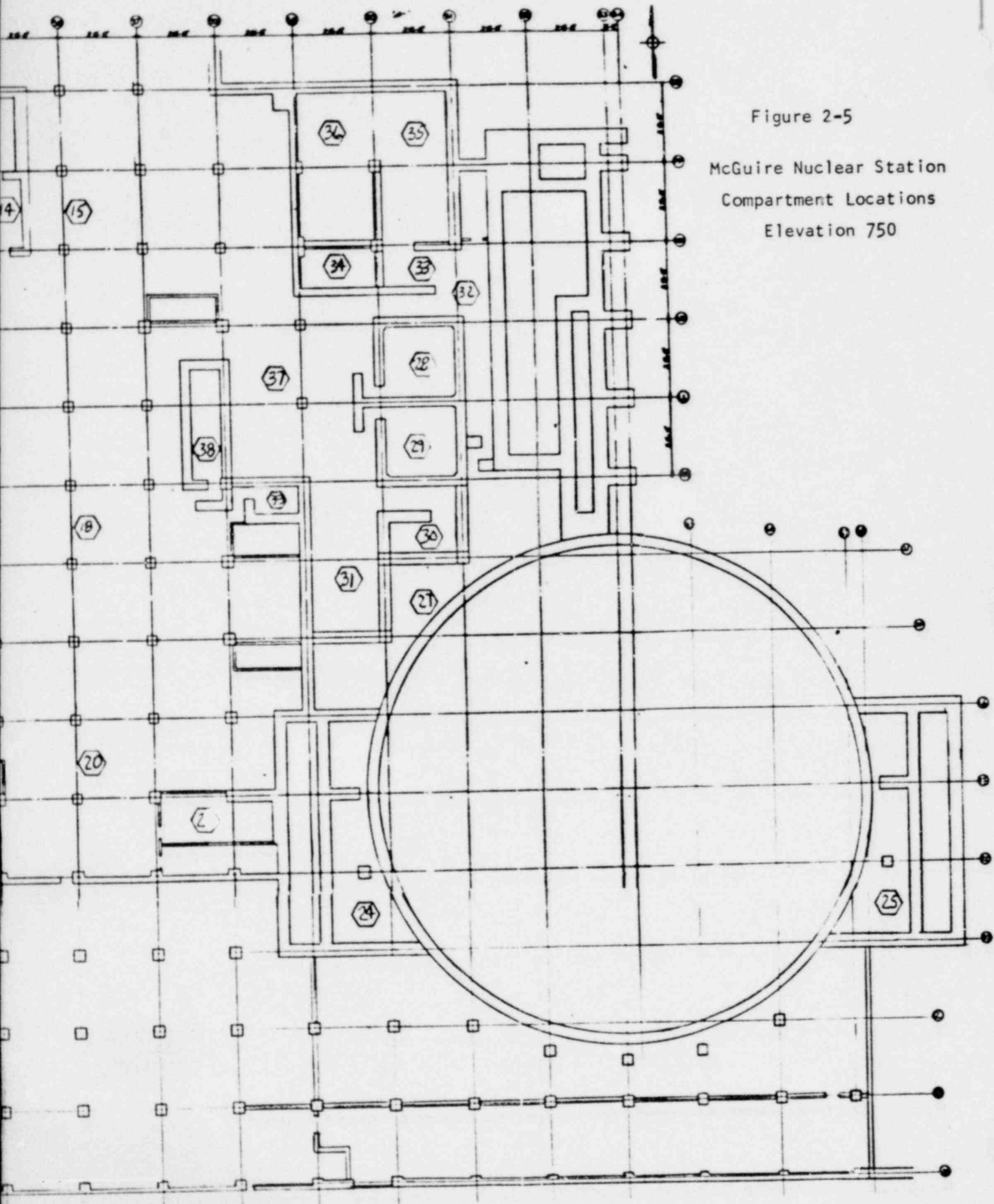


Figure 2-5
McGuire Nuclear Station
Compartment Locations
Elevation 750

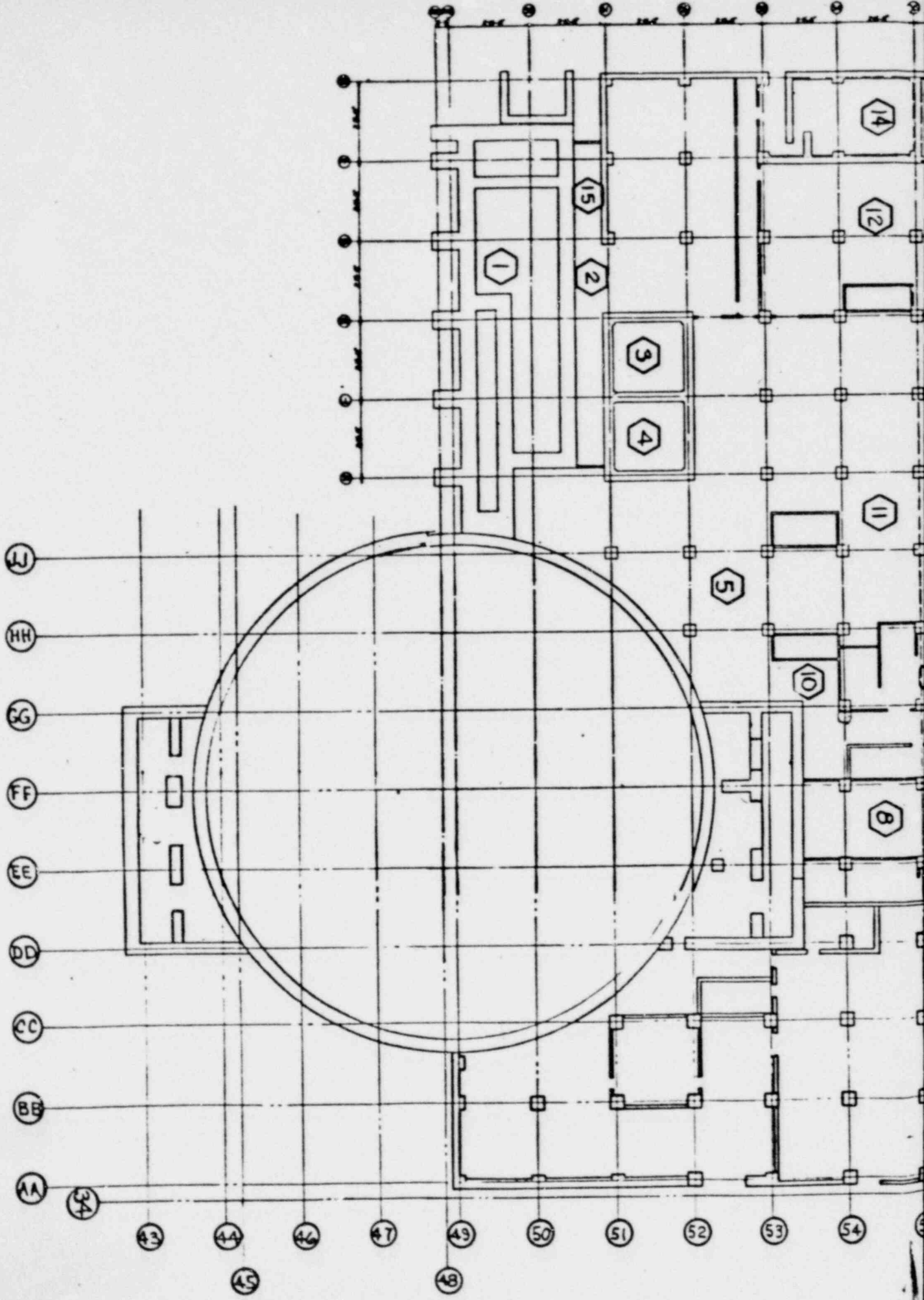
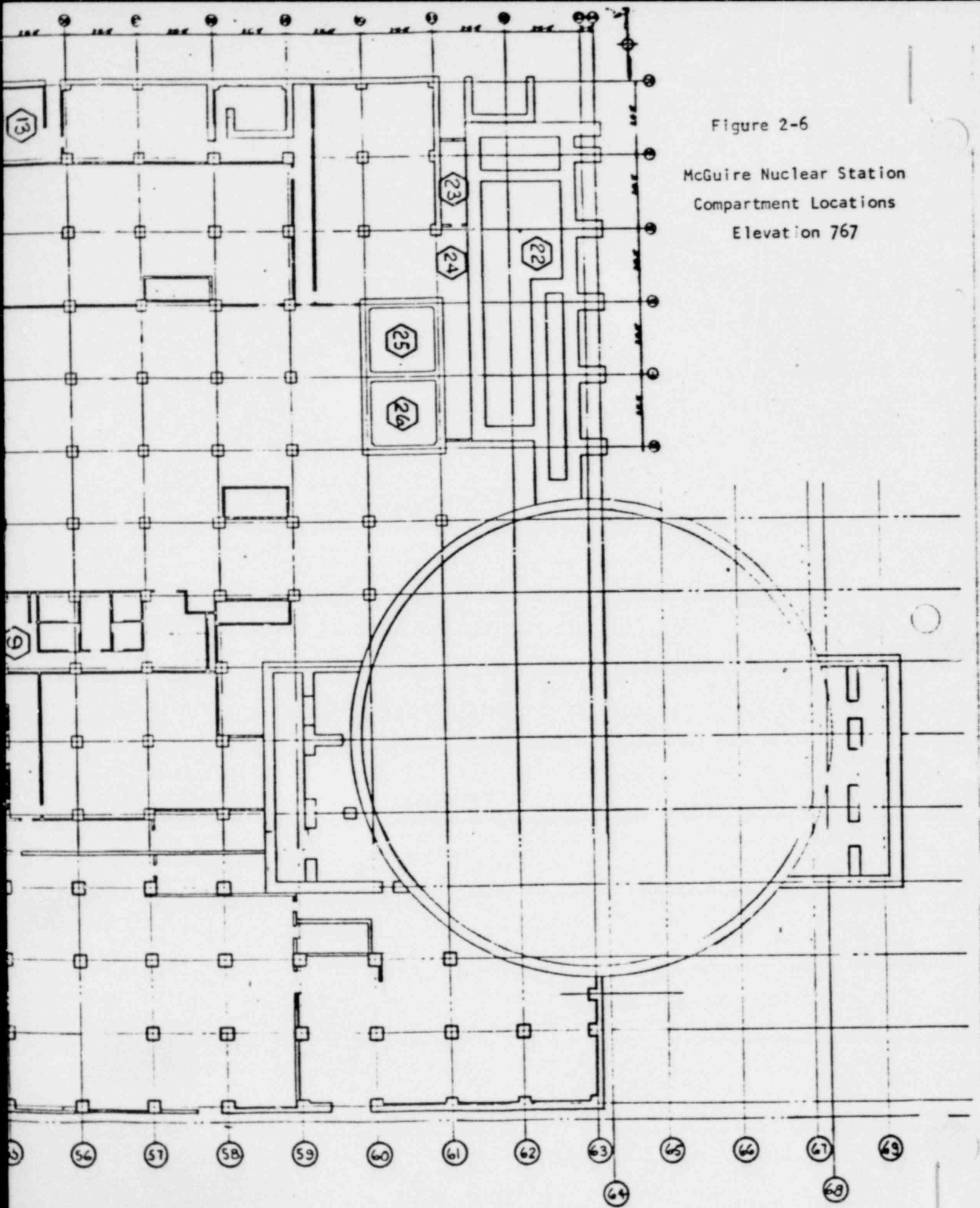


Figure 2-6
McGuire Nuclear Station
Compartment Locations
Elevation 767



3.0 CRITERIA FOR POSTULATING PIPE RUPTURES

3.1 Piping Systems Subject to Postulated Pipe Rupture

Specified ruptures are postulated to occur in plant piping systems, and the potential for damage evaluated on the basis of the level of energy stored in the system. Systems are classified as high-energy or moderate-energy.

3.1.1 High-Energy Piping Systems

High-energy piping systems are those systems, or portions of systems, that during normal plant conditions are either in operation or maintained pressurized under conditions

where either or both of the following are met:

- a) Maximum temperature exceeds 200° F, or
- b) Maximum pressure exceeds 275 psig.

Except that (1) non-liquid piping systems (air, gas, steam) with a maximum pressure less than or equal to 275 psig are not considered high energy regardless of the temperature, and (2) for liquid systems other than water, the atmospheric boiling temperature can be applied.

Systems are classified as moderate energy if the total time that either of the above conditions are met is less than either of the following:

- a) One (1) percent of the normal operating lifespan of the plant, or
- b) Two (2) percent of the time period required to accomplish its system design function.

3.1.2 Moderate-Energy Piping Systems

Moderate-Energy Piping Systems are those systems, or portions of systems, that during normal plant conditions are either in operation or maintained pressurized (above atmospheric pressure) under conditions where both of the following are met:

- a) Maximum temperature is 200° F or less (or less than the atmospheric boiling temperature for non-water systems), and
- b) Maximum pressure is 275 psig or less.

3.2 Break Location and Configuration

3.2.1 Break Locations in Duke Class B, C, and F Piping Runs

Breaks are postulated at the following locations in each Duke Class B, C, and F piping runs.

- a) The terminal ends of the pressurized portions of the run,
- b) At intermediate locations selected by either one of the following methods:
 1. At each location of potential high stress or fatigue, such as pipe fittings (elbows, tees, reducers, etc.), valves, flanges, and welded attachments, or
 2. At all locations where the stress, S , exceeds $0.8(1.2S_h + S_A)$, where
 - S = stresses under the combination of loadings associated with the normal and upset plant condition loadings, as calculated from the sum of equations (9) and (10) in Subarticle NC-3600 of the ASME Boiler and Pressure Vessel Code,

S_h = basic material allowable stress at maximum (hot) temperature from the allowable stress tables in Appendix I of the ASME Boiler and Pressure Vessel Code, Section III.

S_A = allowable stress range for expansion stresses, as defined in Subarticle NC-3600 of the ASME Boiler and Pressure Vessel Code, Section III.

- c) If there are not at least two intermediate locations where S exceeds $0.8 (1.2S_h + S_A)$, a minimum of two separated locations are chosen based upon highest stress. If the piping run has only one change of direction, a minimum of one intermediate break is postulated. The pattern of postulated intermediate break locations shall be determined for the normal plant condition load combination and upset plant condition which has the highest stress. Intermediate breaks are not postulated in sections of straight pipe where there are no pipe fittings, valves, flanges, nor welded attachments, nor in piping runs where S is less than $0.4 (1.2S_h + S_A)$ over the length of the run.

3.2.2 Break Locations In Other Piping Runs

Breaks are postulated to occur at the following locations in each Duke Class E, G, and H piping run.

- a) At each location of potential high stress or fatigue, such as pipe fittings (elbows, tees, reducers, etc.), valves, flanges, and welded attachments.

- b) The terminal ends of the pressurized portions of the run.

3.2.3 Break Configuration

3.2.3.1 High-Energy Piping

The following types of breaks are postulated in high-energy piping systems:

- a) No breaks are postulated in piping having a nominal diameter less than or equal to one (1) inch.
- b) Circumferential breaks are postulated in piping having a nominal diameter between one (1) and four (4) inches.
- c) Longitudinal and circumferential breaks are postulated, but not concurrently, in piping having a nominal diameter equal to or greater than four (4) inches. At terminal ends where piping has no longitudinal welds, no longitudinal breaks are postulated. Also, at intermediate locations where the criterion for a minimum number of break locations must be satisfied, only circumferential breaks are postulated.

Where break locations are postulated at fittings without the benefit of a detailed stress calculation, breaks are assumed to occur at each pipe-to-fitting weld. If detailed stress analyses or tests are performed, the maximum stressed location in the fitting may be selected as the break location. Where two or more breaks are postulated, break locations are

selected with at least 10% difference in stress, or, if stresses differ by less than 10%, locations are selected so as to be separated by a change of direction of the pipe run. A circumferential break results in pipe severance with full separation, except as limited by structural design features. The break is assumed perpendicular to the longitudinal axis of the pipe, and the break area assumed to be the cross-sectional flow area of the pipe at the break location. The break discharge coefficient used may be substantiated analytically or experimentally. In the absence of this data, the discharge coefficient is assumed to be 1.0.

A longitudinal break results in an axial split without severance. The split is assumed to be oriented at any point about the circumference of the pipe, or alternately at the point(s) of highest stress, as justified by detailed stress analyses. For the purpose of design, the longitudinal break is assumed to be circular or elliptical ($2D \times \frac{1}{2}D$) with an area equal to the largest piping cross-sectional flow area at the point of the break, a length equal to twice the piping internal diameter at that cross-section, and have a discharge coefficient of 1.0. Any other values used for the area, diameter, or discharge coefficient associated with a longitudinal break are verified by test data or analysis which define the limiting break geometry.

For purposes of analysis, circumferential and longitudinal breaks are assumed to reach full size within one (1)

millisecond after break initiation.

3.2.3.2 Moderate-Energy Piping

Crack openings are assumed as a circular orifice of cross-sectional flow area equal to that of a rectangle with a length of one-half the pipe inside diameter and a width of one-half the pipe wall thickness. The orifice is assumed to be oriented at any point about the circumference of the pipe.

Cracks are not postulated in piping that contains no pressurization equipment; i.e., systems without pumps, pressurizing tanks, boilers, etc., and which operate only from gravity flow or storage tank head. Also, cracks are not postulated in portions of Class B, C, and F piping where the stresses are less than $0.4 (1.2S_h + S_A)$.

3.3 High Energy Piping Outside Containment

High energy piping outside containment is reviewed by plant area. Auxiliary Building high energy piping is identified in Table 3-1. This table also lists the figure for the applicable flow diagrammatic. All high energy piping is shown by system on figures 3.10 through 3.16. Table 3-2 identifies all Turbine Building piping that has been reviewed for pipe rupture.

3.4 Moderate Energy Piping Outside Containment

Only Auxiliary Building moderate energy piping is reviewed in detail. Turbine Building moderate energy piping is incapable of causing any unacceptable damage due to its location away from essential components. Auxiliary Building moderate energy piping systems are identified in Table 3-3.

McGUIRE NUCLEAR STATION
AUXILIARY BUILDING
HIGH ENERGY PIPING REVIEWED FOR PIPE RUPTURE

1. Steam Generator Blowdown System - BB (See Figure 3.1)
From: Containment Penetrations M-300, M-301, M-303, and M-304;
To: 1. The outlet of valve 1BB101;
2. The service bldg. side of the trench.

2. Auxiliary Feedwater System - CA (See Figure 3.2)
From: Motor Driven Aux. FWD Pumps 1A and 1B;
To: 1. Containment Penetrations M-156, M-286, M-3100 and M-465 up to and including check valves ICA37, ICA49, ICA53 and ICA65;
2. The outlet of valves ICA70 and ICA72.

3. Feedwater System - CF (See Figure 3.3)
From: The Turbine Bldg. header;
To: Containment Penetrations M-153, M-262, M-308, and M-440.

4. Safety Injection System - NI (See Figure 3.4)
From: Upper Head Injection Nitrogen Accumulator;
To: Upper Head injection Water Accumulator.

From: Upper Head Injection Water Accumulator;
To: 1. Containment Penetrations M-334 and M-349;
2. Four 4" Stand Pipes.
3. Valves INI326, INI255B

5. Boron Thermal Regeneration - NR (See Figure 3.5)
From: Valve INR92 (See NV System);
To: Letdown Reheat Hx.

From: Letdown Reheat Hx;
To: Valve INR95 (See NV System).

5. Chemical & Volume Control System - NV (See Figure 3.6)

- 3
- From: 1. Reciprocating charging pump
 2. Letdown Hx
 3. Sealwater Injection Filters 1A and 1B
- To: 1A. Penetrations M-350, M-339, M-344, M-343 and M-329;
 1B. Sealwater Injection Filters 1A and 1B;
 2A. Penetration M-347;
 2B. Valves INR92 and INR95;
 2C. Valves INV121 and INV124;
 3A. Valves INV225, INV231 and BIT outlets
 3B. Sealwater Injection Filters A & B.

7. Main Steam Supply to Auxiliary Equipment - SA (See Figure 3.7)

- From: Main Steam lines 1B and 1C;
 To: The outlet of valves ISA48 and ISA49.

8. Main Steam System - SM (See Figure 3.8)

- 3
- From: Containment Penetrations M-154, M-261, M-393, and M-441;
 To: 1. Turbine Bldg. 48" header;
 2. The outlet of the safety valves and power operated relief valves.
 3. The outlet of low point drain valves.

Note: Piping less than or equal to 1" NPS need not be considered for effects of pipe rupture.

McGUIRE NUCLEAR STATION
TURBINE BUILDING
HIGH ENERGY PIPING

1. Auxiliary Steam - AS

From: Valve 1SM13;
To: Valves 1AS11, 1AS88.

3 |

2. Steam Generator Blowdown - BB

From: Service Bldg. trench;
To: Valves 1BB123, 1BB124, 1BB125, 1BB126.

From: Steam Generator Blowdown Blowoff Tank;
To: Steam Generator Blowdown Blowoff Pumps 1A and 1B.

From: Steam Generator Blowdown Blowoff Tank Pumps 1A and 1B;
To: Condensate System.

3. Heating Boiler Feedwater - CB

From: Aux. Electric Boiler A and B Feedwater Pumps;
To: Aux. Electric Boiler A and B.

4. Feedwater - CF

From: Main FDW Pumps 1A and 1B;
To: 1. HP Heaters 1B1, 1B2, 1B3;
2. Valves 1CF75, 1CF81, 1CF76.

From: HP Heaters 1B1, 1B2, 1B3;
To: HP Heaters 1A1, 1A2, 1A3 respectively.

From: 1. HP Heaters 1A1, 1A2, 1A3;
2. Valve 1CF75;
To: 1. Auxiliary Building Roof;
2. Valves 1CF124, 1CF107.

5. FDWP Condensate Seal - CL

From: Valves 1CL1, 1CL4;
To: FDWP Seal Injection Pumps 1A and 1B.

From: FDWP Seal Injection Pumps 1A and 1B;
To: FDWP Seal Injection Hx 1A and 1B.

From: FDWP Seal Injection Hx 1A and 1B;
To: Main FDW Pump 1A and 1B.

6. Condensate - CM
 From: 1. LP Heaters 1F1, 1F2, 1F3;
 2. Valves 1CM130, 1CM420;
 To: Condensate Booster Pumps 1A, 1B, 1C.
- From: Condensate Booster Pumps 1A, 1B, 1C;
 To: 1. LP Heaters 1E1, 1E2, 1E3;
 2. Valves 1CM230, 1CL1, 1CL4.
- From: LP Heaters 1E1, 1E2, 1E3;
 To: LP Heaters 1D1, 1D2, 1D3 respectively.
- From: 1. LP Heaters 1D1, 1D2, 1D3;
 2. Valve 1CM230;
 To: 1. LP Heaters 1C1, 1C2, 1C3;
 2. Valve 1CM231.
- From: 1. LP Heaters 1C1, 1C2, 1C3;
 2. Valves 1CM231, 1HW82, 1HW83, 1HW84; 1CF124;
 To: 1. UST Dome;
 2. Main Feedwater Pumps 1A and 1B.
7. "A" Heater Bleed Steam - HA
 From: 1. HP Turbine;
 2. Valves 1HM19, 1HM20, 1HM21, 1HM22, 1HM23, 1HM24;
 To: 1. HP Heaters 1A1, 1A2, 1A3;
 2. MSR's 1st Stage.
8. "B" Heater Bleed Steam - HB
 From: 1. HP Turbine;
 2. Valves 1HM34, 1HM35, 1HM36, 1HM37, 1HM38, 1HM39;
 To: HP Heaters 1B1, 1B2, 1B3.
- 3| 9. Moisture - Separator - Reheater Bleed Steam - HM
 From: Valves 1HM1, 1HM2, 1HM3, 1HM4, 1HM5, 1HM6, 1HM13, 1HM14, 1HM15,
 1HM16, 1HM17, 1HM18;
 To: MSR's 2nd Stage.
- From: MSR's 2nd Stage;
 To: Valves 1HM19, 1HM20, 1HM21, 1HM22, 1HM23, 1HM24.
- From: MSR's 1st Stage;
 To: Valves 1HM34, 1HM35, 1HM36, 1HM37, 1HM38, 1HM39.

10. Moisture - Separator - Reheater Drain - HS

From: MSR 1A1, 1A2, 1B1, 1B2, 1C1, 1C2;

- To: 1. 1st Stage Reheater Drain Tank 1A1, 1A2, 1B1, 1B2, 1C1, 1C2 respectively;
2. 2nd Stage Reheater Drain Tank 1A1, 1A2, 1B1, 1B2, 1C1, 1C2 respectively;
3. Moisture Separator Drain Tank 1A1, 1A2, 1B1, 1B2, 1C1, 1C2 respectively.

From: 1st Stage Reheater Drain Tank 1A1, 1A2, 1B1, 1B2, 1C1, 1C2;

- To: 1. MSR, 1st Stage, 1A1, 1A2, 1B1, 1B2, 1C1, 1C2 respectively;
2. Valves 1HS35, 1HS36, 1HS93, 1HS94, 1HS151, 1HS152 respectively.

From: 2nd Stage Reheater Drain Tank 1A1, 1A2, 1B1, 1B2, 1C1, 1C2;

- To: 1. MSR, 2nd Stage, 1A1, 1A2, 1B1, 1B2, 1C1, 1C2 respectively.
2. Valves 1HS57, 1HS58, 1HS115, 1HS116, 1HS173, 1HS174 respectively.

From: 1st Stage Reheater Drain Tank 1A1, 1A2, 1B1, 1B2, 1C1, 1C2;

- To: 1. HP Heaters 1B1, 1B2, 1B3 respectively;
2. Valves 1HS31, 1HS32, 1HS89, 1HS90, 1HS147, 1HS148 respectively.

From: 2nd Stage Reheater Drain Tank 1A1, 1A2, 1B1, 1B2, 1C1, 1C2;

- To: 1. HP Heater 1A1, 1A2, 1A3 respectively;
2. Valves 1HS53, 1HS54, 1HS111, 1HS112, 1HS169, 1HS170, respectively.

From: Moisture Separator Drain Tanks 1A1, 1A2, 1B1, 1B2, 1C1, 1C2;

- To: 1. Heater Drain Tanks 1C1, 1C2, 1C3 respectively;
2. Valves 1HS13, 1HS14, 1HS71, 1HS72, 1HS129, 1HS130 respectively.

11. Heater Drain - HW

From: HP Heaters 1A1, 1A2, 1A3;

- To: 1. HP Heaters 1B1, 1B2, 1B3 respectively;
2. Valves 1HW22, 1HW23, 1HW24 respectively.

From: HP Heaters 1B1, 1B2, 1B3;

- To: 1. Heater Drain Tanks 1C1, 1C2, 1C3 respectively;
2. Valves 1HW37, 1HW38, 1HW39 respectively.

From: Heater Drain Tank 1C1, 1C2, 1C3;

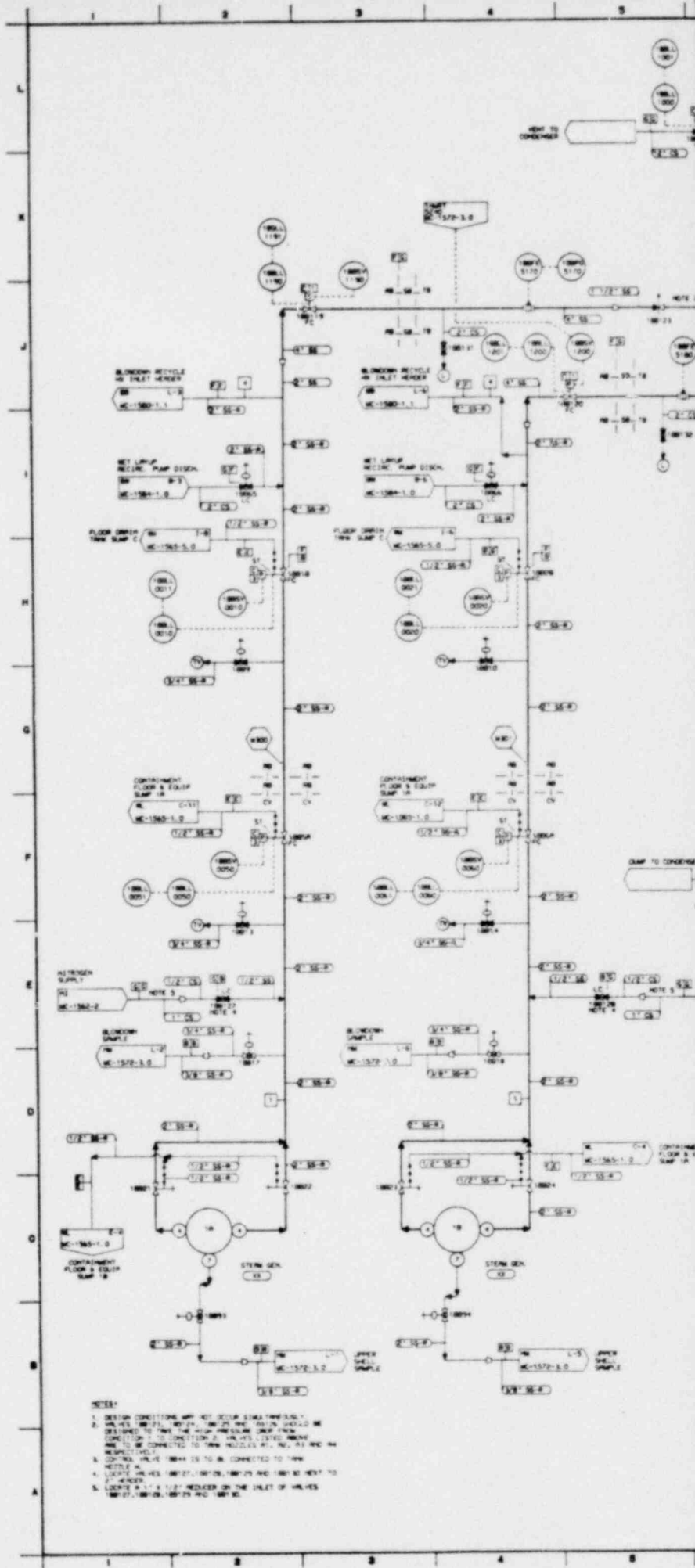
- To: 1. Heater Drain Tank Pumps 1C1, 1C2, 1C3 respectively;
2. Valves 1HW43, 1HW44, 1HW45 respectively.

- From: Heater Drain Tank Pumps 1C1, 1C2, 1C3;
 To: 1. Heater Drain Tank 1C1, 1C2, 1C3 respectively;
 2. LP Heaters 1C1, 1C2, 1C3 respectively;
 3. Valves 1HW85, 1HW86, 1HW87 respectively;
 4. Valves 1HW82, 1HW83, 1HW84 respectively.
- From: LP Heaters 1D1, 1D2, 1D3;
 To: 1. LP Heaters 1E1, 1E2, 1E3 respectively;
 2. Valves 1HW106, 1HW107, 1HW108 respectively;
- From: LP Heaters 1E1, 1E2, 1E3;
 To: 1. LP Heaters 1F1, 1F2, 1F3 respectively
 2. Valves 1HW121, 1HW122, 1HW123 respectively.
12. Main Steam Bypass to Condenser - SB
 From: Valve 1SB1;
 To: Valves 1SB3, 1SB6, 1SB9, 1SB12, 1SB15, 1SB18, 1SB21, 1SB24,
 1SB27.
13. Main Steam - SM
 From: Auxiliary Building Roof;
 To: 1. HP Turbine
 2. Valves 1SV30, 1SV32, 1SV34, 1SV36; 1SV38, 1SV40, 1SV42, 1SV44,
 1SM2, 1SP1, 1SP2, 1ZJ105, 1HM1, 1HM2, 1HM3, 1HM13, 1HM14,
 1HM15, 1HM4, 1HM5, 1HM6, 1HM16, 1HM17, 1HM18, 1SM12, 1SB1.
14. Main Steam Supply to FDWP Turbine - SP
 From: Valves 1SP1, 1SP2;
 To: FDW Pump Turbines 1A and 1B.
15. Main Turbine Leakoff & Steam Seal - TL
 From: Valves 1TL21, 1SM2;
 To: Valves 1TL6 and 1TL8.
16. Plant Heating - YH
17. Condenser Steam Air Ejector - ZJ
 From: Valves 1ZJ105, 1ZJ106;
 To: Valves 1ZJ2, 1ZJ4, 1ZJ96.

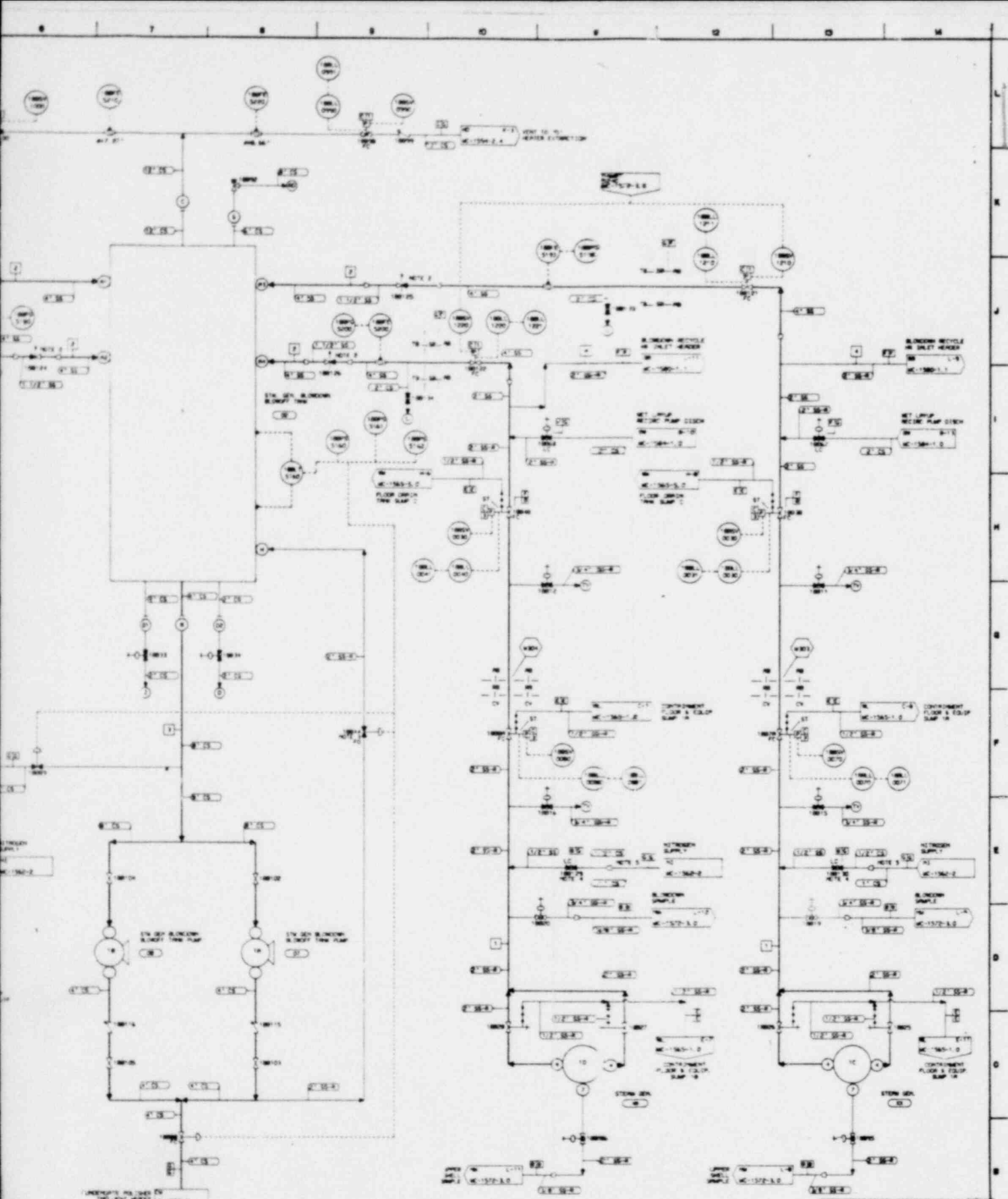
MCGUIRE NUCLEAR STATION
AUXILIARY BUILDING
MODERATE ENERGY PIPING SYSTEMS

1. Auxiliary Steam - AS
2. Steam Generator Wet Layup Recirculation - BW
3. Diesel Generator Engine Fuel Oil - FD
4. Refueling Water - FW
5. Component Cooling - KC
6. Diesel Generator Engine Cooling Water - KD
7. Spent Fuel Cooling - KF
8. Recirculated Cooling Water - KR
9. Diesel Generator Engine Lubricating Oil - LD
10. Boron Recycle - NB
11. Residual Heat Removal - ND
12. Ice Condenser Refrigeration - NF
13. Nuclear Sampling - NM
14. Containment Spray - NS
15. Fire Protection - RF
16. Nuclear Service Water - RN
17. Containment Ventilation Cooling Water - RV
18. Breathing Air - VB
19. Diesel Generator Engine Starting Air - VG
20. Diesel Generator Engine Air Intake & Exhaust - VN
21. Station Air - VS
22. Equipment Decontamination - WE
23. Waste Gas - WG
24. Liquid Waste Recycle - WL

25. Liquid Waste Monitor and Disposal - WM
26. Diesel Generator Room Sump Pump - WN
27. Nuclear Solid Waste Disposal - WS
28. Groundwater Drainage - WZ
29. Chilled Water - YC
30. Plant Heating - YH
31. Makeup Demineralized Water - YM
32. Diesel Generator Engine Crankcase Vacuum - ZD



- NOTES:
1. DESIGN CONDITIONS APPLY TO THE STEAM GENERATOR.
 2. VALVES 18027, 18028, 18029 AND 18030 SHOULD BE DESIGNED TO PASS THE HIGH PRESSURE STEAM FROM THE STEAM GENERATOR TO THE CONDENSER. AN OVER-LOADING WARNING LINE TO BE CONNECTED TO THESE VALVES AT 18027 AND 18030.
 3. CONTROL VALVE 18044 IS TO BE CONNECTED TO THE STEAM GENERATOR.
 4. LOCATE VALVES 18027, 18028, 18029 AND 18030 NEXT TO THE STEAM GENERATOR.
 5. LOCATE VALVES 18027, 18028, 18029 AND 18030 NEXT TO THE STEAM GENERATOR.



NORMAL OPERATING CONDITIONS			
CONDENSER INLET	100.0	100.0	100.0
CONDENSER OUTLET	100.0	100.0	100.0
STEAM GENERATOR INLET	100.0	100.0	100.0
STEAM GENERATOR OUTLET	100.0	100.0	100.0

DESIGN CONDITIONS			
CONDENSER INLET	100.0	100.0	100.0
CONDENSER OUTLET	100.0	100.0	100.0
STEAM GENERATOR INLET	100.0	100.0	100.0
STEAM GENERATOR OUTLET	100.0	100.0	100.0

NO.	REVISION	DATE	BY	CHKD.	DESCRIPTION
1	REV. 1	10/1/68	J. H. ...	J. H.
2	REV. 2	10/1/68	J. H. ...	J. H.
3	REV. 3	10/1/68	J. H. ...	J. H.
4	REV. 4	10/1/68	J. H. ...	J. H.
5	REV. 5	10/1/68	J. H. ...	J. H.

NUCLEAR SAFETY RELATED

DUKE POWER COMPANY

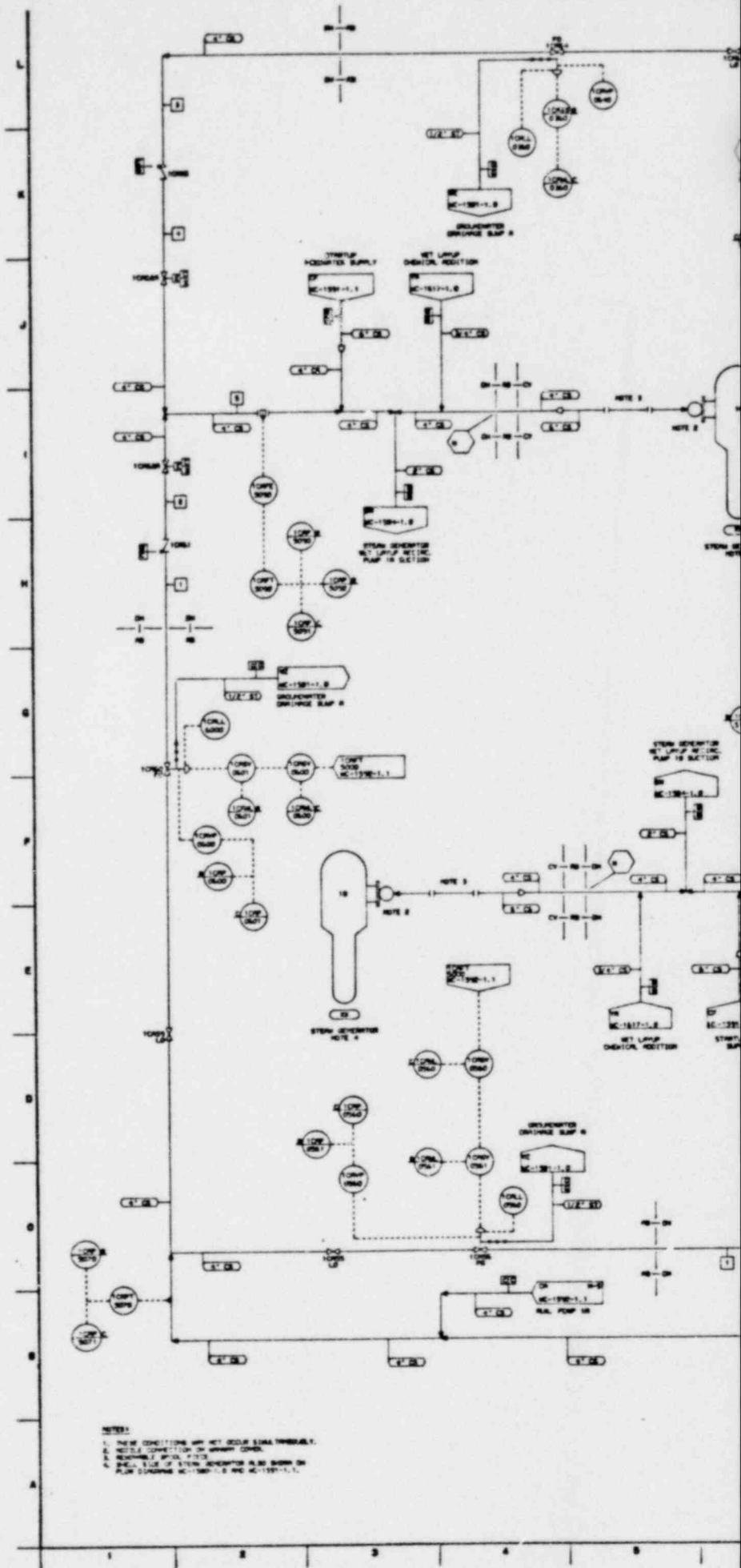
MOORE HULLER STATION UNIT 1

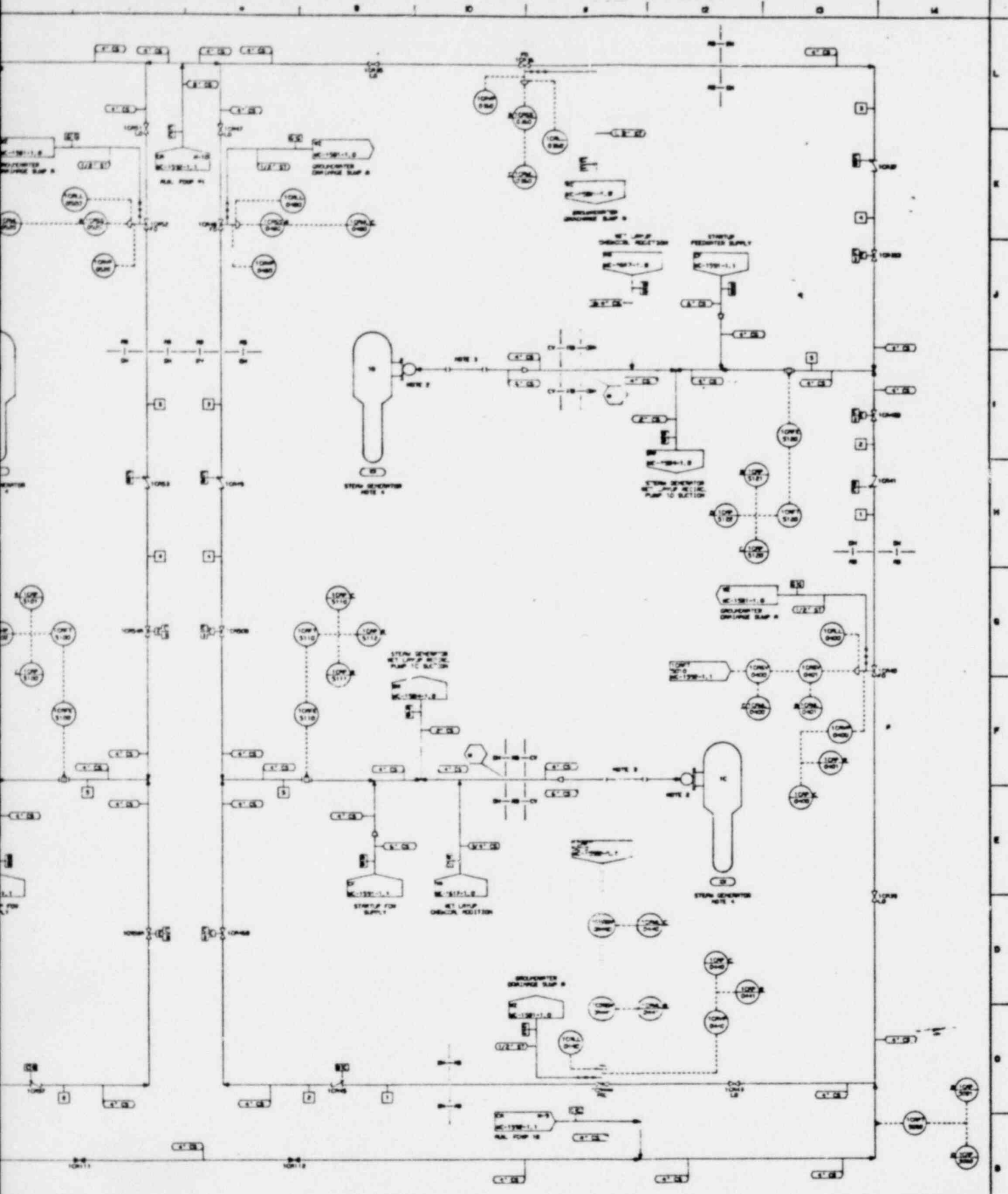
FLOW DIAGRAM OF
STEAM GENERATOR BLEED
CYCLE SYSTEM (S)

SCALE: 1/4" = 1'-0" (VERTICAL DIMENSIONS)
SCALE: 1/4" = 1'-0" (HORIZONTAL DIMENSIONS)

DRAWN BY: J. H. ...
CHECKED BY: J. H. ...
DATE: 10/1/68

FIGURE 3.1
SHEET 1 of 1





NORMAL OPERATING CONDITIONS	
TEMPERATURE	110-115
PRESSURE	22-23
FLOW	110-115
LEVEL	110-115

DESIGN CONDITIONS (NOTE 1)	
TEMPERATURE	110-115
PRESSURE	22-23
FLOW	110-115
LEVEL	110-115

FIGURE 3.2
SHEET 1 of 2

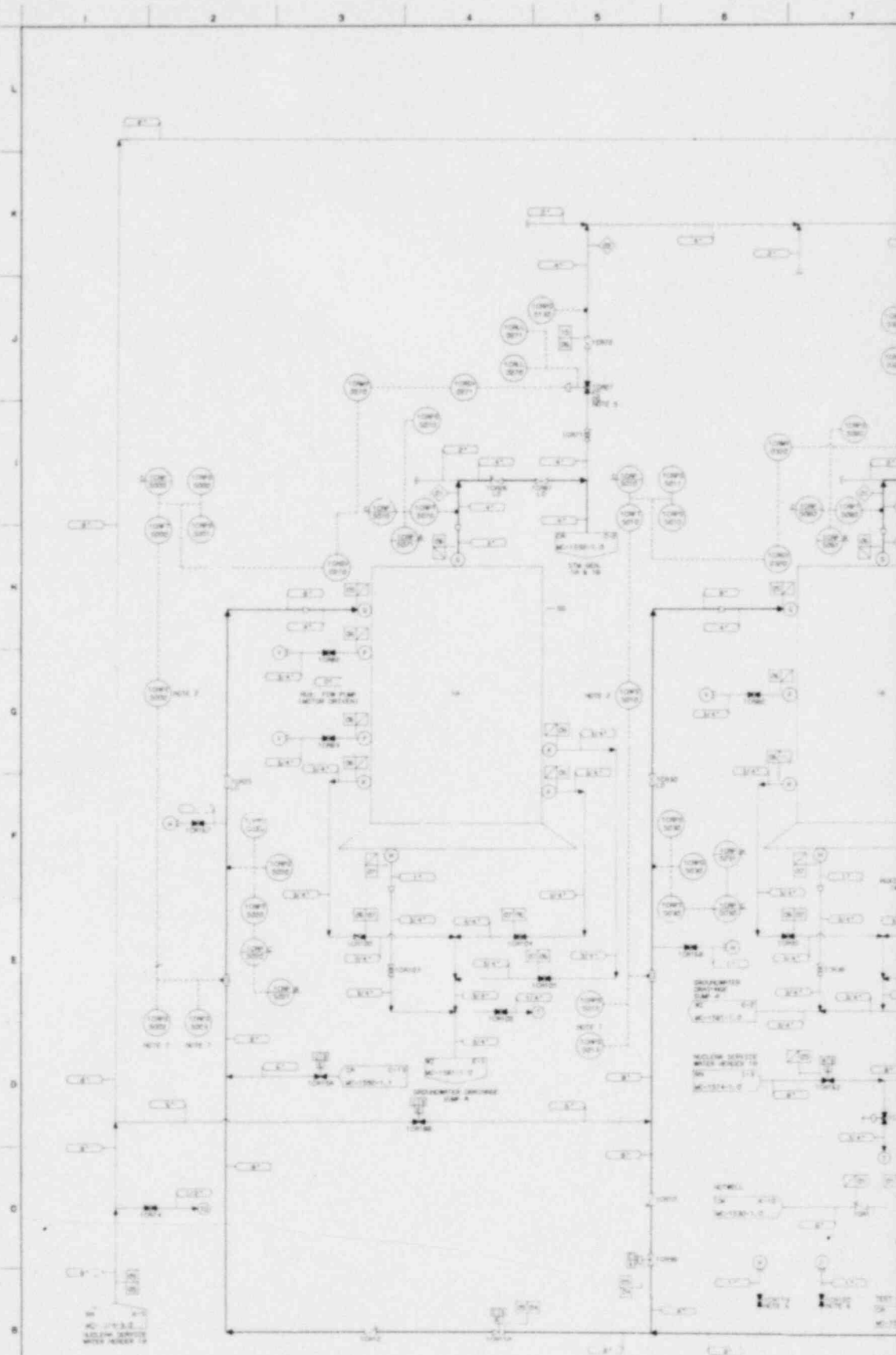
NO.	REVISION	DESCRIPTION	DATE	BY	CHKD.
1		REVISED FOR			
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7		REVISED FOR			
8		REVISED FOR			
9		REVISED FOR			
10		REVISED FOR			

NUCLEAR SAFETY RELATED

DUKE POWER COMPANY
WILHELM NUCLEAR STATION UNIT 1

PLUM DAMPING OF
AUXILIARY FEEDWATER SYSTEM
(COP)

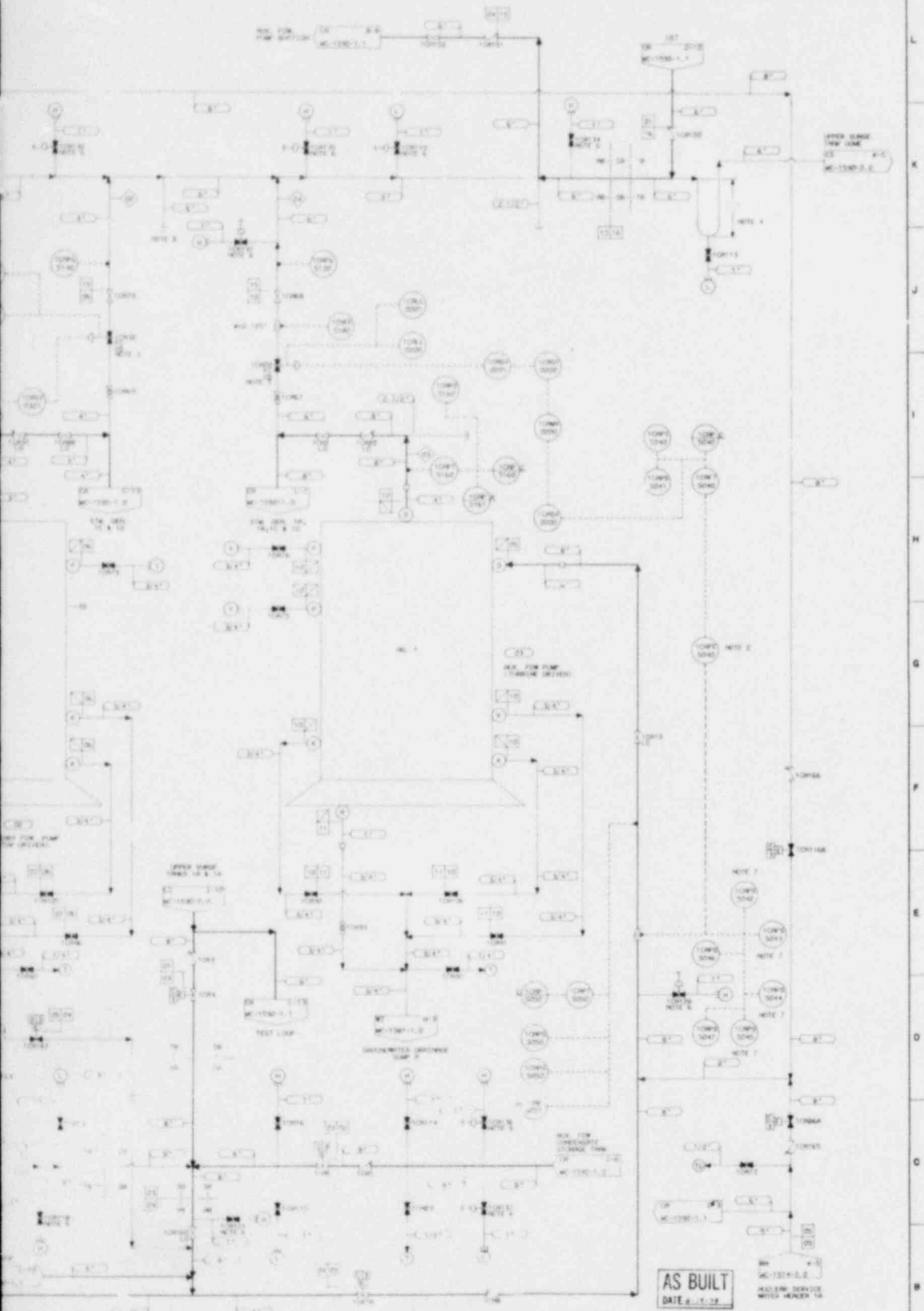
DATE: 11/11/64
DRAWN: J. J. BROWN
CHECKED: J. J. BROWN
SCALE: AS SHOWN
SHEET NO. 1 OF 2



- NOTES:
1. FLOW TO FURNACE TO CLASSING POTENTIAL TO SHUTTY EQUIPMENT
 2. FLOW TO TRANSPORTATION PROCESS, PREVENTION FOR
 3. INSTANTLY SHUTDOWN FLOW CONTROL, ALSO PREVENTS
 4. REVERSE FLOW CONTROL, ON A FLOW CONTROL VALVE ONLY.
 5. FLOW TO COIL, USE SHUT OFF IN TEST BELOW SET POINT CORRECTION
 6. ALL FLOW PUMP TEST LOOP, 1 AND FLOW LOOP ARE USED ON
 7. OPERATOR CONTROL, SHUT OFF
 8. FLOW AND SHUT OFF NEED AT CONNECTION FOR FLOW AND WORK
 9. PRESSURE SWITCH, CONTROL METHOD 2, 3, 4, 5, 6, 7, 8, 9, 10, 11, 12, 13, 14, 15, 16, 17, 18, 19, 20, 21, 22, 23, 24, 25, 26, 27, 28, 29, 30, 31, 32, 33, 34, 35, 36, 37, 38, 39, 40, 41, 42, 43, 44, 45, 46, 47, 48, 49, 50, 51, 52, 53, 54, 55, 56, 57, 58, 59, 60, 61, 62, 63, 64, 65, 66, 67, 68, 69, 70, 71, 72, 73, 74, 75, 76, 77, 78, 79, 80, 81, 82, 83, 84, 85, 86, 87, 88, 89, 90, 91, 92, 93, 94, 95, 96, 97, 98, 99, 100
 10. FLOW TO CONNECTION WITH ALCO FURNACE

CA SYSTEM
 FIG 3.2 SHT 2 OF 2
 REV 1

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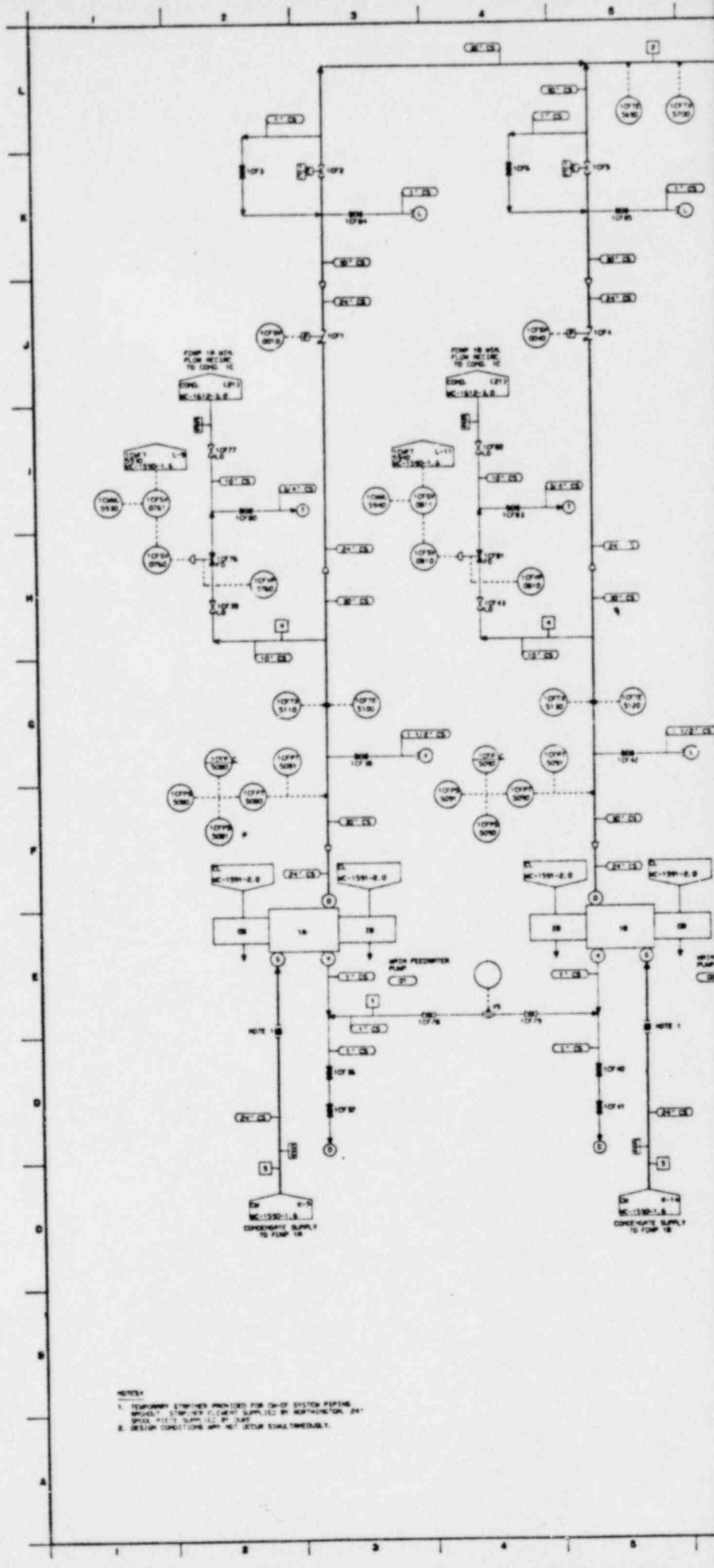
AS BUILT
DATE 8-11-78

- ▲ INTEGRATED PERFORM WITHIN REVISION - REV. 2 (10/12/78)
- ▲ CANCELED FOR PERFORM WITHIN REVISION - REV. 7 (10/11/78)

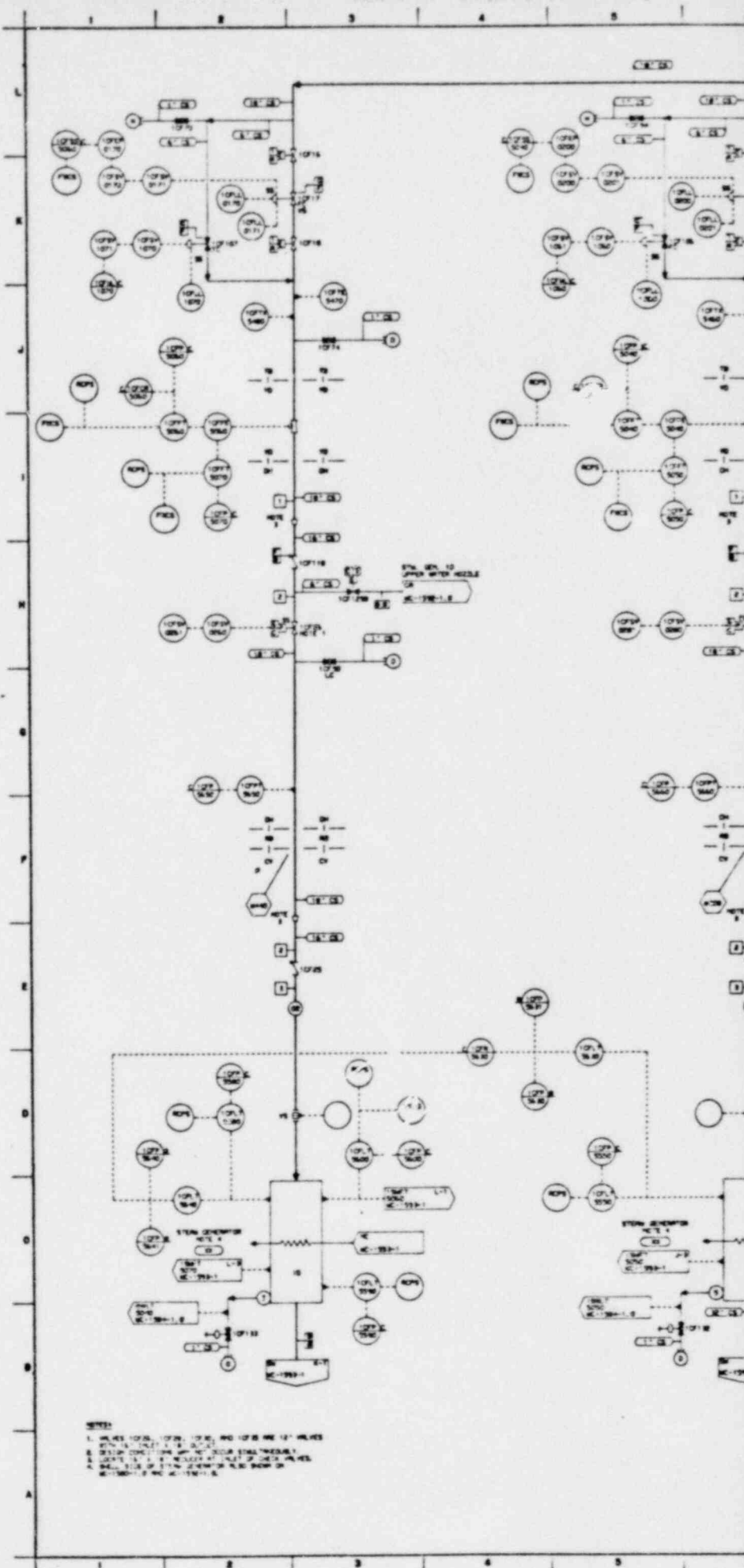
NUCLEAR SAFETY RELATED
 DUKE POWER COMPANY
 WILSON NUCLEAR STATION UNIT 1
 FLOW CONTROL OF
 AUXILIARY FEEDWATER SYSTEM
 CO.

NO.	REVISIONS	DATE	BY	CHKD	APP'D	DATE	REASON
1	REV. PERFORM WITHIN REVISION	10/12/78
2	REV. PERFORM WITHIN REVISION	10/11/78

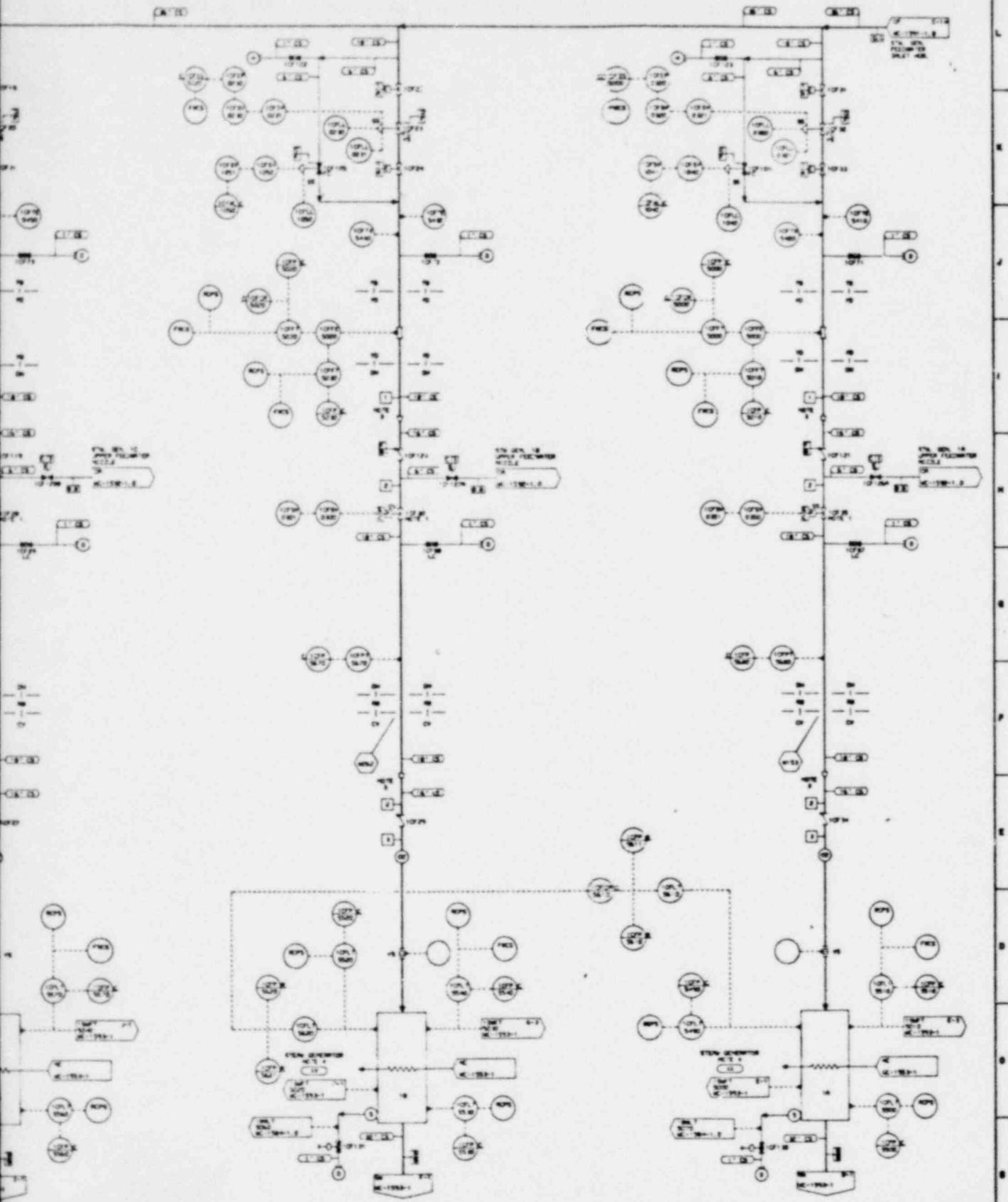
REVISION IN GENERAL - NOT AFFECTING REVISION - DATE
 REVISION IN GENERAL - NOT AFFECTING REVISION - DATE
 REVISION IN GENERAL - NOT AFFECTING REVISION - DATE
 DWG. NO. AC-1500-1.1



NOTES
 1. TEMPORARY STOPPERS PROVIDED FOR ON-OF SYSTEM PIPING.
 UNQUALIFIED STOP-KEY ELEMENT SUPPLIED BY NORTHINGTON 24"
 2. DESIGN CONDITIONS MAY NOT OCCUR SIMULTANEOUSLY.



1. VALUES 107A, 107B, 107C, AND 107D ARE 10" VALUES
 2. SWITCHES 107E, 107F, AND 107G
 3. DESIGN CONNECTIONS WHICH OCCUR SIMULTANEOUSLY
 4. CONTACTS 107H, 107I, 107J, AND 107K ARE 10" CONTACTS
 5. RELAY STATES OF SYSTEMS OF WHICH RELAY SYMBOLS
 ARE 107L-107R AND 107S-107T



NO.	DESCRIPTION	DATE
1	ISSUED FOR CONSTRUCTION	11/15/55
2	REVISED TO SHOW REVISIONS	11/15/55
3	REVISED TO SHOW REVISIONS	11/15/55
4	REVISED TO SHOW REVISIONS	11/15/55
5	REVISED TO SHOW REVISIONS	11/15/55

NO.	DESCRIPTION	DATE
1	ISSUED FOR CONSTRUCTION	11/15/55
2	REVISED TO SHOW REVISIONS	11/15/55
3	REVISED TO SHOW REVISIONS	11/15/55
4	REVISED TO SHOW REVISIONS	11/15/55
5	REVISED TO SHOW REVISIONS	11/15/55

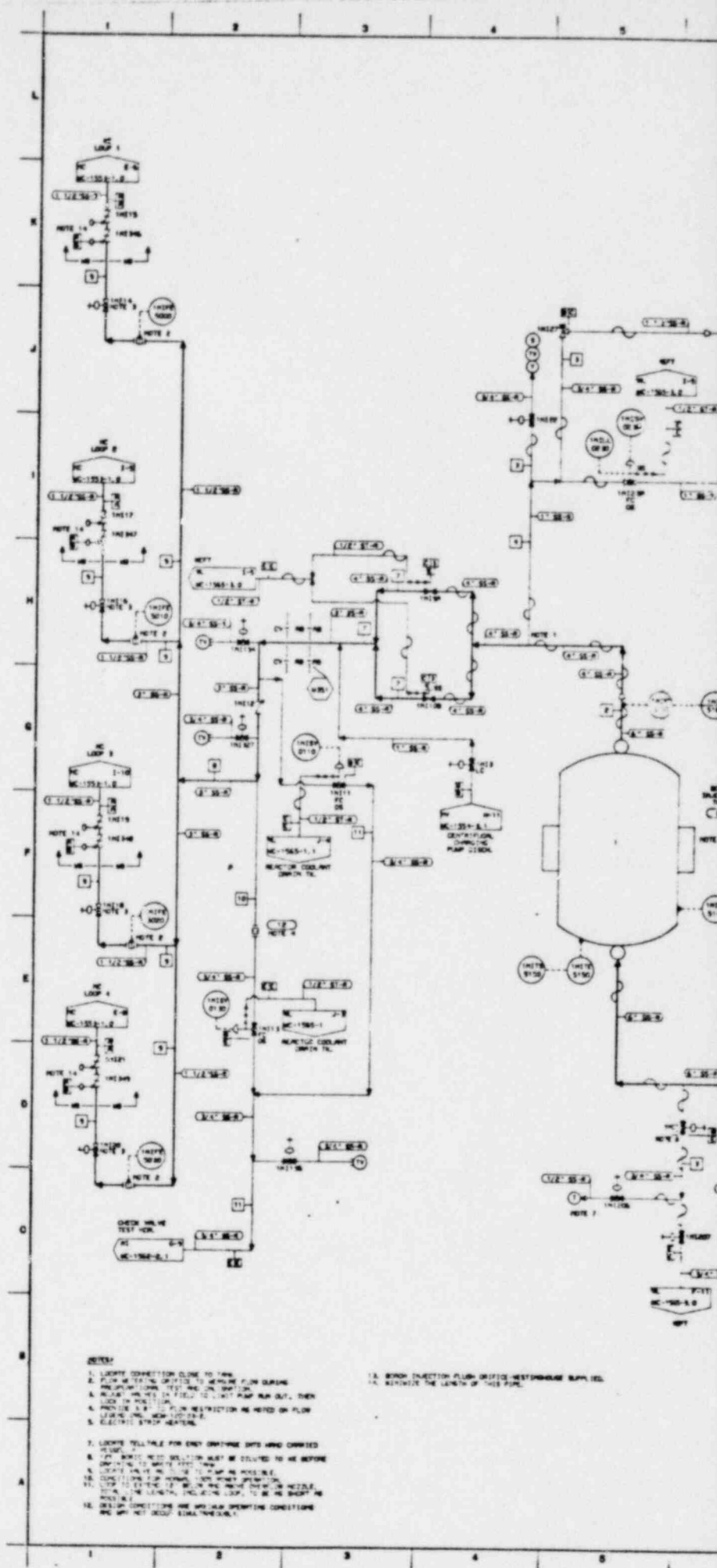
FIGURE 3.3
SHEET 2 of 2

NO.	REVISIONS	DATE	BY	CHECKED

NUCLEAR SAFETY RELATED

DUKE POWER COMPANY
WELLS FARGO STATION UNIT 1
ALSO KNOWN AS
REACTOR SYSTEM
1071

DATE: 11/15/55
DRAWN BY: [Name]
CHECKED BY: [Name]
SCALE: 1/4" = 1'-0"
SHEET NO. 1071-1



- NOTES:**
1. LOCKPIE CONNECTION CLOSE TO 100%
 2. FLOW METERING DEVICES TO MONITOR FLOW DURING OPERATION SHALL BE TESTED ON SHUT DOWN
 3. ALL VALVES IN THIS SYSTEM SHALL BE TESTED ON SHUT DOWN
 4. LOCK IN POSITION
 5. PROVIDE A FLOW RESTRICTION AS NOTED ON FLOW LEAKS ON NEW CONNECTIONS
 6. ELECTRIC STOP MEANS
 7. LOCKPIE TELLTALE FOR EACH SHUTTING DOWN AND CORRECTED STATUS
 8. ALL SHUTTING DOWN DEVICES MUST BE COLLECTED TO BE BEFORE SHUTTING DOWN THE SYSTEM
 9. LOCKPIE SHALL BE TESTED ON SHUT DOWN
 10. CONNECTIONS FOR NORMAL AND EMERGENCY OPERATION SHALL BE TESTED IN BOTH DIRECTIONS AND THE LENGTH OF THE LINE, INCLUDING LOOP, TO BE AS SHORT AS PRACTICABLE
 11. DESIGN CONNECTIONS ARE TO BE MADE UNDER CONDITIONS AND NOT TESTED UNDER CONDITIONS
 12. BENCH INJECTION FLOW DEVICES TESTS SHALL BE APPLIED TO RESPECTIVE THE LENGTH OF THIS PIPE

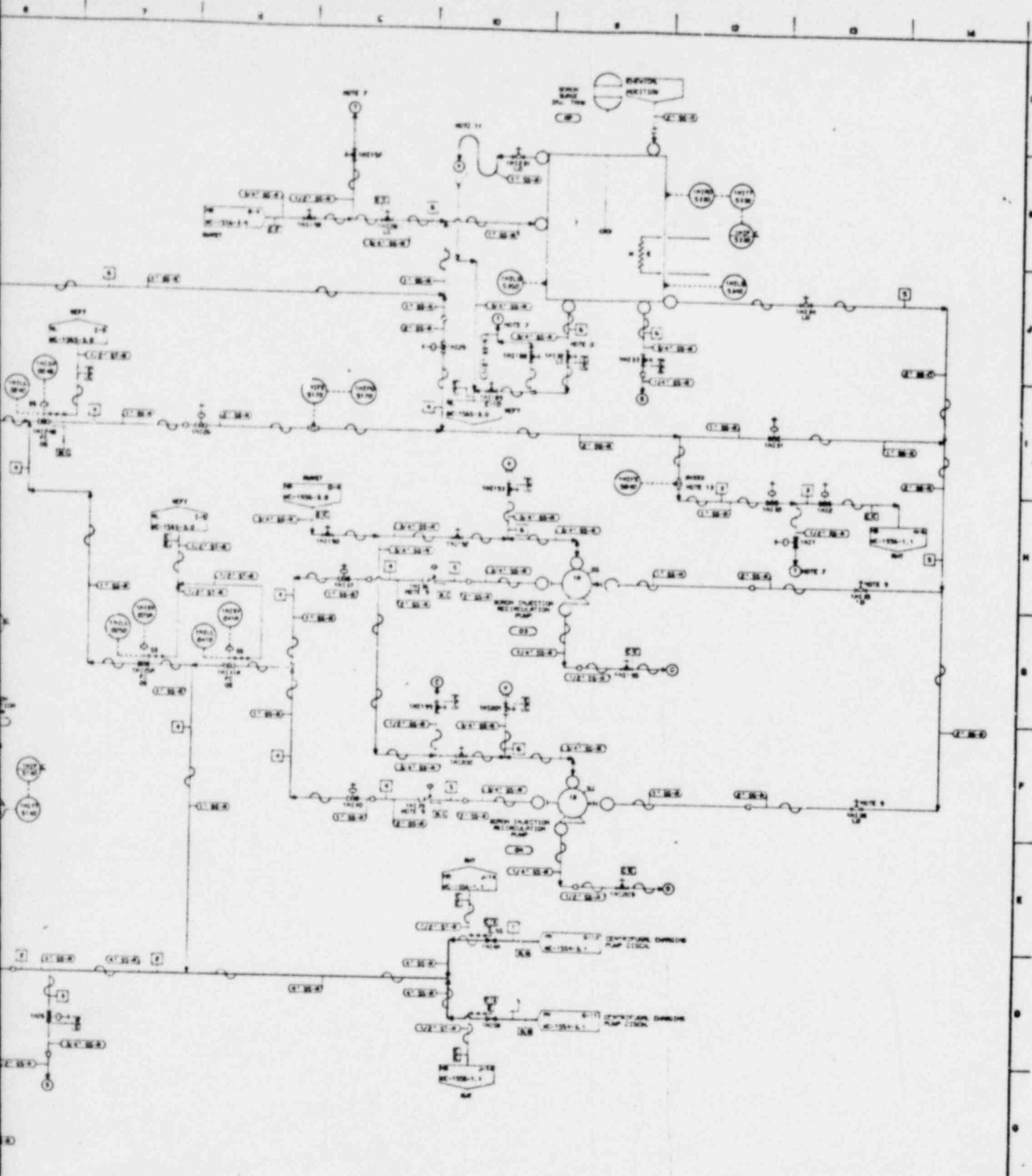


FIGURE 3.4
SHEET 1 of 2

REVISION	DATE	DESCRIPTION
1	11/11/68	ISSUED FOR CONSTRUCTION
2	11/11/68	REVISIONS TO TANK 1000-1.1
3	11/11/68	REVISIONS TO TANK 1000-1.2
4	11/11/68	REVISIONS TO TANK 1000-1.3
5	11/11/68	REVISIONS TO SAFETY INJECTION PUMP
6	11/11/68	REVISIONS TO SAFETY INJECTION SYSTEM
7	11/11/68	REVISIONS TO SAFETY INJECTION TANKS
8	11/11/68	REVISIONS TO SAFETY INJECTION PIPING
9	11/11/68	REVISIONS TO SAFETY INJECTION VALVES
10	11/11/68	REVISIONS TO SAFETY INJECTION CONTROLS

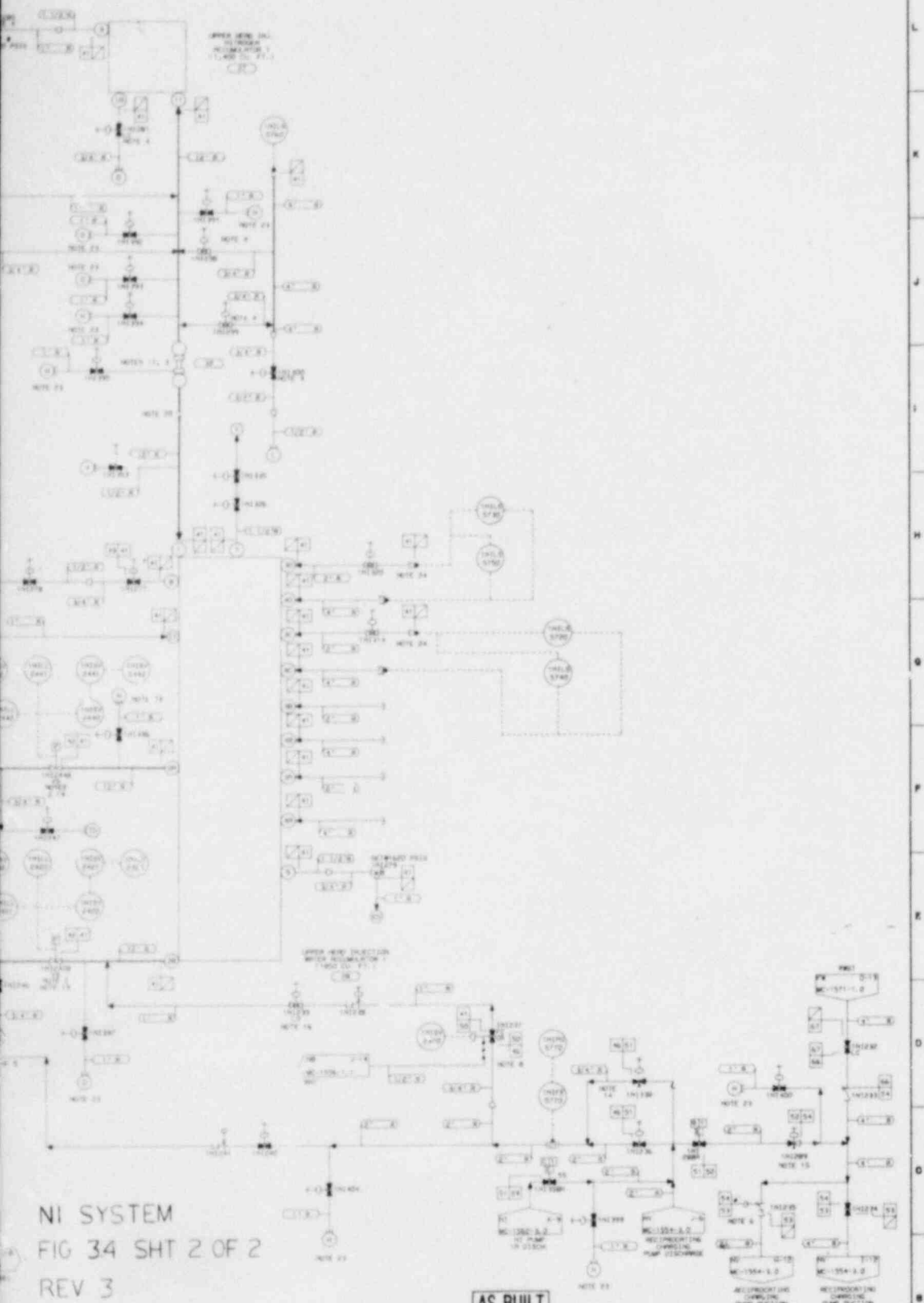
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NUCLEAR SAFETY RELATED

ORNL POWER COMPANY
ORNL REACTOR SAFETY SYSTEM (ORSS)

FLOW DIAGRAM OF
SAFETY INJECTION
SYSTEM (SIS)

ORNL NO. W-1000-1.1



NI SYSTEM
FIG 34 SHT 2 OF 2
REV 3

AS BUILT
DATE 4-19-78

NO.	DESCRIPTION	DATE	BY	CHKD
1	ISSUED FOR CONSTRUCTION	11-15-77	J. J. ...	J. J. ...
2	REVISED FOR
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★ ISSUED FOR CONSTRUCTION REVISION NO. 1 DATE 11-15-77
★ ISSUED FOR CONSTRUCTION REVISION NO. 2 DATE 4-19-78

NO.	REVISIONS	DATE	BY	CHKD
1	ISSUED FOR CONSTRUCTION	11-15-77	J. J. ...	J. J. ...
2	REVISED FOR
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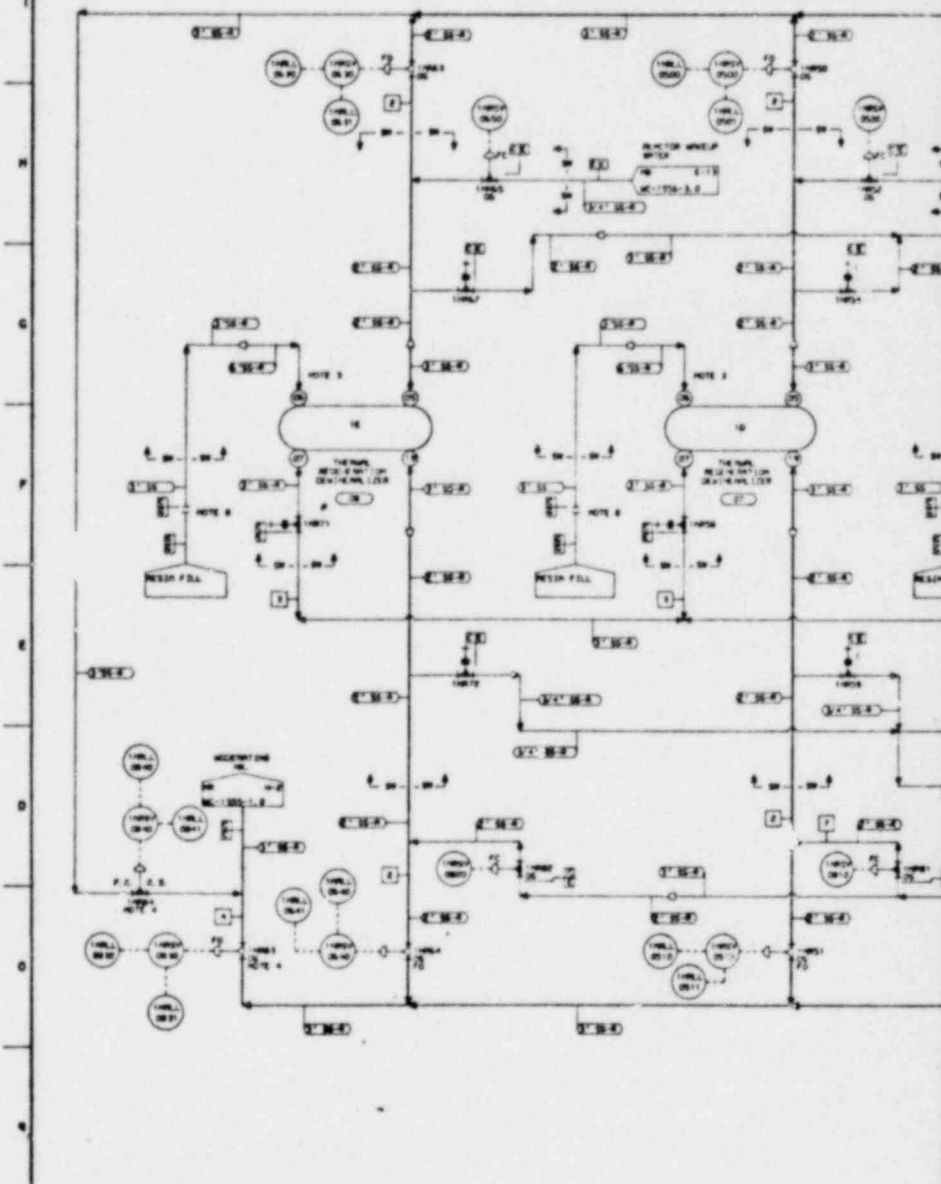
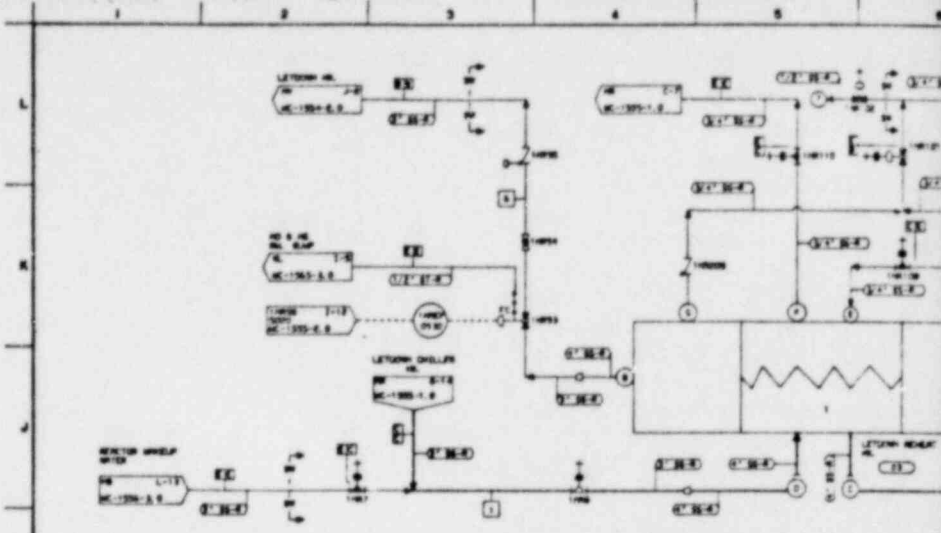
NUCLEAR SAFETY RELATED

DUKE POWER COMPANY
WILSON NUCLEAR STATION UNIT 1

FLOW DIAGRAM OF
SAFETY SELECTION SYSTEM
UPPER HEAD UNIT

NO.	REVISIONS	DATE	BY	CHKD
1	ISSUED FOR CONSTRUCTION	11-15-77	J. J. ...	J. J. ...
2	REVISED FOR
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DWG NO. M-150-4.0



- NOTES:**
1. ALL VALVES ARE USED FOR BURN HEAVY ISOLATION.
 2. FAIL TO BURN HEAVY POSITION.
 3. PRESSURE 1/2" OF 1/2" HIGH PLUMB BETWEEN DECONTAMINATOR AND LETTERING.
 4. BURN HEAVY ISOLATION SIGNAL.
 5. BURN HEAVY ISOLATION SIGNAL NOT USED IN FINAL RESULT.
 6. BURN HEAVY ISOLATION SIGNAL NOT USED IN FINAL RESULT.
 7. LETTERING HEAD SIGNAL TO DECONTAMINATOR ISOLATION SIGNAL.
 8. 1/2" BURN HEAVY ISOLATION SIGNAL USED FOR BURN HEAVY ISOLATION.

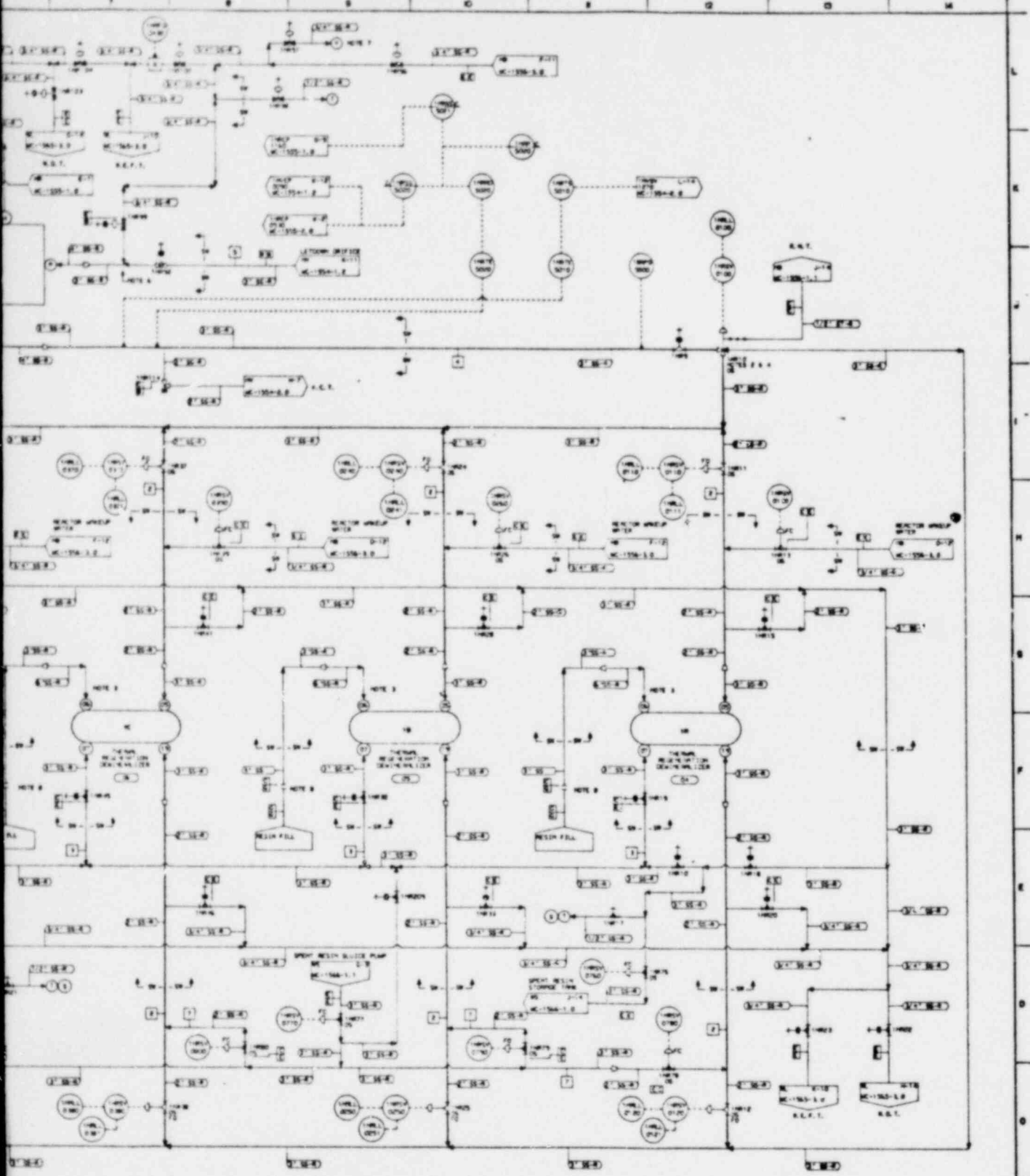


FIGURE 3.5
SHEET 1 of 1

NO.	REV.	DATE	DESCRIPTION
001	001	000000	ISSUED FOR CONSTRUCTION
002	001	000000	ISSUED FOR CONSTRUCTION
003	001	000000	ISSUED FOR CONSTRUCTION
004	001	000000	ISSUED FOR CONSTRUCTION
005	001	000000	ISSUED FOR CONSTRUCTION
006	001	000000	ISSUED FOR CONSTRUCTION
007	001	000000	ISSUED FOR CONSTRUCTION
008	001	000000	ISSUED FOR CONSTRUCTION
009	001	000000	ISSUED FOR CONSTRUCTION
010	001	000000	ISSUED FOR CONSTRUCTION

NO.	REV.	DATE	DESCRIPTION
001	001	000000	ISSUED FOR CONSTRUCTION
002	001	000000	ISSUED FOR CONSTRUCTION
003	001	000000	ISSUED FOR CONSTRUCTION
004	001	000000	ISSUED FOR CONSTRUCTION
005	001	000000	ISSUED FOR CONSTRUCTION
006	001	000000	ISSUED FOR CONSTRUCTION
007	001	000000	ISSUED FOR CONSTRUCTION
008	001	000000	ISSUED FOR CONSTRUCTION
009	001	000000	ISSUED FOR CONSTRUCTION
010	001	000000	ISSUED FOR CONSTRUCTION

NUCLEAR SAFETY RELATED

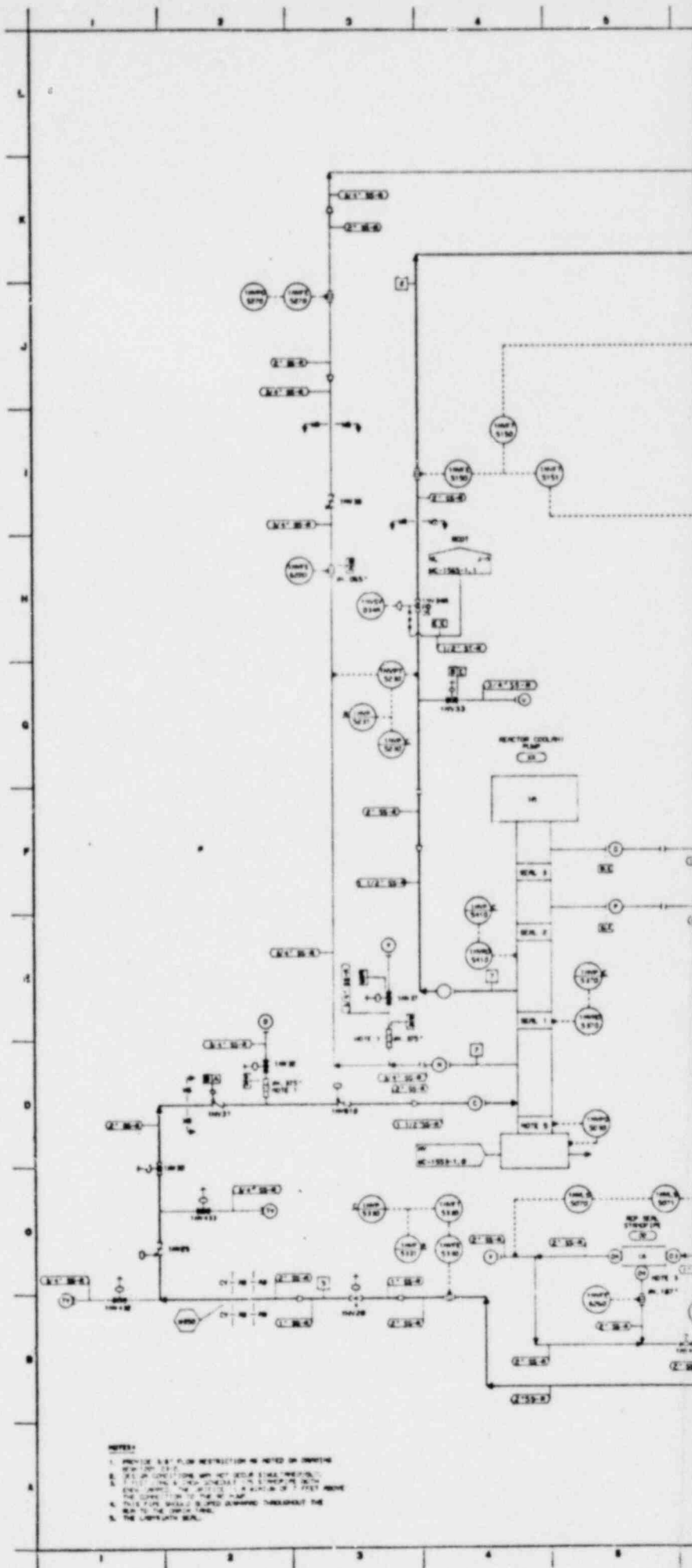
DUKE POWER COMPANY
WALTON RAYBURN STATION UNIT 1

PLAN NUMBER OF
WATER FILLS

DATE: 000000

SCALE: 1/4" = 1'-0"

NO. 100-100-1



- NOTES:**
1. PROVIDE 5:47 FLOW RESTRICTION AS NOTED ON DRAWING.
 2. ALL FLOW DIRECTIONS ARE AS SHOWN UNLESS OTHERWISE NOTED.
 3. FLOW DIRECTION IS INDICATED BY AN ARROW IN THE PIPE.
 4. THIS TYPE SYMBOL IS USED THROUGHOUT THE DRAWING TO INDICATE THE DIRECTION OF FLOW.
 5. THE LETTERS IN THE SYMBOLS ARE AS SHOWN UNLESS OTHERWISE NOTED.

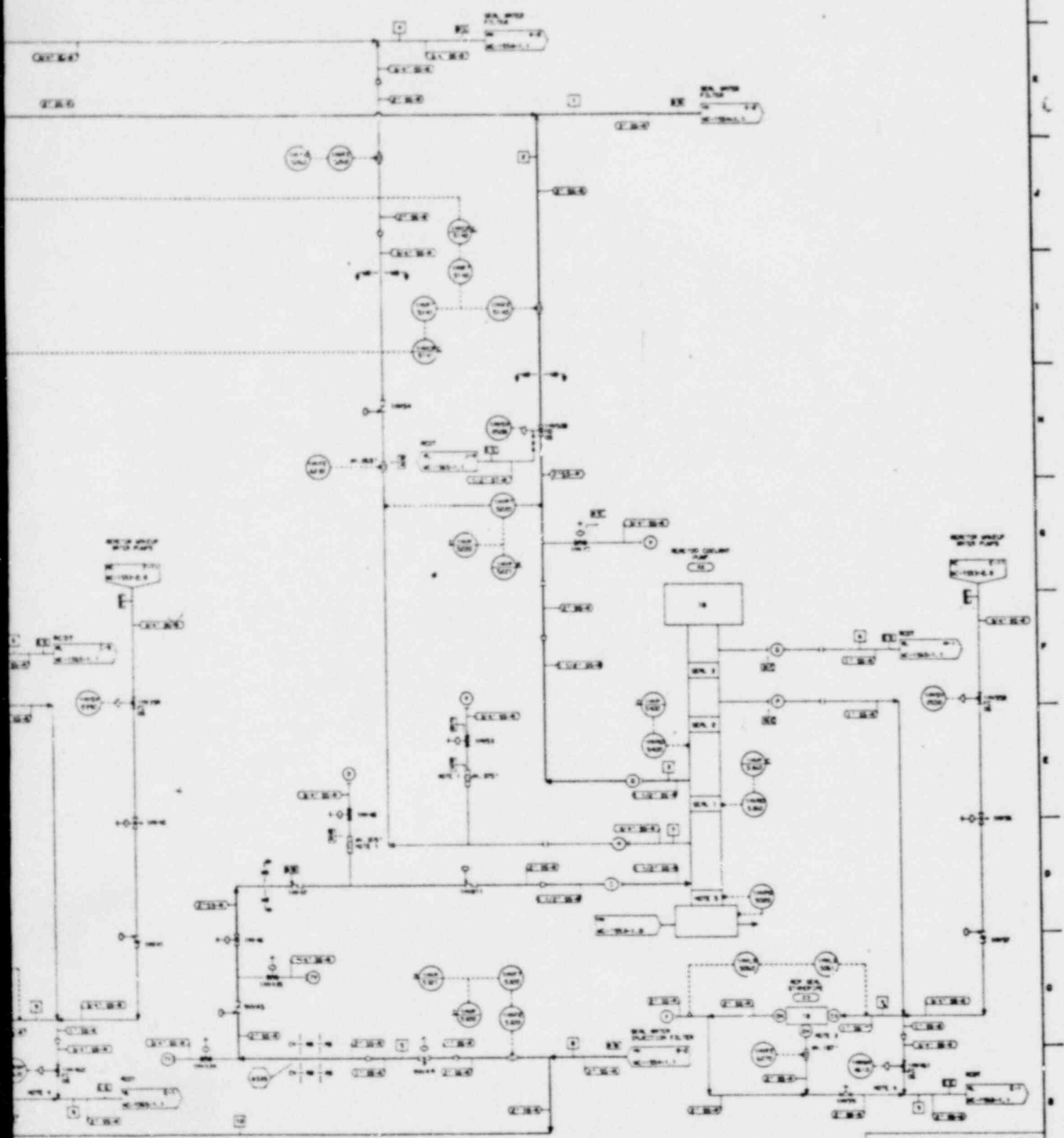
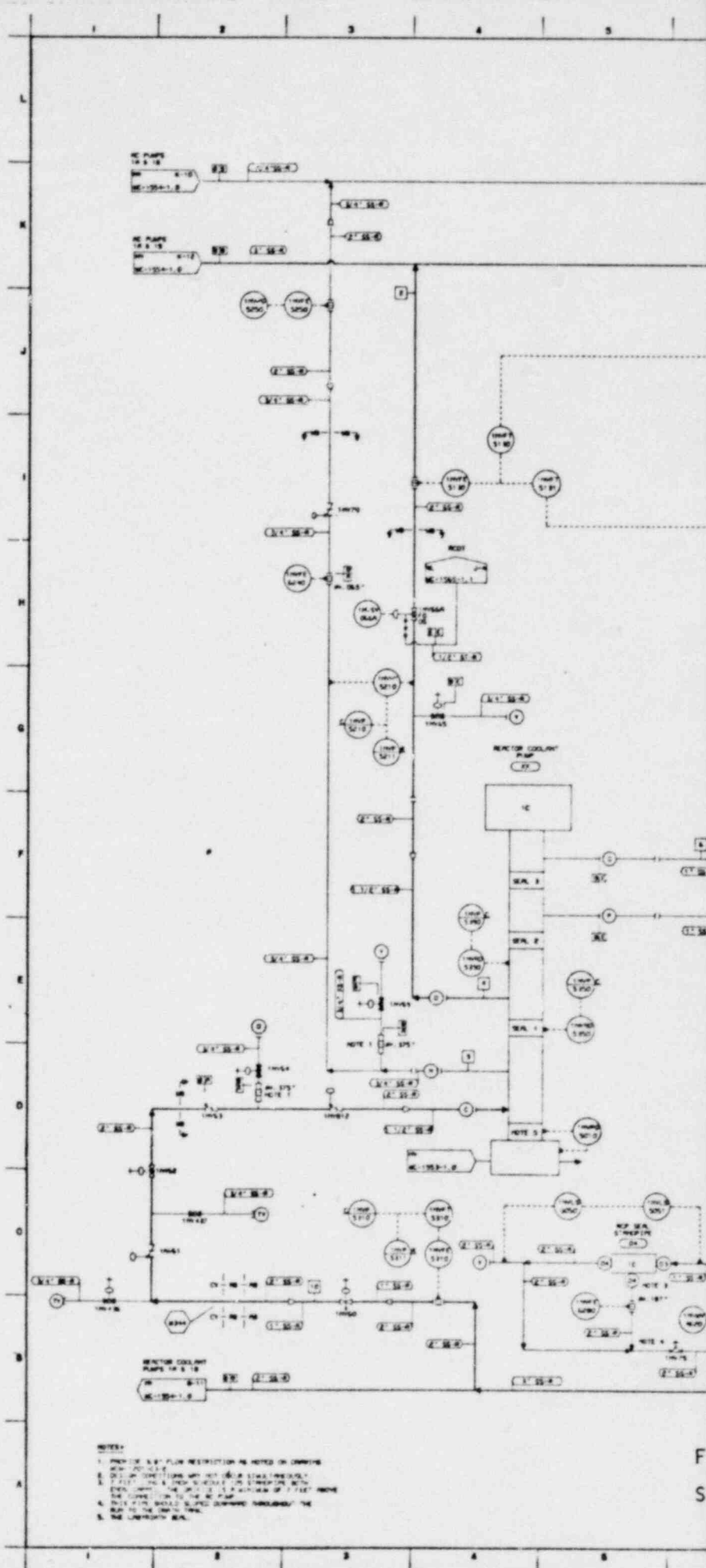


FIGURE 3.6
SHEET 1 of 7

NO.	DESCRIPTION	REVISION	DATE
1	ISSUED FOR CONSTRUCTION		7-28-66
2	REVISION		8-1-66
3	REVISION		8-1-66
4	REVISION		8-1-66
5	REVISION		8-1-66
6	REVISION		8-1-66
7	REVISION		8-1-66
8	REVISION		8-1-66
9	REVISION		8-1-66
10	REVISION		8-1-66

NO.	REVISIONS	DATE	BY	CHKD.
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NUCLEAR SAFETY RELATED
 DUKE POWER COMPANY
 WEDGEMOUNTED REACTOR UNIT 1
 PLAN SKETCH OF
 REACTOR & VALVE CONTROL
 SYSTEM (REV)
 DRAWING NO. 3-100-1-1
 SHEET 1 OF 7
 DATE 7-28-66
 DESIGNED BY: [Name]
 CHECKED BY: [Name]
 APPROVED BY: [Name]



NOTES:

1. CHECK FOR AIR FLOW RESTRICTION AS NOTED ON DRAWING.
2. CHECK FOR AIR FLOW RESTRICTION AS NOTED ON DRAWING.
3. CHECK FOR AIR FLOW RESTRICTION AS NOTED ON DRAWING.
4. CHECK FOR AIR FLOW RESTRICTION AS NOTED ON DRAWING.
5. THE LOWEST SEAL.

F
S

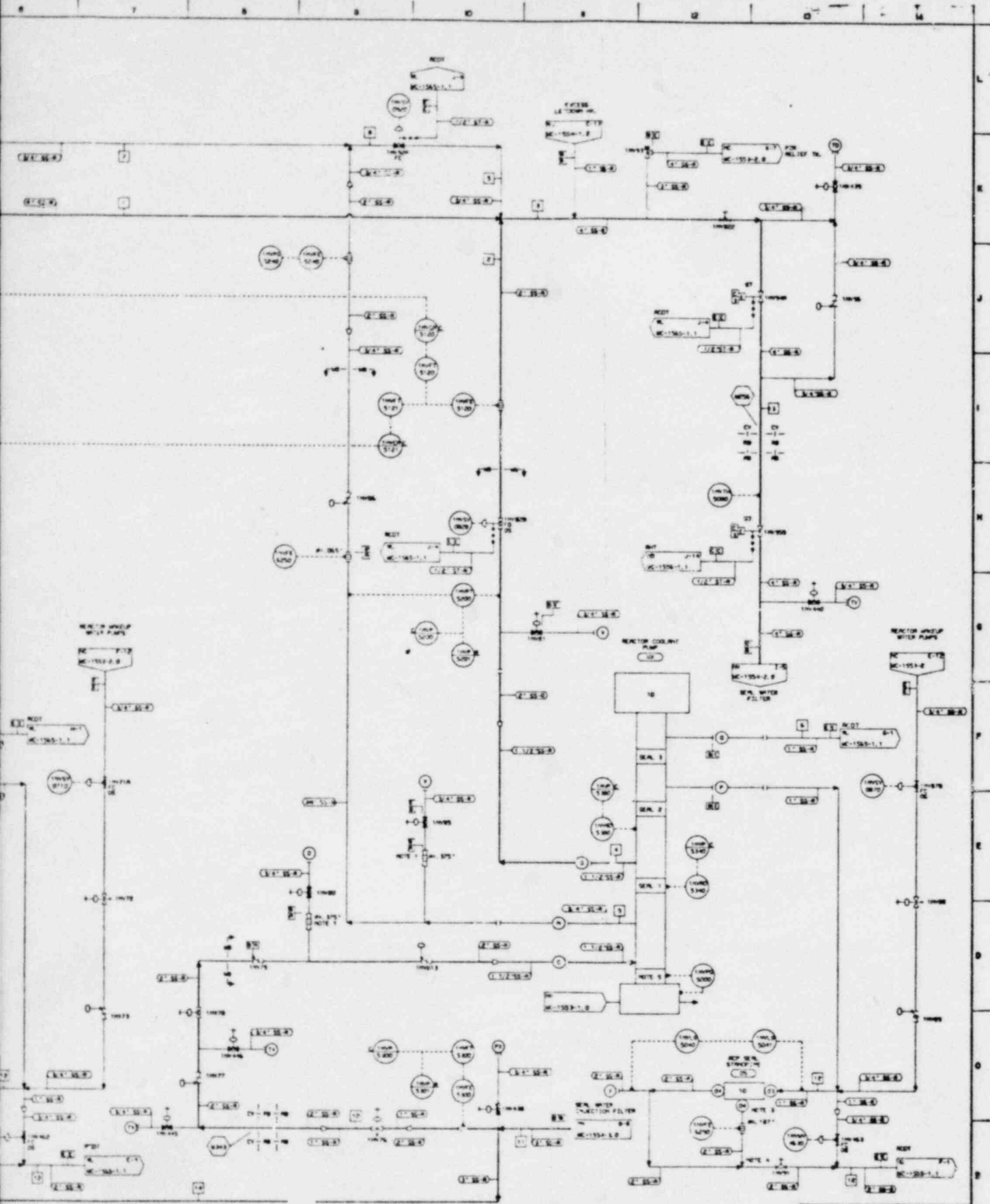


FIGURE 3.6
SHEET 2 of 7

NO.	DESCRIPTION	DATE	BY	CHKD.
1	ISSUED FOR CONSTRUCTION	10/1/58	J. W. BROWN	H. W. BROWN
2	REVISION	10/1/58	J. W. BROWN	H. W. BROWN
3	REVISION	10/1/58	J. W. BROWN	H. W. BROWN
4	REVISION	10/1/58	J. W. BROWN	H. W. BROWN
5	REVISION	10/1/58	J. W. BROWN	H. W. BROWN
6	REVISION	10/1/58	J. W. BROWN	H. W. BROWN
7	REVISION	10/1/58	J. W. BROWN	H. W. BROWN
8	REVISION	10/1/58	J. W. BROWN	H. W. BROWN
9	REVISION	10/1/58	J. W. BROWN	H. W. BROWN
10	REVISION	10/1/58	J. W. BROWN	H. W. BROWN

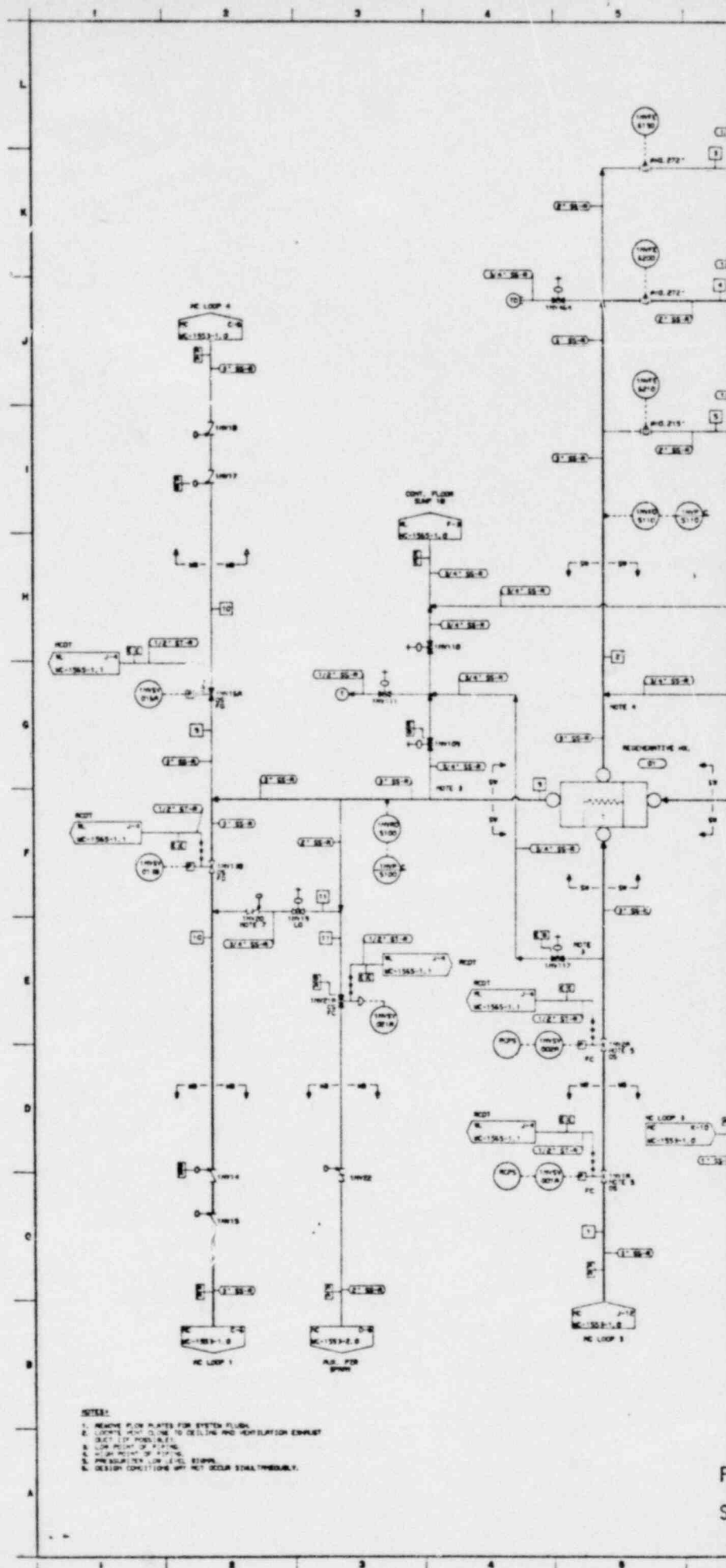
NUCLEAR SAFETY RELATED

DUKE POWER COMPANY
WEDGEMOUNT NUCLEAR STATION UNIT 1

PLUM DIAGRAM OF
GENERATOR 1 (LINE CONTROL)
SYSTEM (C-1)

NO.	REVISIONS	DATE	BY	CHKD.
1	ISSUED FOR CONSTRUCTION	10/1/58	J. W. BROWN	H. W. BROWN
2	REVISION	10/1/58	J. W. BROWN	H. W. BROWN
3	REVISION	10/1/58	J. W. BROWN	H. W. BROWN
4	REVISION	10/1/58	J. W. BROWN	H. W. BROWN
5	REVISION	10/1/58	J. W. BROWN	H. W. BROWN
6	REVISION	10/1/58	J. W. BROWN	H. W. BROWN
7	REVISION	10/1/58	J. W. BROWN	H. W. BROWN
8	REVISION	10/1/58	J. W. BROWN	H. W. BROWN
9	REVISION	10/1/58	J. W. BROWN	H. W. BROWN
10	REVISION	10/1/58	J. W. BROWN	H. W. BROWN

DWG. NO. MC-155A-1.1
K-6-1



NOTES:

1. MAKE UP PLATE FOR SYSTEM FLUID.
2. LOCATE AND CLAMP TO DRAINAGE AND VENTILATION EXHAUST.
3. LOW POINT OF PIPELINE.
4. HIGH POINT OF PIPELINE.
5. PRESSURE TEST LOW & HIGH POINTS.
6. DESIGN CONDITIONS SHY NOT OCCUR SIMULTANEOUSLY.

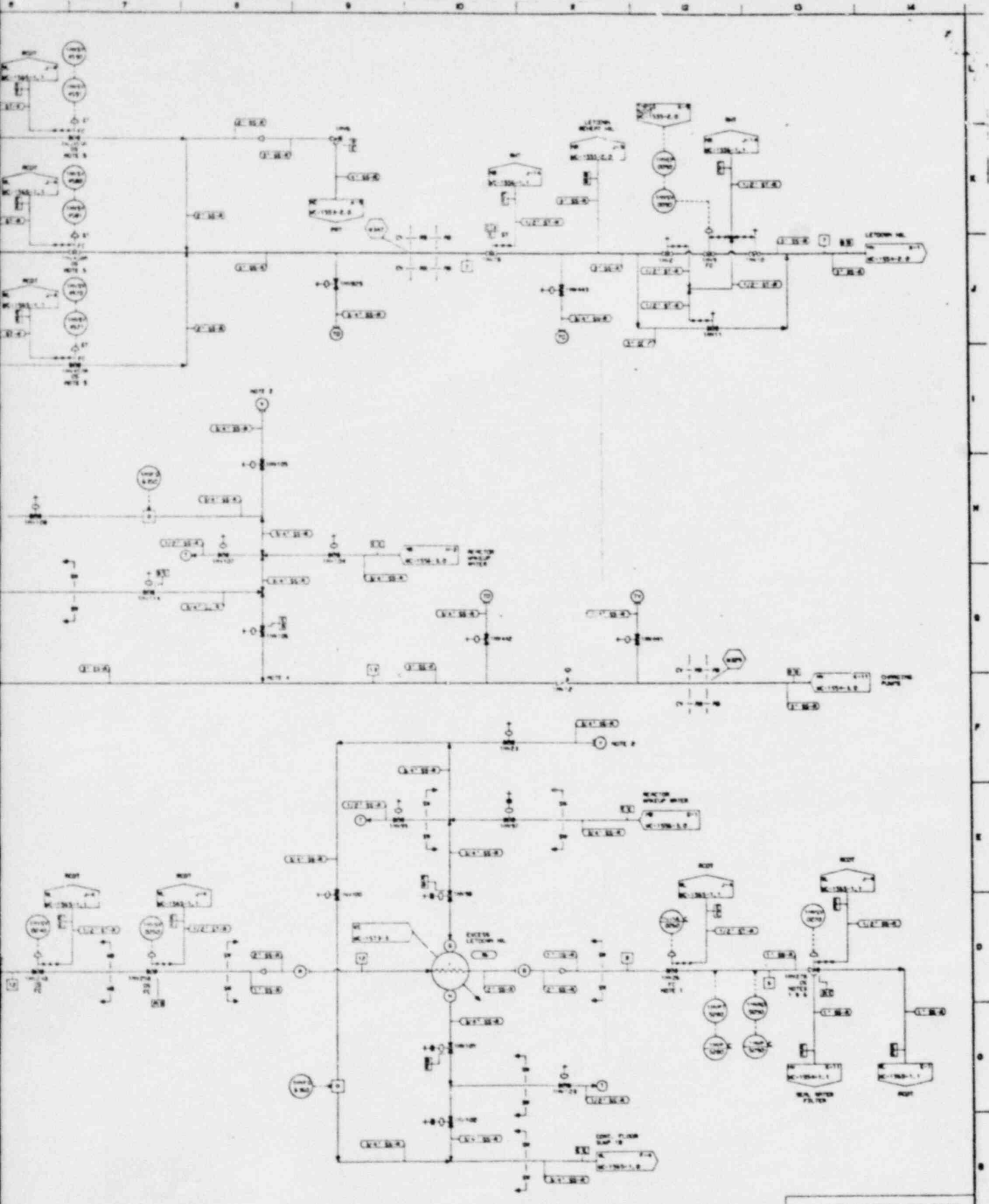
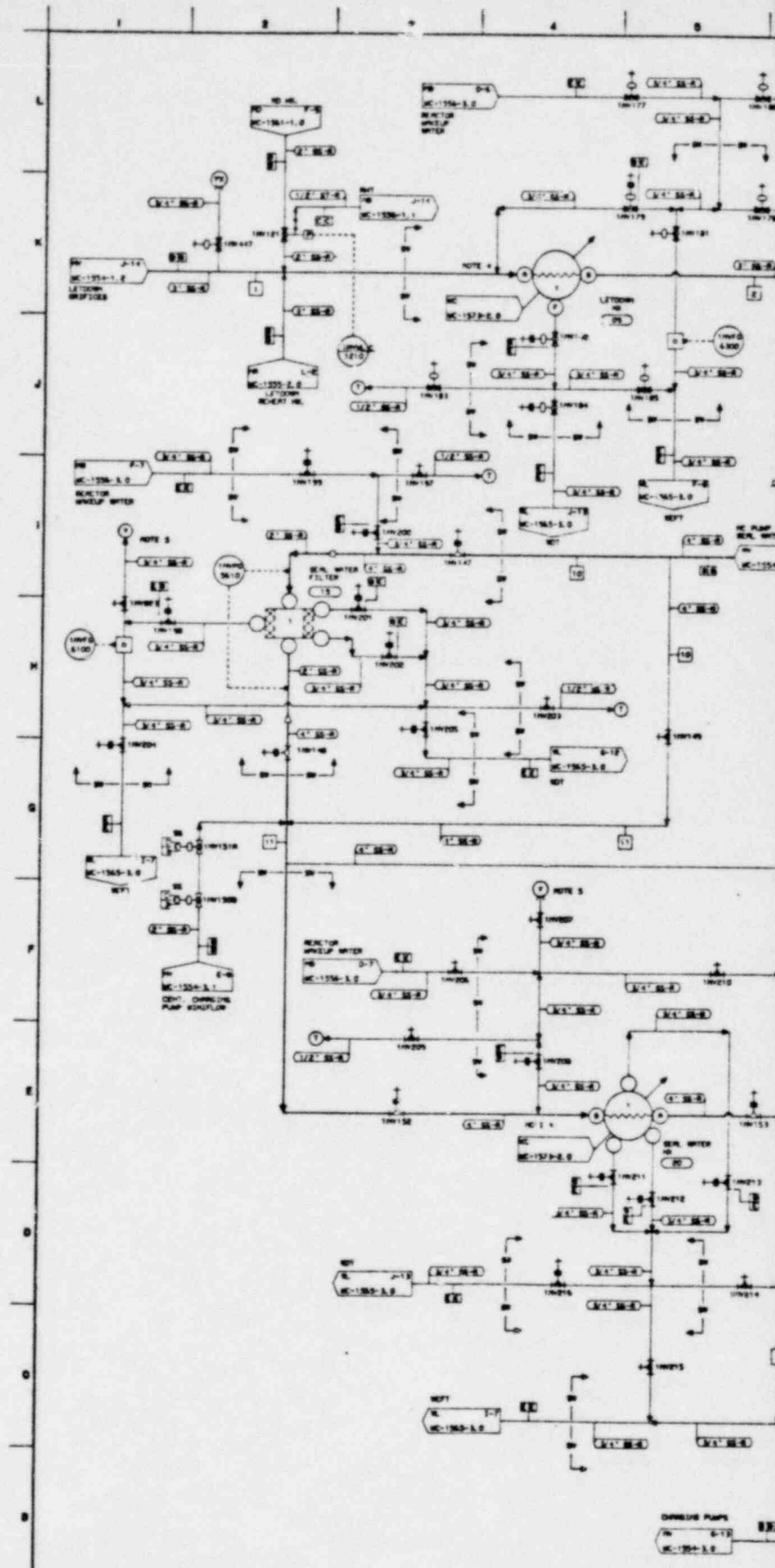


FIGURE 3.6
SHEET 3 of 7

1	1	1	1	1	1
2	2	2	2	2	2
3	3	3	3	3	3
4	4	4	4	4	4
5	5	5	5	5	5
6	6	6	6	6	6
7	7	7	7	7	7
8	8	8	8	8	8
9	9	9	9	9	9
10	10	10	10	10	10
11	11	11	11	11	11
12	12	12	12	12	12
13	13	13	13	13	13
14	14	14	14	14	14
15	15	15	15	15	15
16	16	16	16	16	16
17	17	17	17	17	17
18	18	18	18	18	18
19	19	19	19	19	19
20	20	20	20	20	20

NUCLEAR SAFETY RELATED
 BUREAU OF NUCLEAR POWER
 WASHINGTON, D.C. 20545
 DIVISION OF DESIGN & VALUE CONTROL
 SYSTEM UNIT

4861



NOTES:
 1. VALVES AND PIPES WITH FLOW TO HOUSE CONTROL ROOM.
 2. FLOW TO HOUSE CONTROL ROOM MUST BE STOPPED IN ORDER TO BE SHOWN ON P. 1 OF SCHEMATIC.
 3. THE FLOW DIRECTION MUST NOT BE STOPPED UNLESS NECESSARY.
 4. FLOW MUST BE STOPPED IN ORDER TO BE SHOWN ON P. 1 OF SCHEMATIC.
 5. VALVES MUST BE CLOSED TO CEILING AND VENTILATION EXHAUST SUCTION IF POSSIBLE.

NORMAL OPERATING CONDITIONS		DESIGN CONDITIONS	
NO.	DESCRIPTION	NO.	DESCRIPTION
1	...	1	...
2	...	2	...
3	...	3	...
4	...	4	...
5	...	5	...
6	...	6	...
7	...	7	...
8	...	8	...
9	...	9	...
10	...	10	...
11	...	11	...
12	...	12	...
13	...	13	...
14	...	14	...
15	...	15	...
16	...	16	...
17	...	17	...
18	...	18	...
19	...	19	...
20	...	20	...

CONDENSER PUMPS
 MC-130-3.0

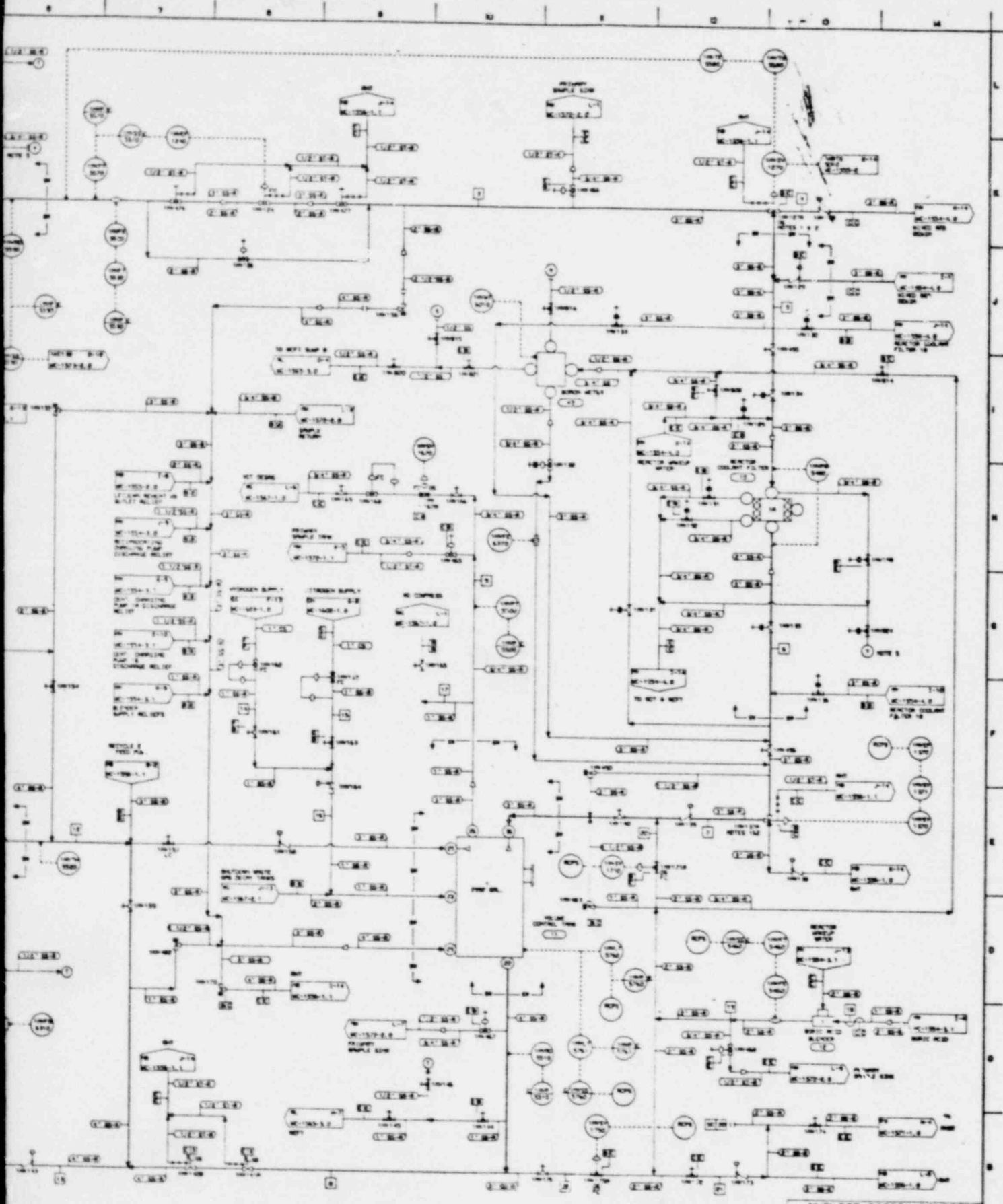
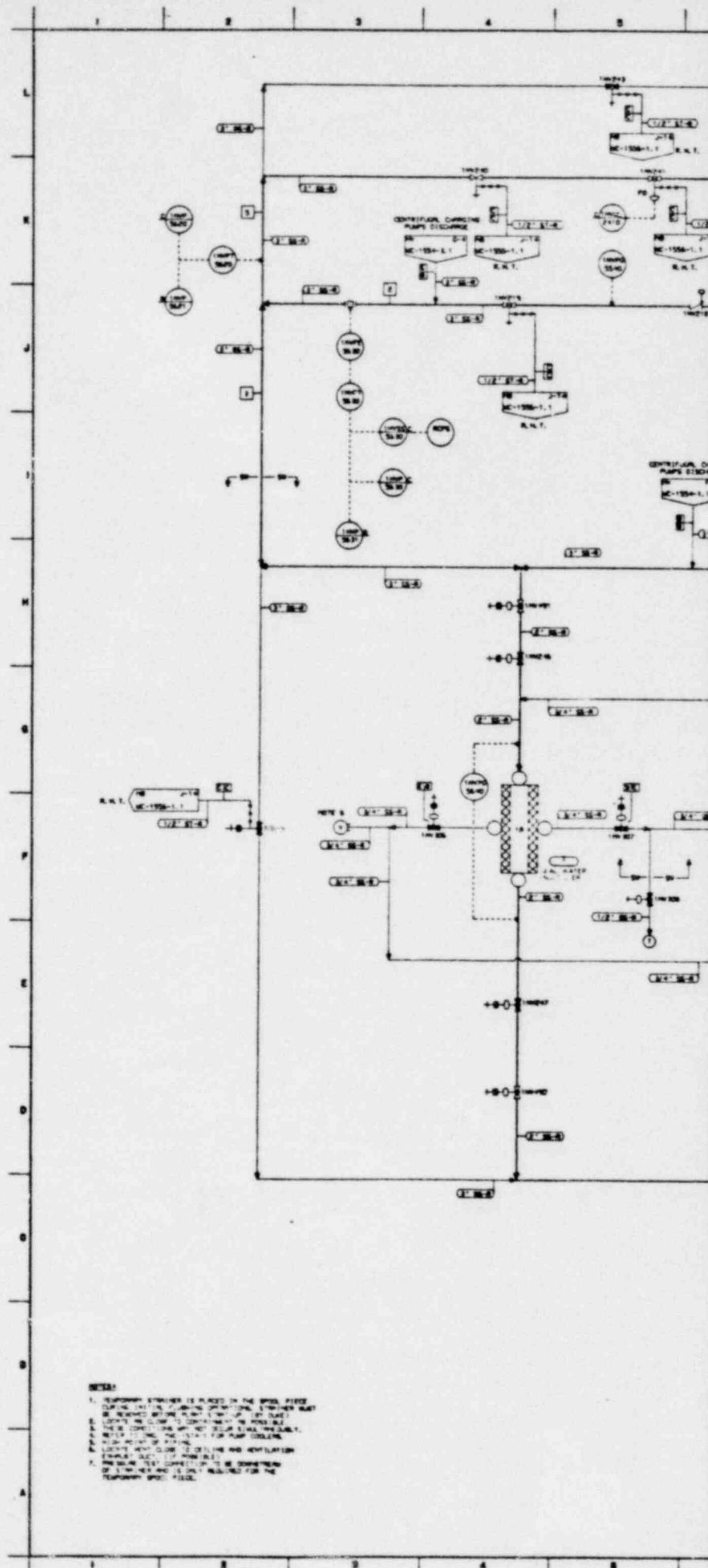


FIGURE 3.6
SHEET 4 of 7

REVISIONS	
NO.	DESCRIPTION
1	...
2	...
3	...
4	...
5	...
6	...
7	...
8	...
9	...
10	...

NUCLEAR SAFETY RELATED
 DUKES POWER COMPANY
 WINDY HOLLOW STATION UNIT 1
 PLAN DRAWING OF
 CHECKER & VALVE CONTROL
 SYSTEM UNIT

REV



- NOTES:**
1. TEMPORARY STOPPER IS PLACED IN THE SPILL PIECE FOR THE INITIAL OPERATIONS. STOPPER MUST BE REMOVED BEFORE PLANT IS STARTED.
 2. LOCATE THE STOPPER AS CLOSE TO THE SPILL PIECE AS POSSIBLE.
 3. THE STOPPER MUST BE REMOVED BEFORE STARTING.
 4. NOTIFY THE OPERATOR FOR PUMP COOLING.
 5. CHECK THE OIL LEVEL.
 6. LOCATE THE STOPPER AS CLOSE TO THE SPILL PIECE AS POSSIBLE.
 7. THE STOPPER MUST BE REMOVED BEFORE STARTING.
 8. THE STOPPER MUST BE REMOVED BEFORE THE TEMPORARY STOPPER IS PLACED.

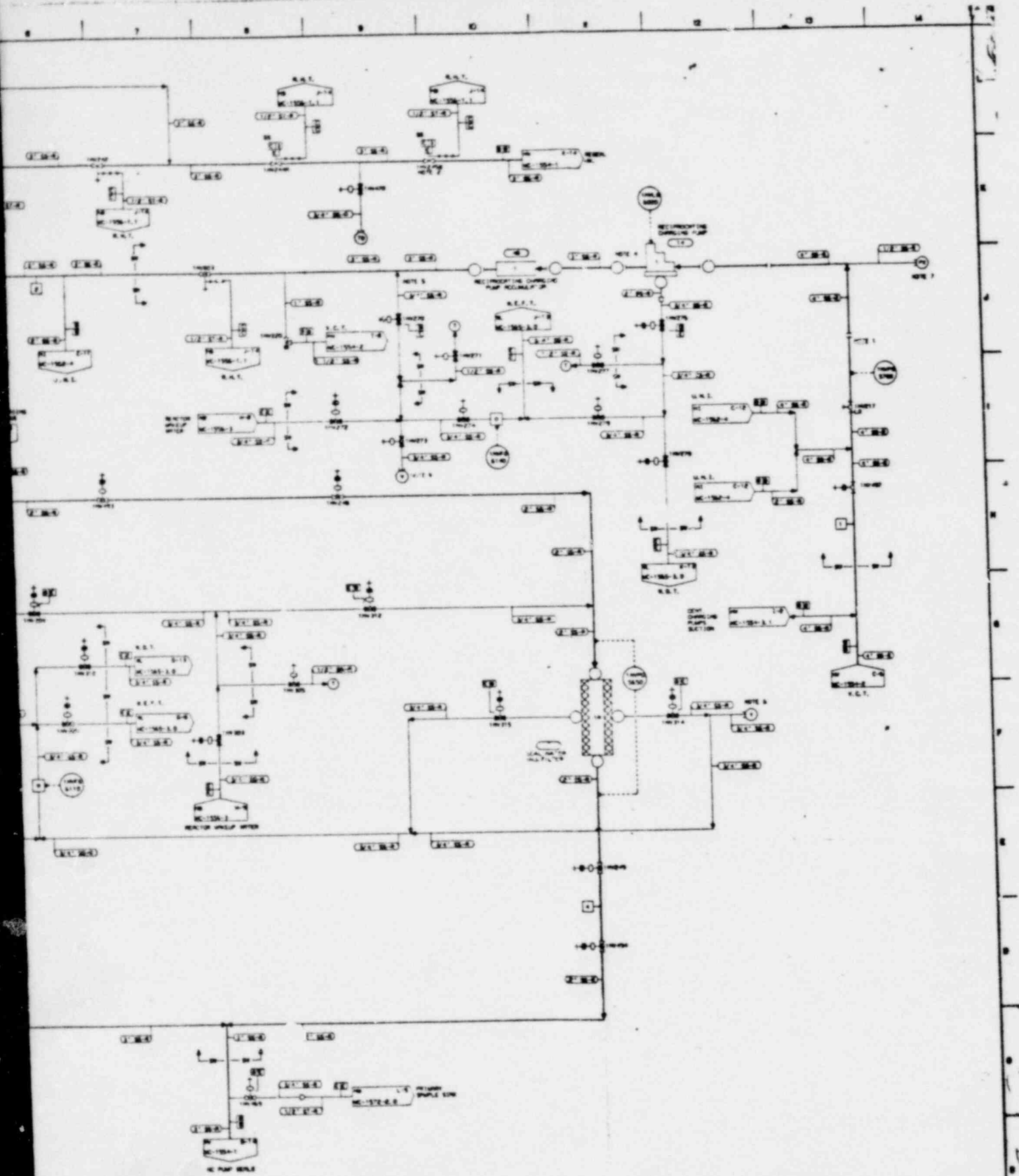


FIGURE 3.6
SHEET 5 of 7

NO.	DESCRIPTION	DATE
1	DESIGNED	11/15/54
2	CHECKED	11/15/54
3	APPROVED	11/15/54
4	REVISION	
5	REVISION	
6	REVISION	
7	REVISION	
8	REVISION	
9	REVISION	
10	REVISION	
11	REVISION	
12	REVISION	

NO.	REVISIONS	DATE	BY	CHKD.
1	DESIGNED	11/15/54		
2	CHECKED	11/15/54		
3	APPROVED	11/15/54		
4	REVISION			
5	REVISION			
6	REVISION			
7	REVISION			
8	REVISION			
9	REVISION			
10	REVISION			
11	REVISION			
12	REVISION			

Sheet 5 of 7

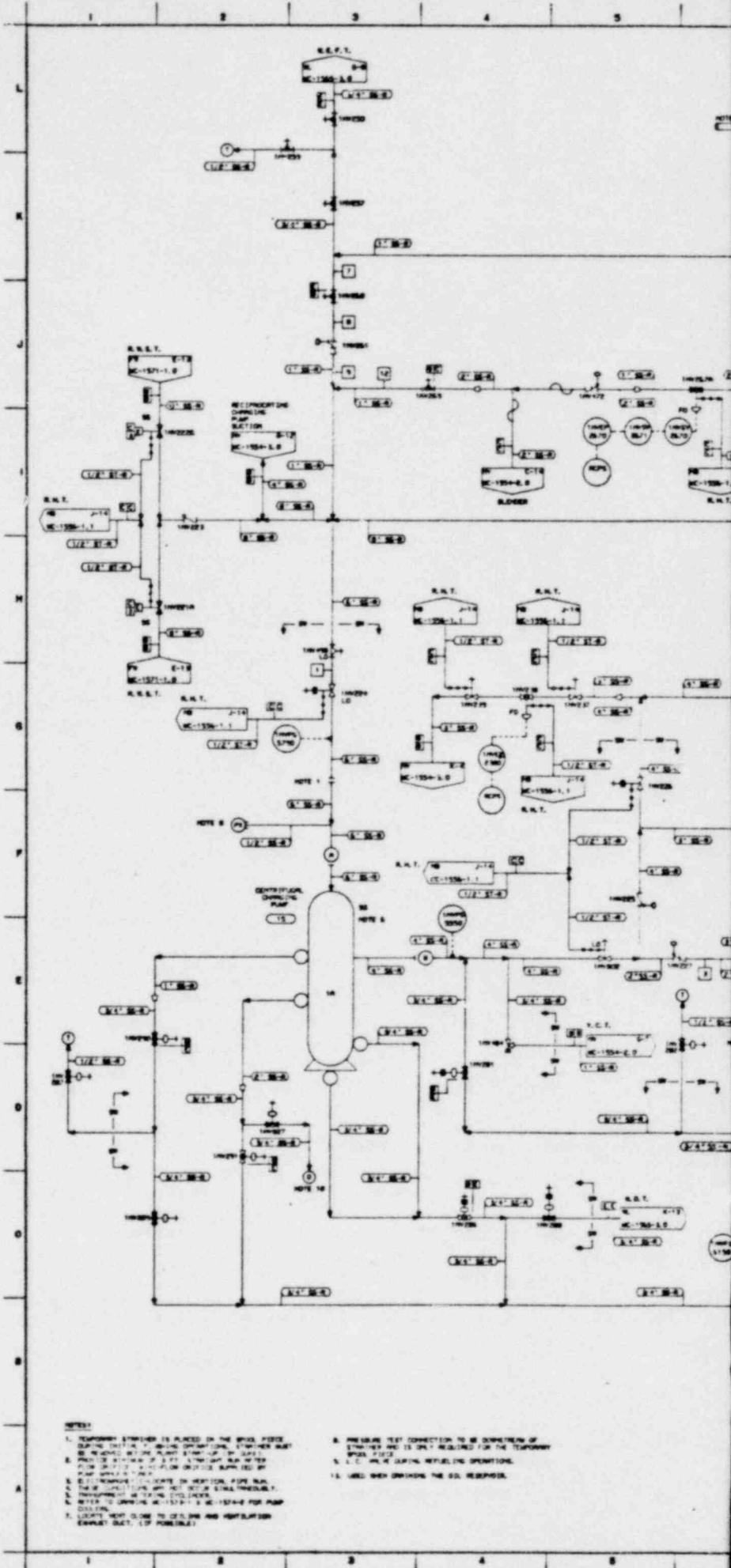
NUCLEAR SAFETY RELATED

DUKE POWER COMPANY
WALTON NUCLEAR STATION UNIT 1

PLUM TRAINING OF
PERSONNEL & VALUE CONTROL SYSTEM
(1957)

DRAWING NO. 4-1000-1.1

16-2



NOTES:

1. TURBOCOMPRESSOR IS PLACED IN THE SPECIAL POSITION DURING INITIAL STARTING OPERATIONS. STARTER MUST BE RELEASED BY THE STARTER STOP UP TO DATE.
2. PRESSURE TEST CONNECTION TO BE DISCONNECTED FROM THE TURBOCOMPRESSOR BY THE STARTER STOP UP TO DATE.
3. PRESSURE TEST CONNECTION TO BE DISCONNECTED FROM THE TURBOCOMPRESSOR BY THE STARTER STOP UP TO DATE.
4. PRESSURE TEST CONNECTION TO BE DISCONNECTED FROM THE TURBOCOMPRESSOR BY THE STARTER STOP UP TO DATE.
5. PRESSURE TEST CONNECTION TO BE DISCONNECTED FROM THE TURBOCOMPRESSOR BY THE STARTER STOP UP TO DATE.
6. PRESSURE TEST CONNECTION TO BE DISCONNECTED FROM THE TURBOCOMPRESSOR BY THE STARTER STOP UP TO DATE.
7. PRESSURE TEST CONNECTION TO BE DISCONNECTED FROM THE TURBOCOMPRESSOR BY THE STARTER STOP UP TO DATE.
8. PRESSURE TEST CONNECTION TO BE DISCONNECTED FROM THE TURBOCOMPRESSOR BY THE STARTER STOP UP TO DATE.
9. PRESSURE TEST CONNECTION TO BE DISCONNECTED FROM THE TURBOCOMPRESSOR BY THE STARTER STOP UP TO DATE.
10. PRESSURE TEST CONNECTION TO BE DISCONNECTED FROM THE TURBOCOMPRESSOR BY THE STARTER STOP UP TO DATE.
11. PRESSURE TEST CONNECTION TO BE DISCONNECTED FROM THE TURBOCOMPRESSOR BY THE STARTER STOP UP TO DATE.
12. PRESSURE TEST CONNECTION TO BE DISCONNECTED FROM THE TURBOCOMPRESSOR BY THE STARTER STOP UP TO DATE.
13. PRESSURE TEST CONNECTION TO BE DISCONNECTED FROM THE TURBOCOMPRESSOR BY THE STARTER STOP UP TO DATE.
14. PRESSURE TEST CONNECTION TO BE DISCONNECTED FROM THE TURBOCOMPRESSOR BY THE STARTER STOP UP TO DATE.
15. PRESSURE TEST CONNECTION TO BE DISCONNECTED FROM THE TURBOCOMPRESSOR BY THE STARTER STOP UP TO DATE.

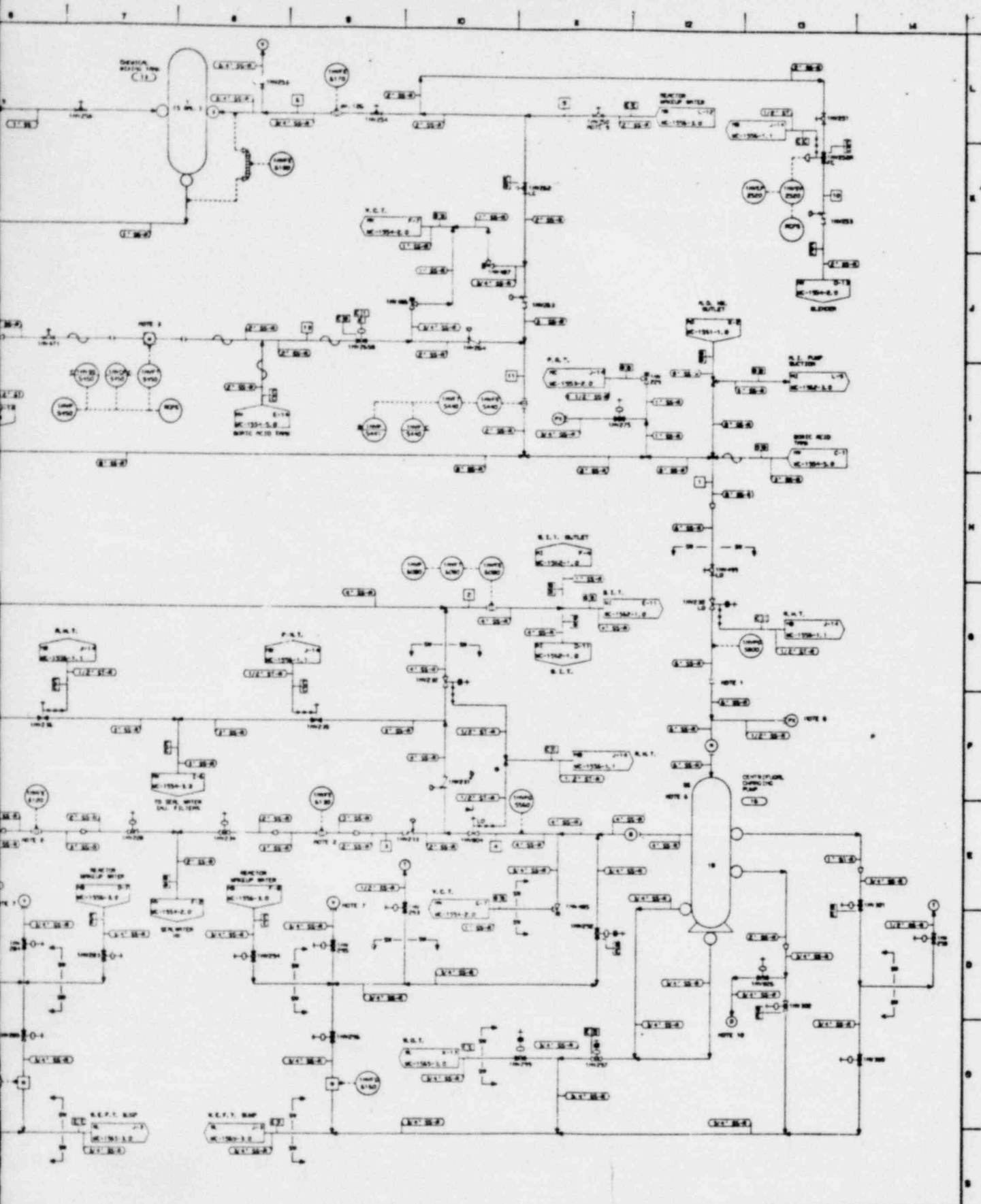
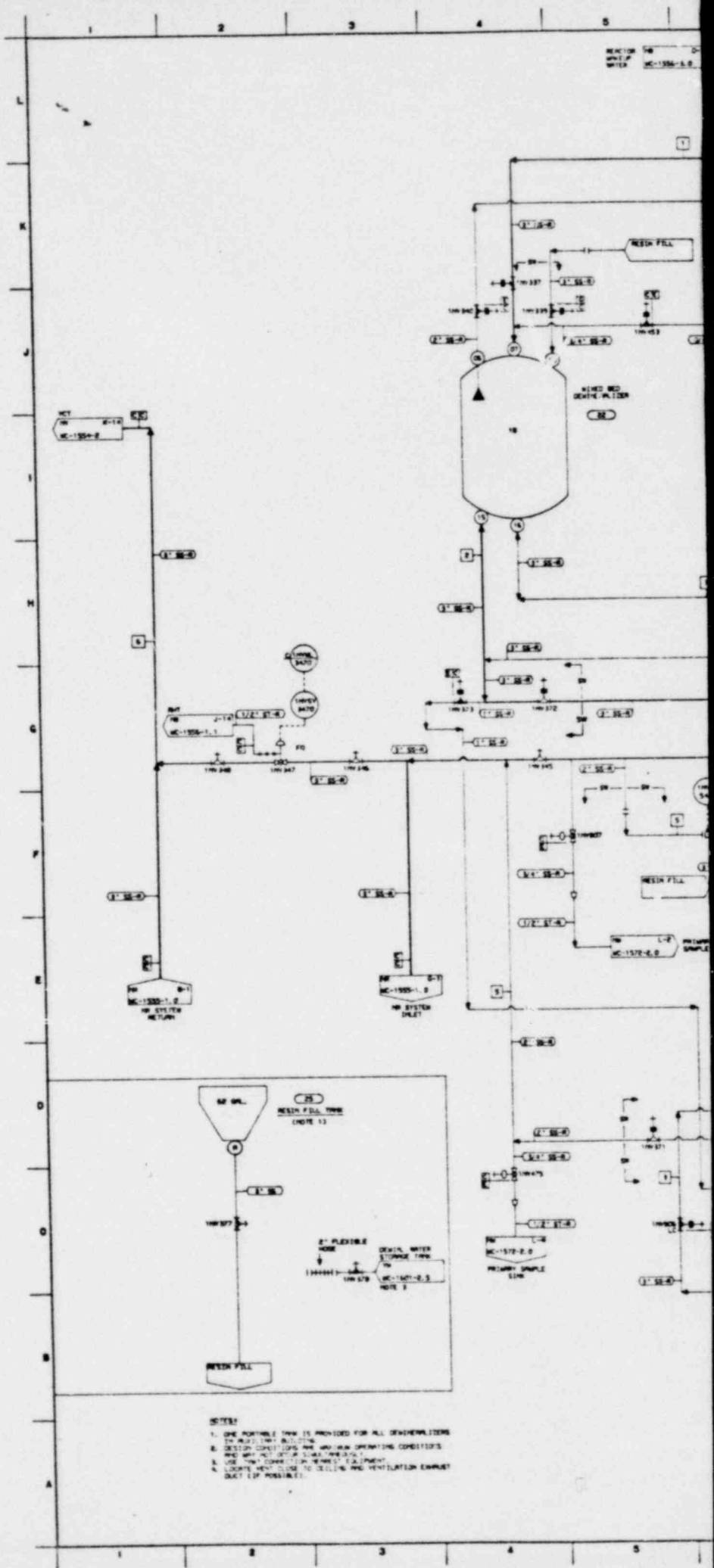


FIGURE 3.6
SHEET 6 of 7

NORMAL OPERATING CONDITIONS		DESIGN CONDITIONS (NOTE 1)	
TEMPERATURE	110°F	TEMPERATURE	110°F
PRESSURE	100 PSIA	PRESSURE	100 PSIA
FLOW RATE	100 GPM	FLOW RATE	100 GPM
PH	7.0	PH	7.0
SOLIDITY	0.01%	SOLIDITY	0.01%
CONDUCTIVITY	1000	CONDUCTIVITY	1000
DENSITY	62.4	DENSITY	62.4
VISCOSITY	1.0	VISCOSITY	1.0
HEAT CAPACITY	1.0	HEAT CAPACITY	1.0
EXPANSION COEFF	0.0002	EXPANSION COEFF	0.0002
REFRACTIVE INDEX	1.33	REFRACTIVE INDEX	1.33
ABSORPTIVITY	0.001	ABSORPTIVITY	0.001
EMISSIVITY	0.9	EMISSIVITY	0.9
REFLECTIVITY	0.1	REFLECTIVITY	0.1
TRANSMITTANCE	0.9	TRANSMITTANCE	0.9
REFRACTIVE INDEX	1.33	REFRACTIVE INDEX	1.33
ABSORPTIVITY	0.001	ABSORPTIVITY	0.001
EMISSIVITY	0.9	EMISSIVITY	0.9
REFLECTIVITY	0.1	REFLECTIVITY	0.1
TRANSMITTANCE	0.9	TRANSMITTANCE	0.9

NUCLEAR SAFETY RELATED																												
DUKE POWER COMPANY																												
WILCOX NUCLEAR STATION UNIT 1																												
PLUM CONTROL OF																												
DISCHARGE & VALVE CONTROL																												
SYSTEM UNIT																												
<table border="1" style="width: 100%;"> <tr> <th>NO.</th> <th>REVISIONS</th> <th>DATE</th> <th>BY</th> <th>CHKD.</th> </tr> <tr> <td> </td> <td> </td> <td> </td> <td> </td> <td> </td> </tr> <tr> <td> </td> <td> </td> <td> </td> <td> </td> <td> </td> </tr> <tr> <td> </td> <td> </td> <td> </td> <td> </td> <td> </td> </tr> <tr> <td> </td> <td> </td> <td> </td> <td> </td> <td> </td> </tr> </table>				NO.	REVISIONS	DATE	BY	CHKD.																				
NO.	REVISIONS	DATE	BY	CHKD.																								
DWG. NO. W-1504-1.1																												



- NOTES:**
1. ONE PORTABLE TOWER IS PROVIDED FOR ALL DEMONSTRATIONS TO BE MADE BY THE BUREAU.
 2. DESIGN CAPACITY AND AIR FLOW OPERATING CONDITIONS ARE NOT TO BE EXCEEDED.
 3. USE THE CORRECT CONNECTION POINTS.
 4. LOCKS MUST BE USED TO SECURE THE VENTILATION EXHAUST DUCT IF POSSIBLE.

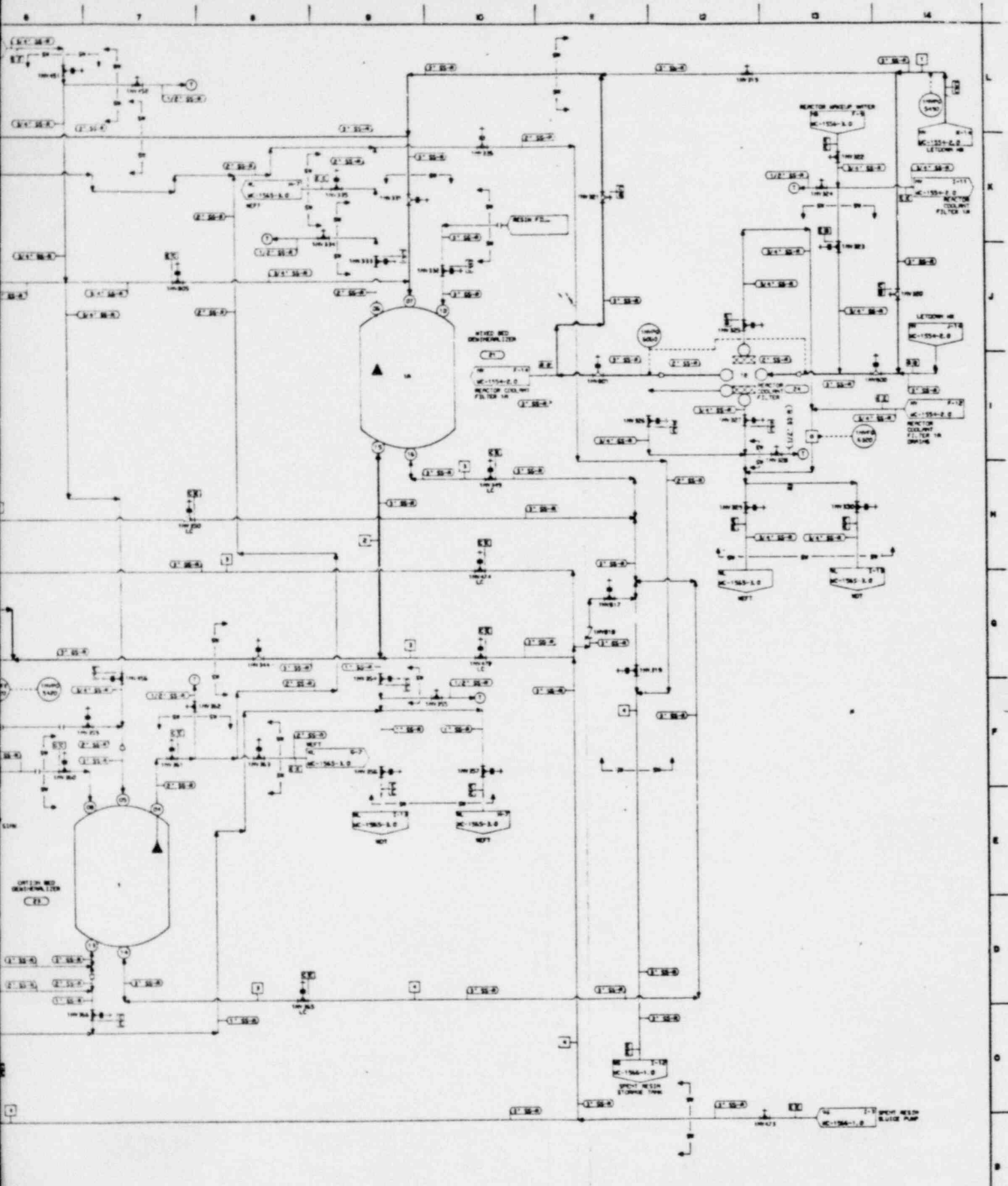


FIGURE 3.6
SHEET 7 of 7

REVISION	DATE	BY	DESCRIPTION
1	08/08/68	J. J.
2	08/08/68	J. J.
3	08/08/68	J. J.
4	08/08/68	J. J.
5	08/08/68	J. J.
6	08/08/68	J. J.
7	08/08/68	J. J.
8	08/08/68	J. J.
9	08/08/68	J. J.
10	08/08/68	J. J.
11	08/08/68	J. J.
12	08/08/68	J. J.
13	08/08/68	J. J.
14	08/08/68	J. J.
15	08/08/68	J. J.
16	08/08/68	J. J.
17	08/08/68	J. J.
18	08/08/68	J. J.
19	08/08/68	J. J.
20	08/08/68	J. J.

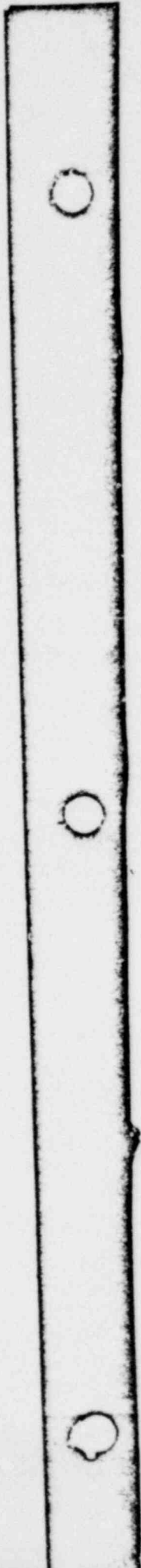
NO.	REVISIONS	DATE	BY	DESCRIPTION
1
2
3
4
5
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7
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9
10
11
12
13
14
15
16
17
18
19
20

NUCLEAR SAFETY RELATED

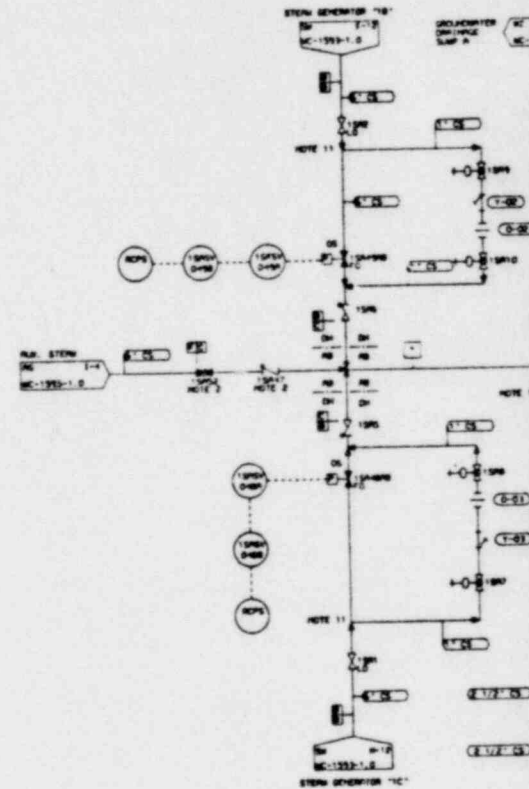
DUKE POWER COMPANY
WOLFE CRUISE REACTOR UNIT 1

FLOW DIAGRAM OF
CONTROL & VALVE CONTROL
SYSTEM (REV)

DWG. NO. MC-1554-1.0

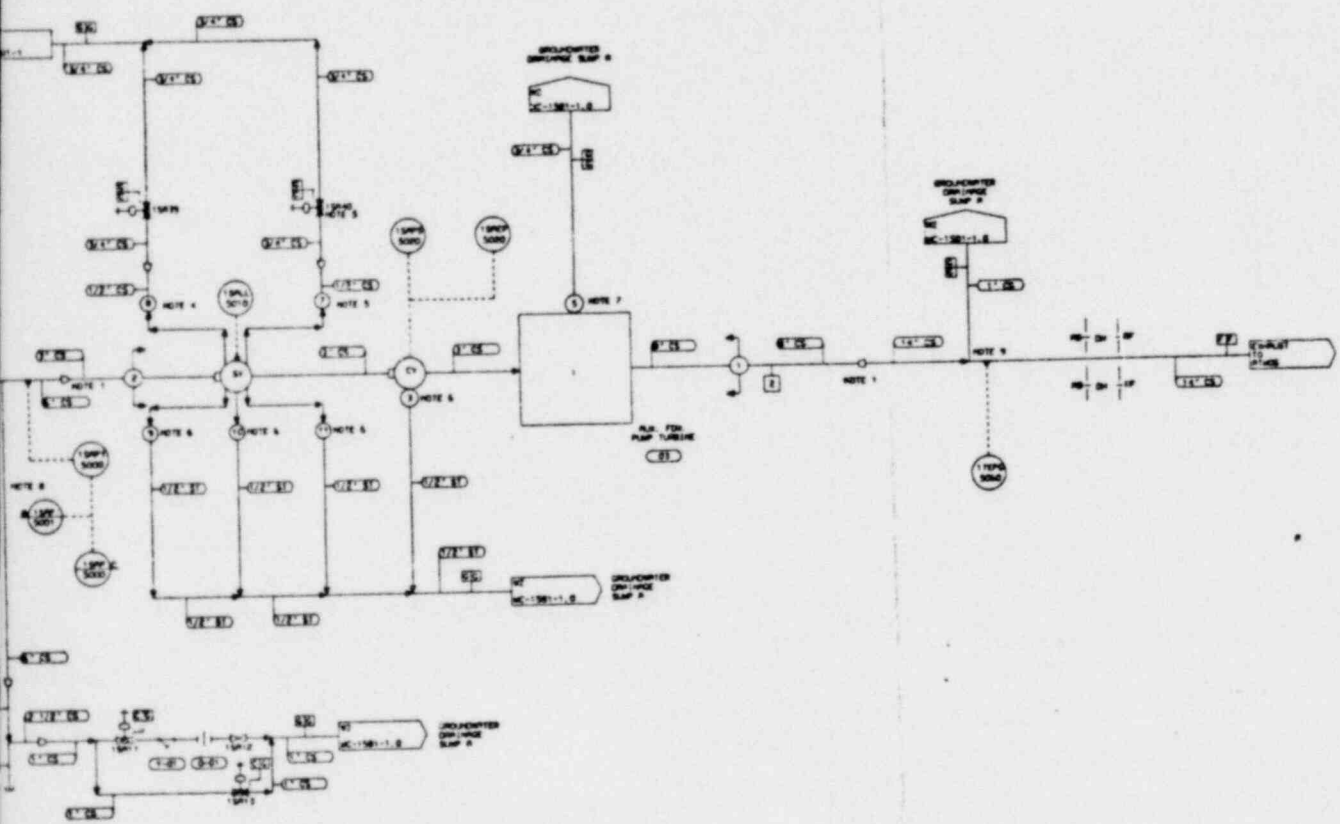


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K
J
I
H
G
F
E
D
C
B
A



- NOTES
1. LOCATE WELDER AS CLOSE TO PUMP TURNING AS POSSIBLE.
 2. LOCATE VALVES AS CLOSE TO 2 INCH STEAM LINE FROM PUMP AS POSSIBLE AND AS POSSIBLE.
 3. DETAIL LINE TO THE WELDER BEING WELDED.
 4. WELDER BEAT DOWN.
 5. WELDER BEAT DOWN.
 6. WELDER BEAT DOWN.
 7. WELDER BEAT DOWN.
 8. WELDER BEAT DOWN.
 9. WELDER BEAT DOWN.
 10. CONNECTION TO WELDER BEAT DOWN.
 11. LOW POINT DRAIN.

1 2 3 4 5 6 7 8 9 10 11 12 13 14 15 16 17 18 19 20 21 22 23 24 25 26 27 28 29 30 31 32 33 34 35 36 37 38 39 40 41 42 43 44 45 46 47 48 49 50 51 52 53 54 55 56 57 58 59 60 61 62 63 64 65 66 67 68 69 70 71 72 73 74 75 76 77 78 79 80 81 82 83 84 85 86 87 88 89 90 91 92 93 94 95 96 97 98 99 100



NORMAL OPERATING CONDITIONS
 1. 100 PSIA 1700 F
 2. 100 PSIA 1700 F

DESIGN CONDITIONS (NOTE 11)
 1. 100 PSIA 1700 F
 2. 100 PSIA 1700 F

FIGURE 3.7
SHEET 1 of 1

NO.	REVISION	DATE	BY	CHKD.	APP'D.

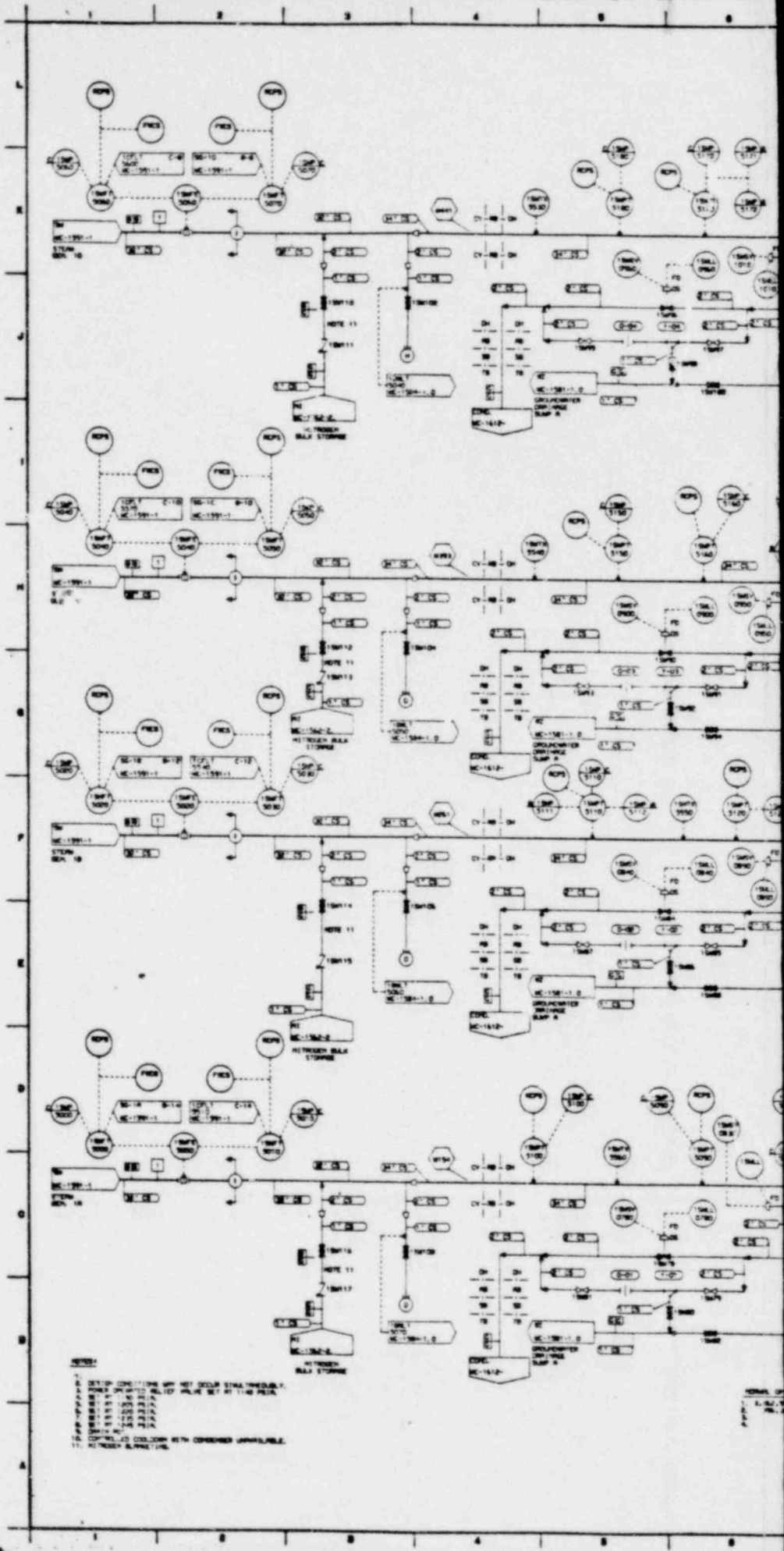
Fig. 10 8 2-1 187

NUCLEAR SAFETY RELATED

DUKE POWER COMPANY
WATKINS NUCLEAR STATION UNIT 1

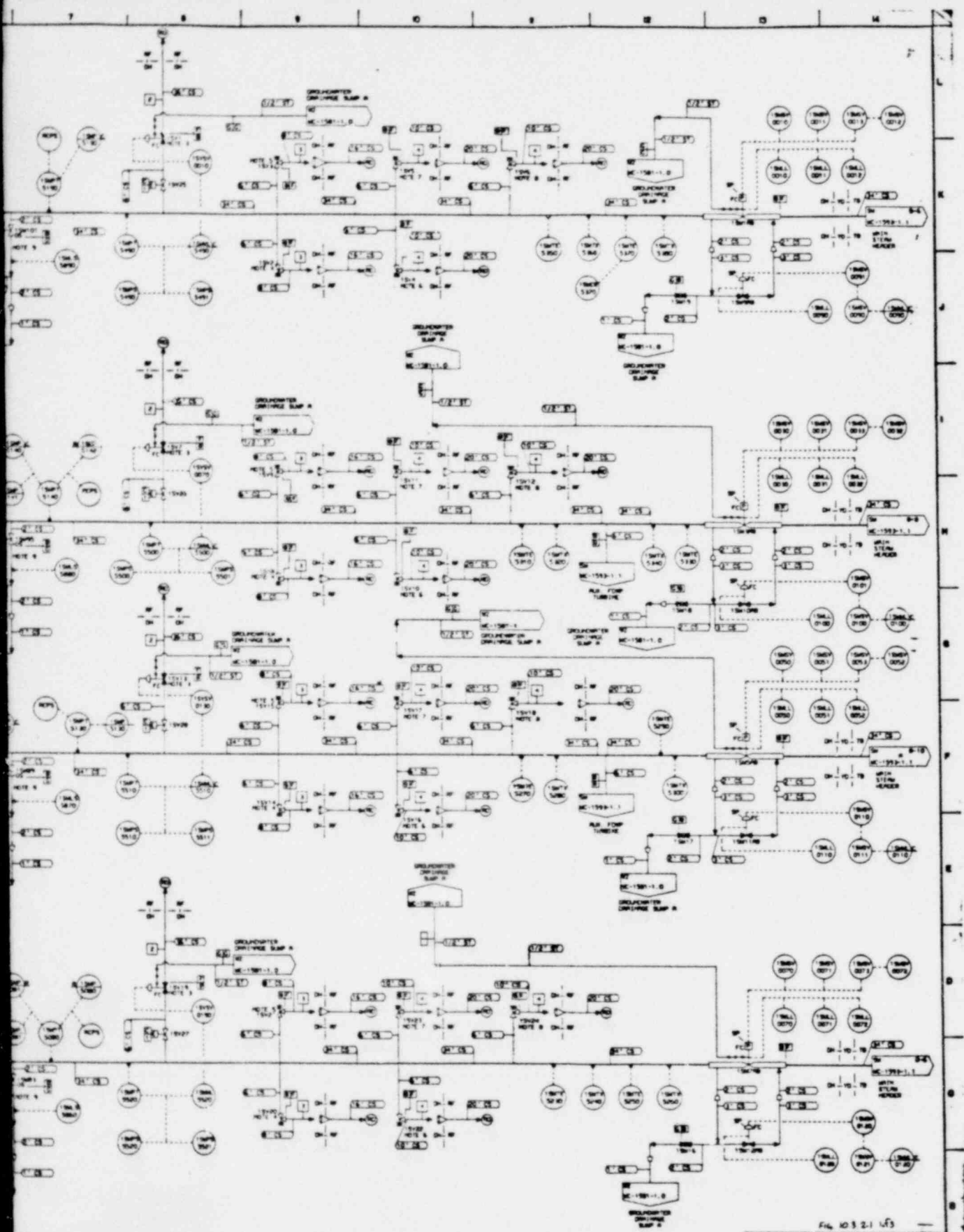
FLOW DIAGRAM OF
STEAM SUPPLY TO PLU. EQUIPMENT
SYSTEM (SFP)
TURBINE EXHAUST (TE)

(7/72) (M) 10-1582-1-2



1. SET IN POSITION AND NOT OPEN SIMULTANEOUSLY.
 2. POWER SUPPLIED ONLY BY 115V AC.
 3. SET BY 115V AC.
 4. SET BY 115V AC.
 5. SET BY 115V AC.
 6. SET BY 115V AC.
 7. SET BY 115V AC.
 8. SET BY 115V AC.
 9. SET BY 115V AC.
 10. SET BY 115V AC.
 11. SET BY 115V AC.

11. SET BY 115V AC.



NOTE: CONDITIONS
 1. 1507-1.0
 2. 1507-1.1
 3. 1507-1.2

REVISION CONDITIONS (DATE BY)
 1. 1507-1.0
 2. 1507-1.1
 3. 1507-1.2

FIGURE 3.8
 SHEET 1 of 2

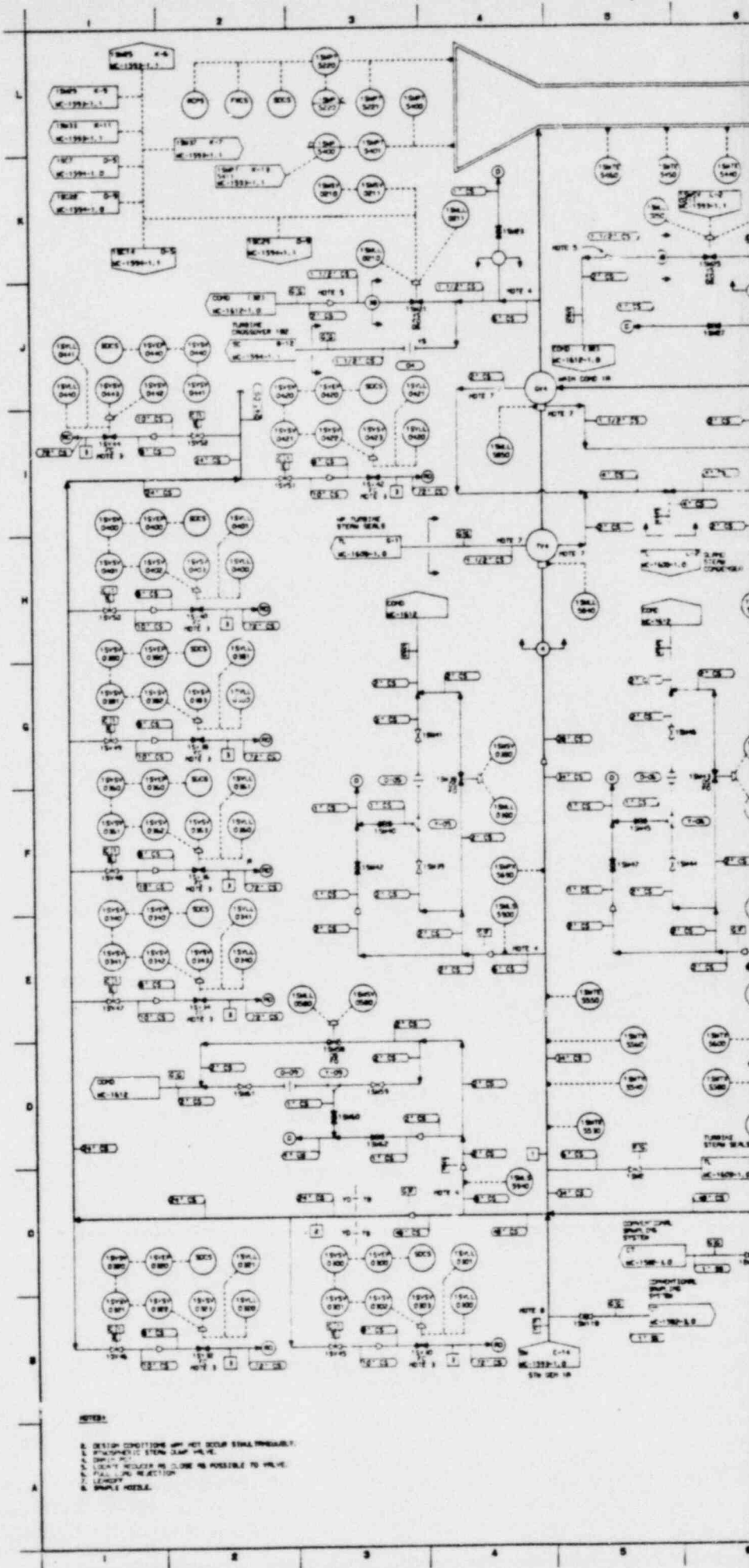
NO.	REVISIONS	DATE	BY
1			
2			
3			
4			
5			
6			
7			
8			
9			
10			

FIG. 10-3-21-143

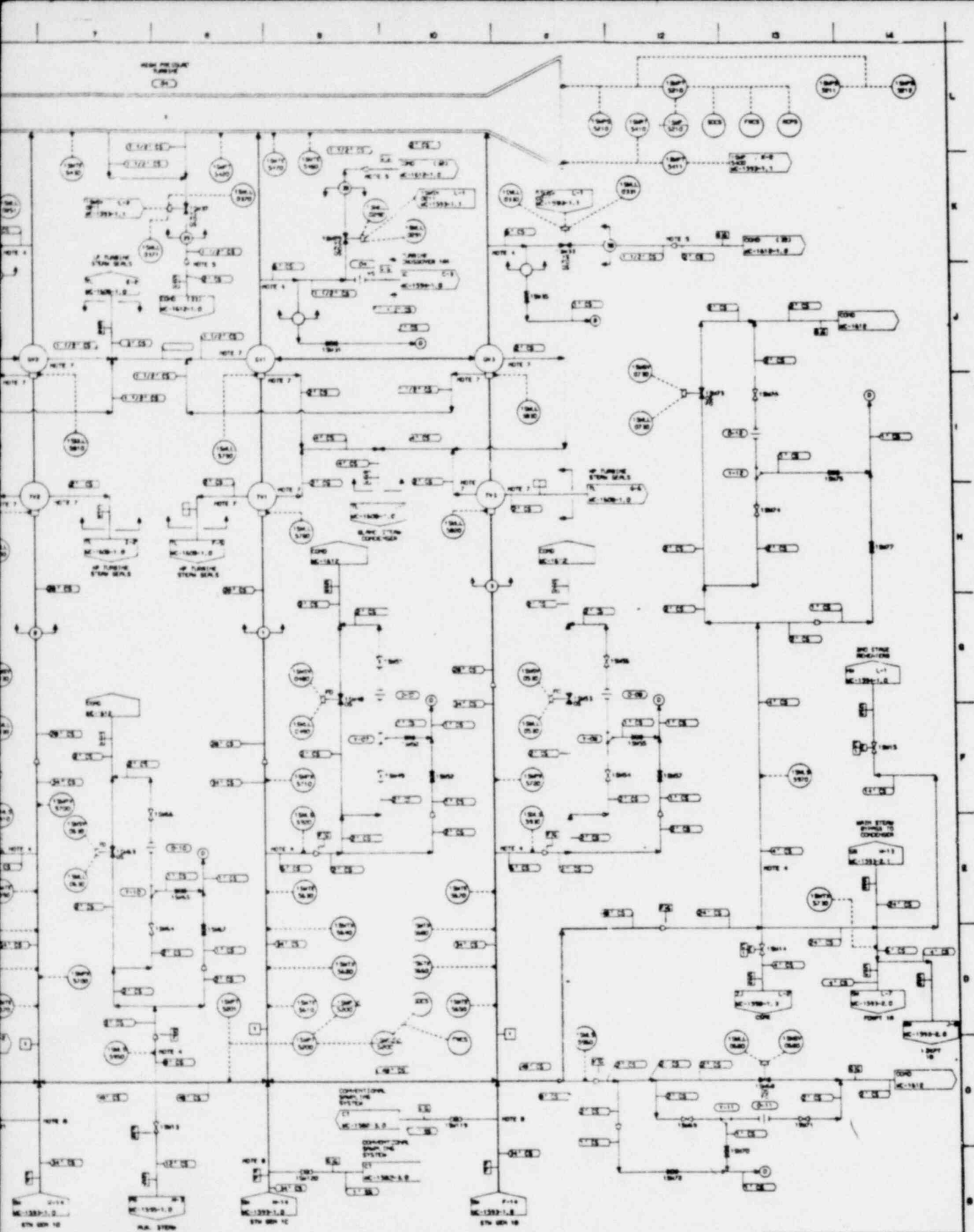
NUCLEAR SAFETY RELATED

BUES POWER COMPANY
 2500 W. HOLLAND STREET, JAY 1

PLAN NUMBER OF
 1507-1.0
 WITH STUDY NUMBER 10-3-21-143



- NOTES:**
1. DESIGN CONDITIONS MAY NOT OCCUR SIMULTANEOUSLY.
 2. PNEUMATIC STEAM JUMP PREVENTION.
 3. LOCAL RESISTOR AS CLOSE AS POSSIBLE TO VALVE.
 4. FULL-ON REACTOR.
 5. LEADOFF.
 6. SAMPLE NEEDLE.



NO.	DESCRIPTION	DATE	BY	CHKD.	APP.
1	DESIGN	11/17/54	W.S.	W.S.	W.S.
2	REVISED	11/17/54	W.S.	W.S.	W.S.
3	REVISED	11/17/54	W.S.	W.S.	W.S.
4	REVISED	11/17/54	W.S.	W.S.	W.S.
5	REVISED	11/17/54	W.S.	W.S.	W.S.
6	REVISED	11/17/54	W.S.	W.S.	W.S.
7	REVISED	11/17/54	W.S.	W.S.	W.S.
8	REVISED	11/17/54	W.S.	W.S.	W.S.
9	REVISED	11/17/54	W.S.	W.S.	W.S.
10	REVISED	11/17/54	W.S.	W.S.	W.S.
11	REVISED	11/17/54	W.S.	W.S.	W.S.
12	REVISED	11/17/54	W.S.	W.S.	W.S.
13	REVISED	11/17/54	W.S.	W.S.	W.S.
14	REVISED	11/17/54	W.S.	W.S.	W.S.

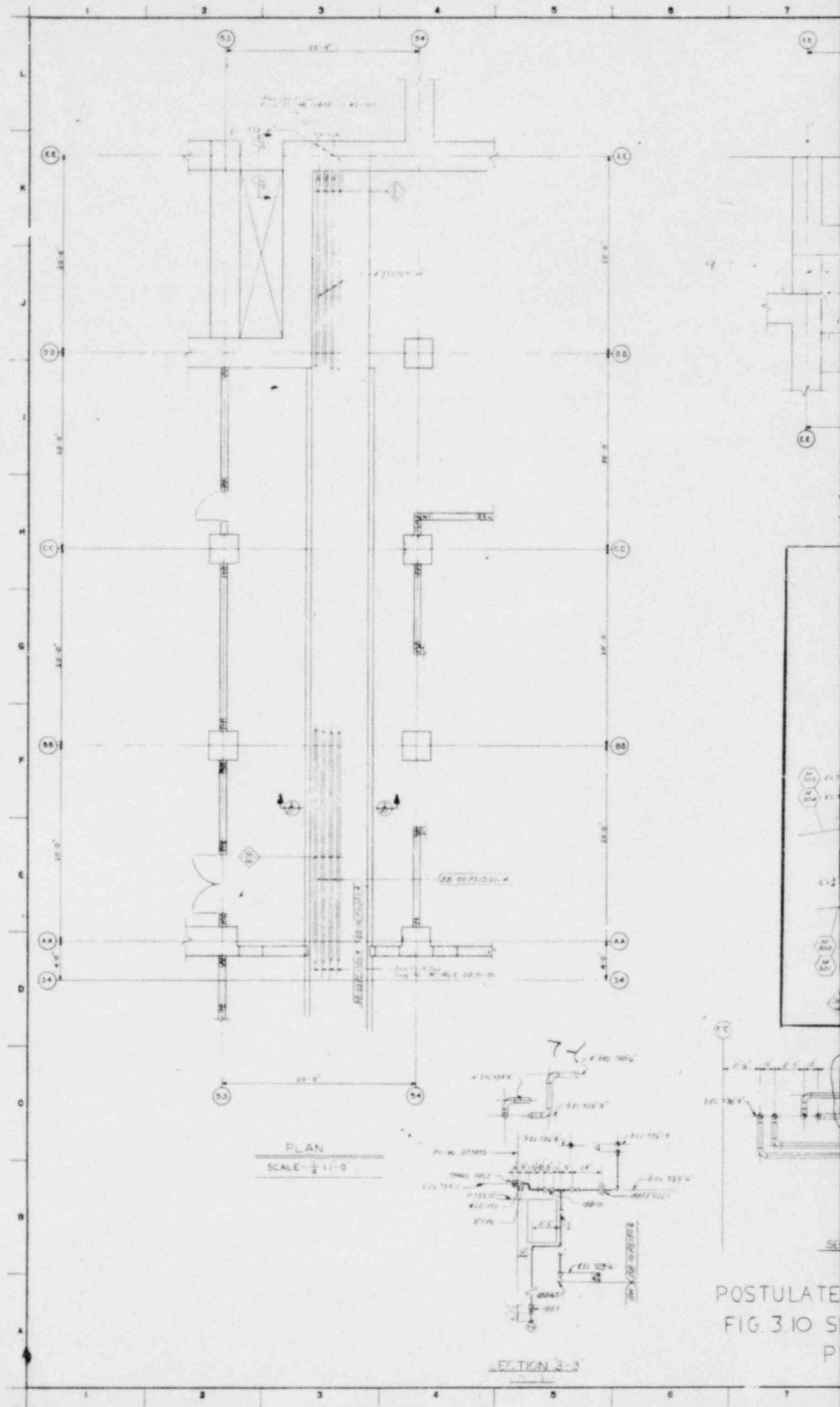
FIGURE 3.8
SHEET 2 of 2

NUCLEAR SAFETY RELATED

DUKE POWER COMPANY
WASH STEAM SYSTEM (WSS) &
WASH STEAM VENT TO ATMOSPHERE (WSVA)
PLANT DESIGN OF
WASH STEAM SYSTEM (WSS)
WASH STEAM VENT TO ATMOSPHERE (WSVA)
(27)

NO.	REVISIONS	DATE	BY	CHKD.	APP.
1	DESIGN	11/17/54	W.S.	W.S.	W.S.
2	REVISED	11/17/54	W.S.	W.S.	W.S.
3	REVISED	11/17/54	W.S.	W.S.	W.S.
4	REVISED	11/17/54	W.S.	W.S.	W.S.
5	REVISED	11/17/54	W.S.	W.S.	W.S.
6	REVISED	11/17/54	W.S.	W.S.	W.S.
7	REVISED	11/17/54	W.S.	W.S.	W.S.
8	REVISED	11/17/54	W.S.	W.S.	W.S.
9	REVISED	11/17/54	W.S.	W.S.	W.S.
10	REVISED	11/17/54	W.S.	W.S.	W.S.
11	REVISED	11/17/54	W.S.	W.S.	W.S.
12	REVISED	11/17/54	W.S.	W.S.	W.S.
13	REVISED	11/17/54	W.S.	W.S.	W.S.
14	REVISED	11/17/54	W.S.	W.S.	W.S.

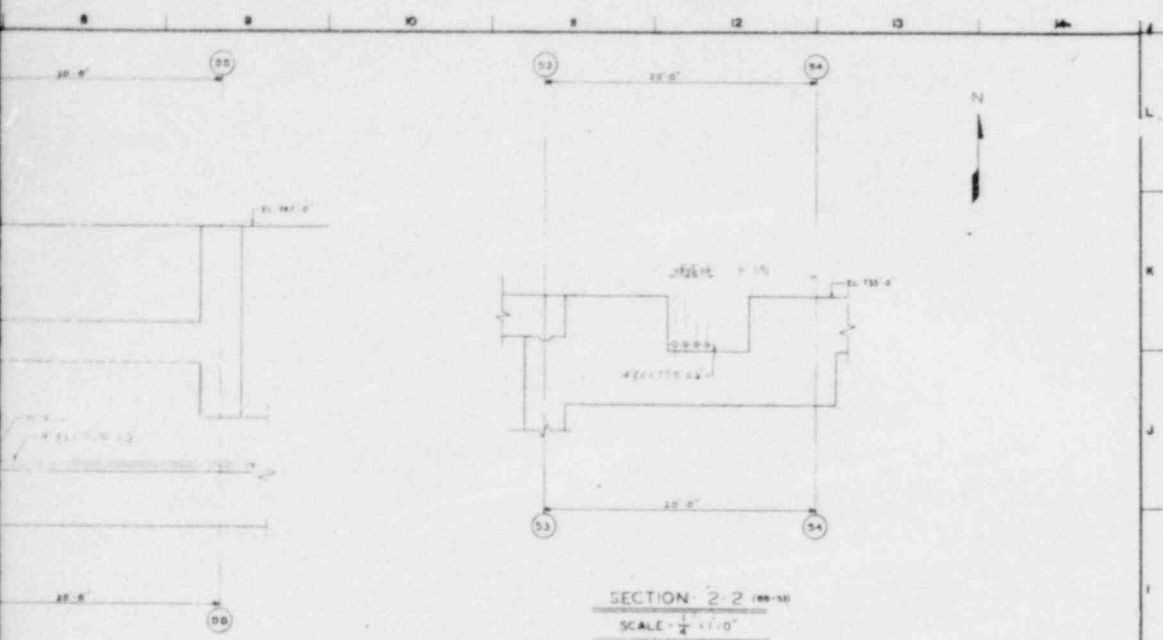
DUKE POWER COMPANY
WASH STEAM SYSTEM (WSS) &
WASH STEAM VENT TO ATMOSPHERE (WSVA)
PLANT DESIGN OF
WASH STEAM SYSTEM (WSS)
WASH STEAM VENT TO ATMOSPHERE (WSVA)
(27)



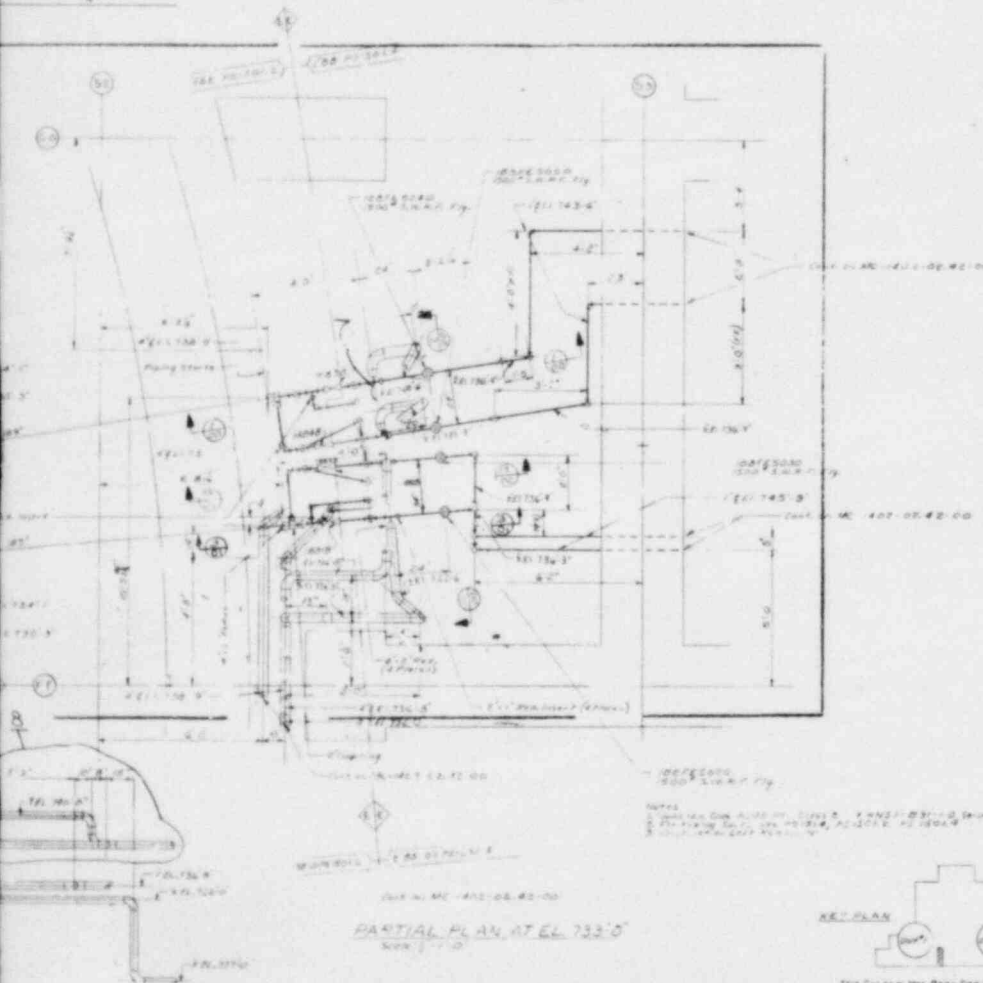
PLAN
SCALE 1/8" = 1'-0"

SECTION 3-3

POSTULATED
FIG. 3.10 S
P



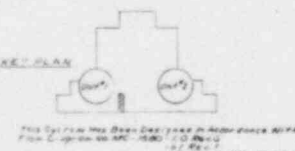
FOR BLOWUP OF THIS AREA
SEE FIG. 3.10 SHEET 3 OF 4



SECTION 4-4
SCALE - 1/4" = 1'-0"

D BREAKS - BB SYSTEM
SHEET 1 OF 4 REV. 3
WG

NO.	REVISION	DATE	BY	CHECKED
1	REVISED FOR NUCLEAR SAFETY RELATED	11/20/58	J. H. BERRY	J. H. BERRY
2	REVISED FOR NUCLEAR SAFETY RELATED	11/20/58	J. H. BERRY	J. H. BERRY
3	REVISED FOR NUCLEAR SAFETY RELATED	11/20/58	J. H. BERRY	J. H. BERRY
4	REVISED FOR NUCLEAR SAFETY RELATED	11/20/58	J. H. BERRY	J. H. BERRY
5	REVISED FOR NUCLEAR SAFETY RELATED	11/20/58	J. H. BERRY	J. H. BERRY
6	REVISED FOR NUCLEAR SAFETY RELATED	11/20/58	J. H. BERRY	J. H. BERRY
7	REVISED FOR NUCLEAR SAFETY RELATED	11/20/58	J. H. BERRY	J. H. BERRY
8	REVISED FOR NUCLEAR SAFETY RELATED	11/20/58	J. H. BERRY	J. H. BERRY
9	REVISED FOR NUCLEAR SAFETY RELATED	11/20/58	J. H. BERRY	J. H. BERRY
10	REVISED FOR NUCLEAR SAFETY RELATED	11/20/58	J. H. BERRY	J. H. BERRY

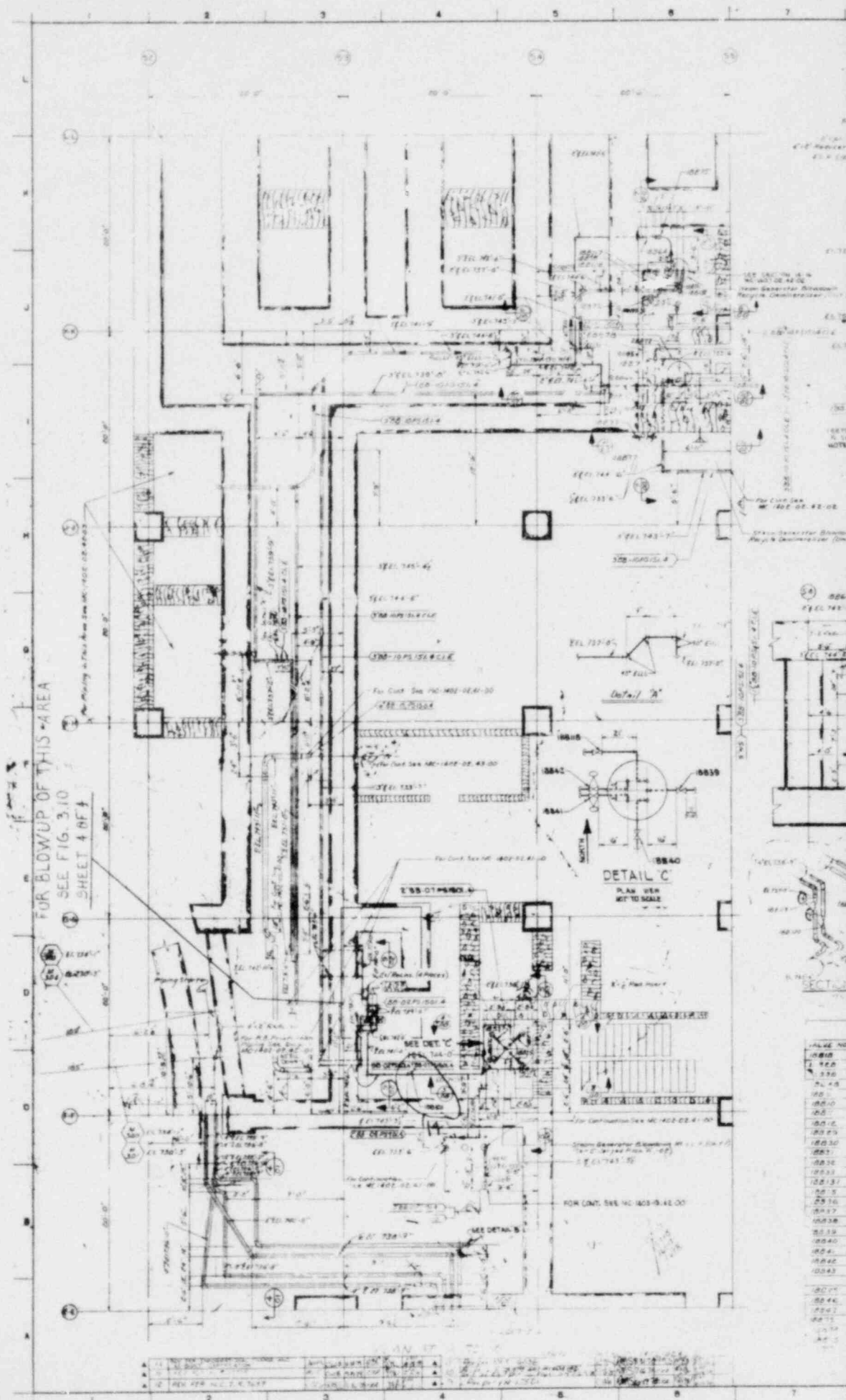


NUCLEAR SAFETY RELATED

DUKE POWER COMPANY
FIGURE 1 - NUCLEAR STATION UNIT 1

PIPING LAYOUT
SYSTEM-BB
PLAN & SECTIONS AT ELEV. 733'-0"
AUXILIARY BUILDING

NO. 1020242018



FOR BLOWUP OF THIS AREA
SEE FIG. 3.10
SHEET 4 OF 4

DETAIL C
PLAN VIEW
1/4\"/>

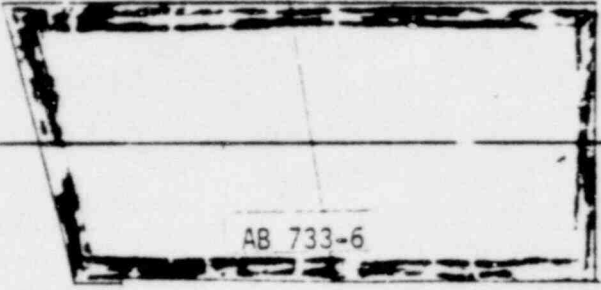
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00.35	8.00
00.40	9.00
00.45	10.00
00.50	11.00
00.55	12.00
01.00	13.00
01.05	14.00
01.10	15.00
01.15	16.00
01.20	17.00
01.25	18.00
01.30	19.00
01.35	20.00
01.40	21.00
01.45	22.00
01.50	23.00
01.55	24.00
02.00	25.00

NO.	DESCRIPTION	DATE	BY
1	REV. FOR NO. 1, 2, 3, 4, 5, 6, 7, 8, 9, 10, 11, 12, 13, 14, 15, 16, 17, 18, 19, 20, 21, 22, 23, 24, 25		
2	REV. FOR NO. 1, 2, 3, 4, 5, 6, 7, 8, 9, 10, 11, 12, 13, 14, 15, 16, 17, 18, 19, 20, 21, 22, 23, 24, 25		
3	REV. FOR NO. 1, 2, 3, 4, 5, 6, 7, 8, 9, 10, 11, 12, 13, 14, 15, 16, 17, 18, 19, 20, 21, 22, 23, 24, 25		
4	REV. FOR NO. 1, 2, 3, 4, 5, 6, 7, 8, 9, 10, 11, 12, 13, 14, 15, 16, 17, 18, 19, 20, 21, 22, 23, 24, 25		

52

BBE-PS1501.2

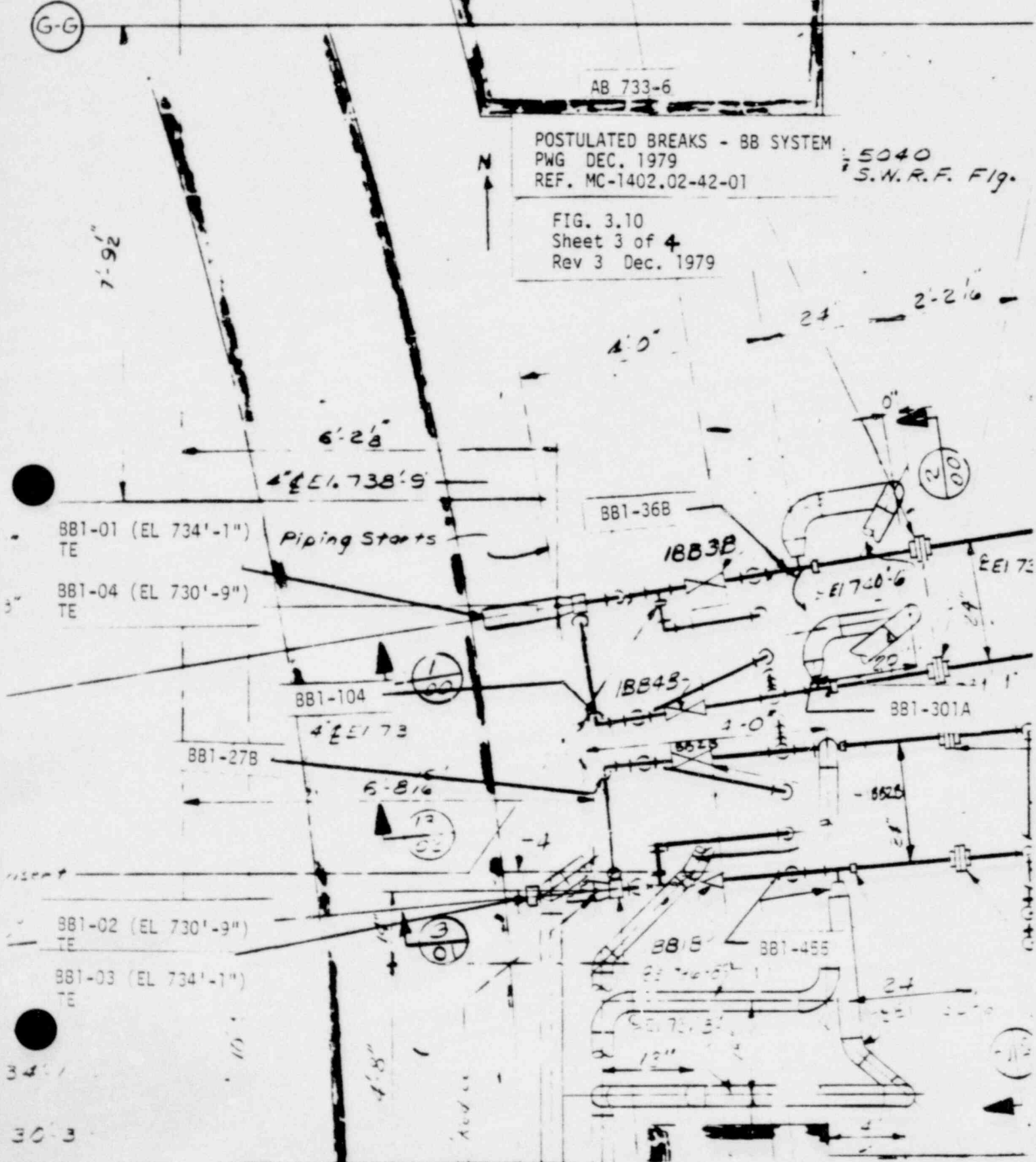
BBB-PS1501.4



POSTULATED BREAKS - BB SYSTEM
 PWG DEC. 1979
 REF. MC-1402.02-42-01

5040
 S.N.R.F. Fig.

FIG. 3.10
 Sheet 3 of 4
 Rev 3 Dec. 1979



POSTULATED BREAKS IN BB SYSTEM
PWG DEC. 1979

For Cont. See MC-1402-02.43-00

AB 733-13
REF. MC-1402-02.42-00

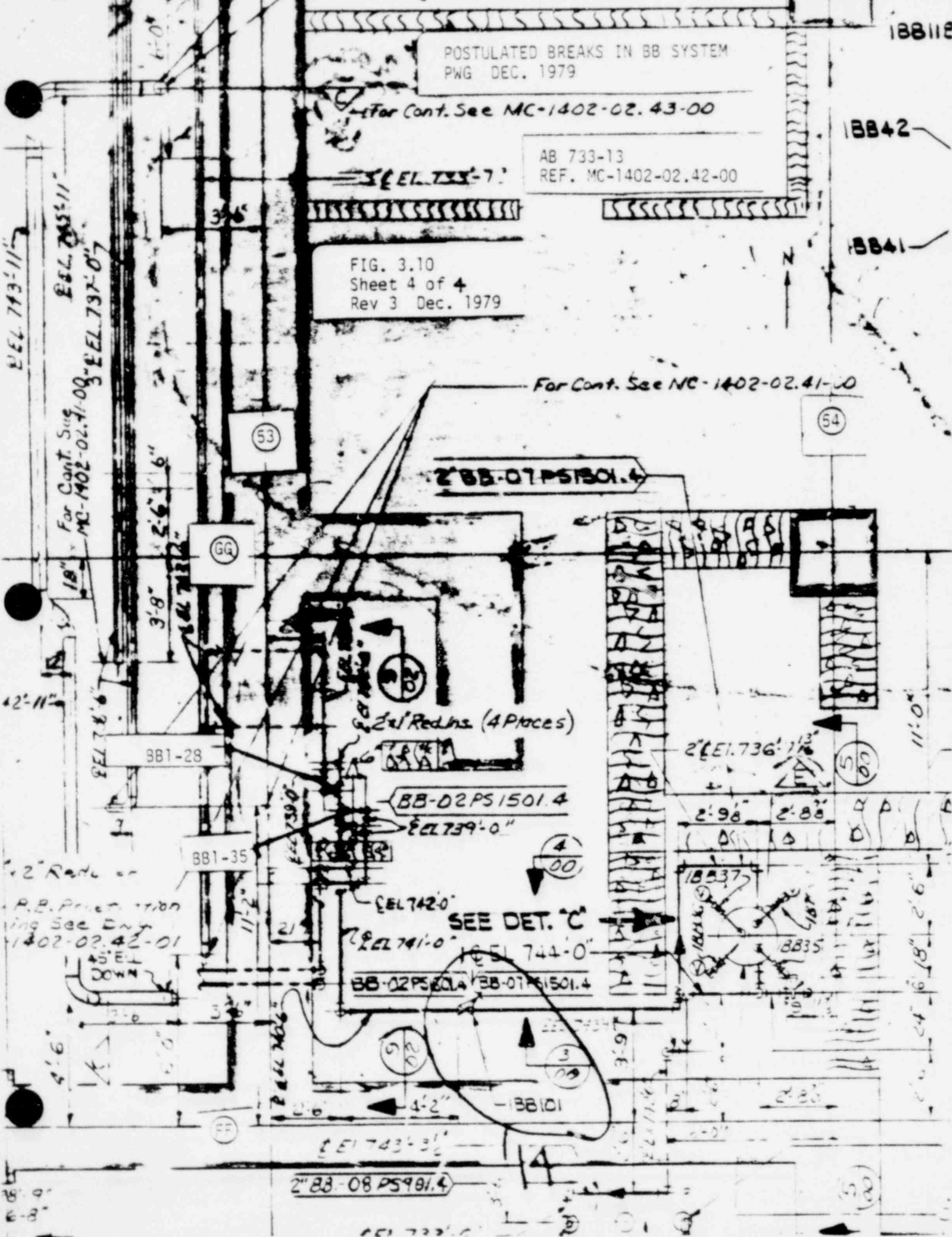
FIG. 3.10
Sheet 4 of 4
Rev 3 Dec. 1979

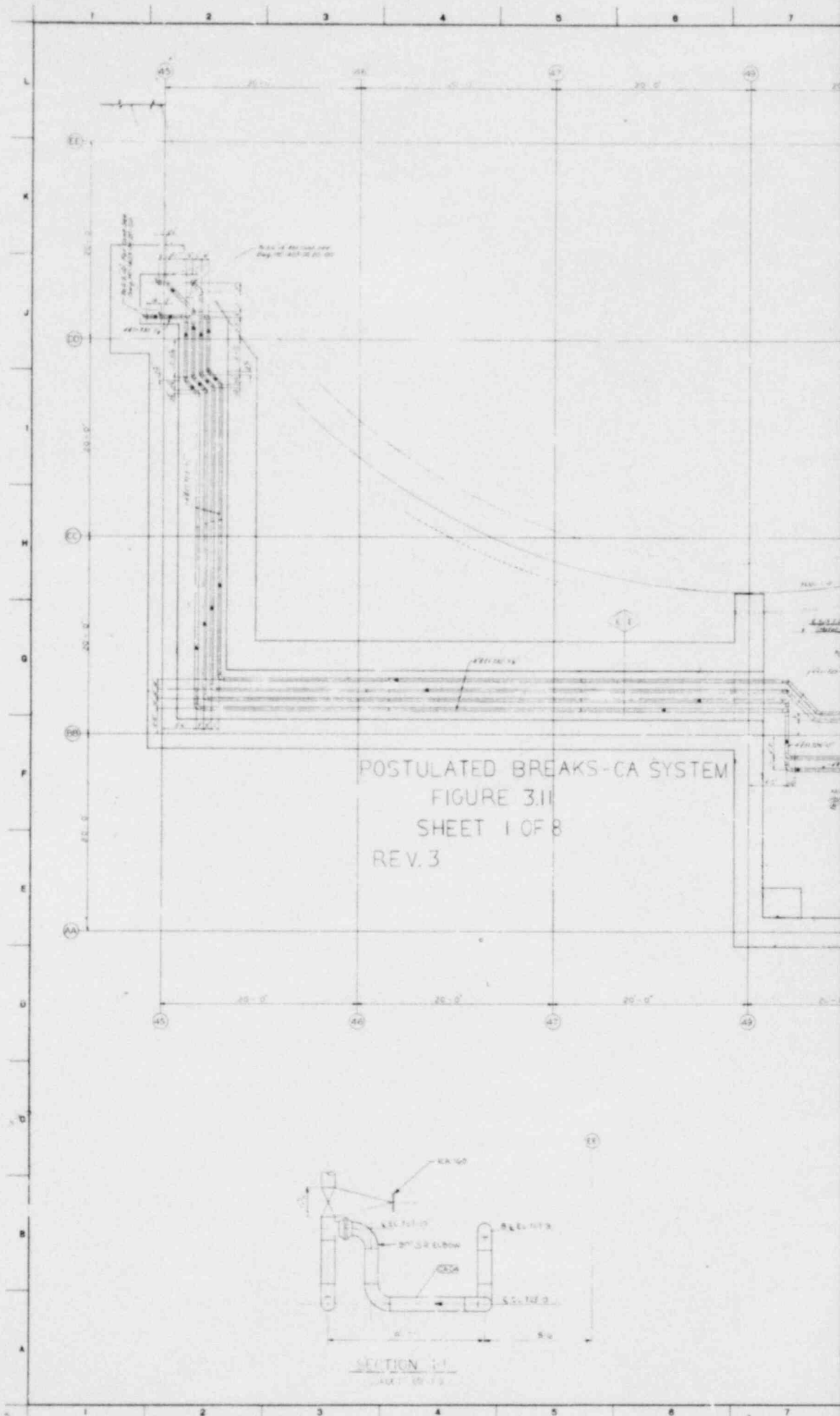
For Cont. See MC-1402-02.41-00

18B11E

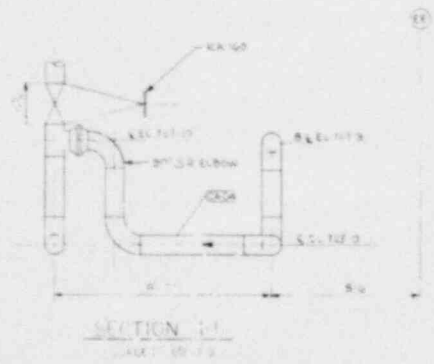
18B42

18B41



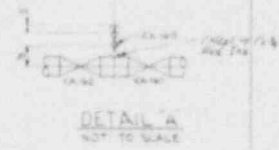


POSTULATED BREAKS-CA SYSTEM
 FIGURE 3.11
 SHEET 1 OF 8
 REV. 3

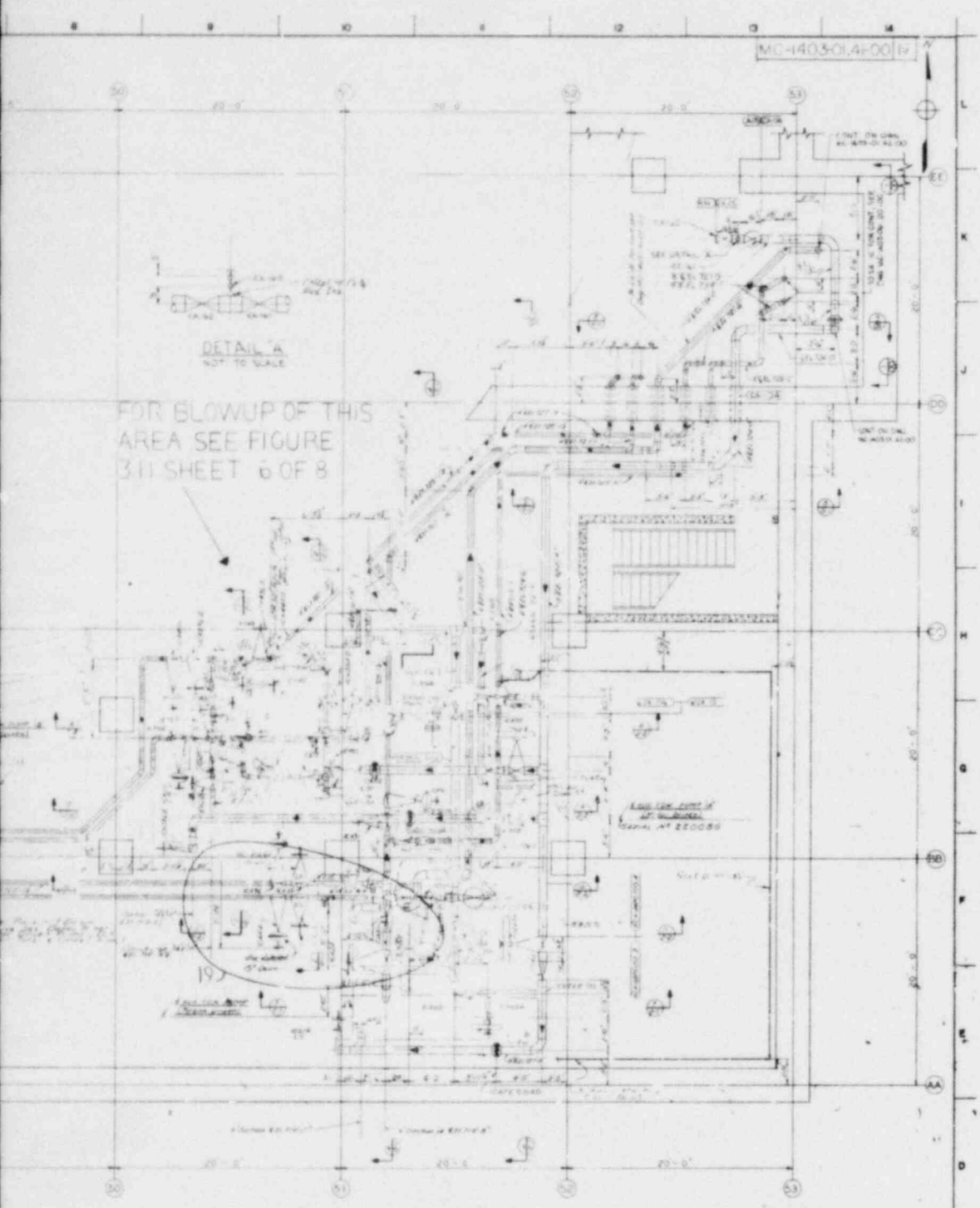


SECTION 1-1
 K&L 701-10

MC-1403-01.4-001P



FOR BLOWUP OF THIS AREA SEE FIGURE 3.11 SHEET 6 OF 8



1. All piping shall be installed in accordance with the applicable codes and standards.

2. All piping shall be installed in accordance with the applicable codes and standards.

3. All piping shall be installed in accordance with the applicable codes and standards.

DESIGNED BY: [Name] CHECKED BY: [Name] DATE: [Date]

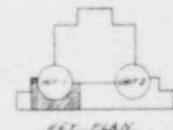
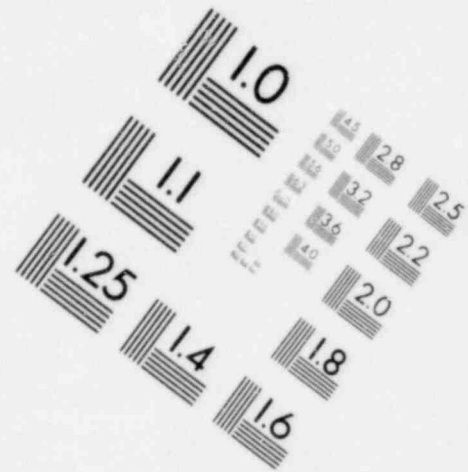
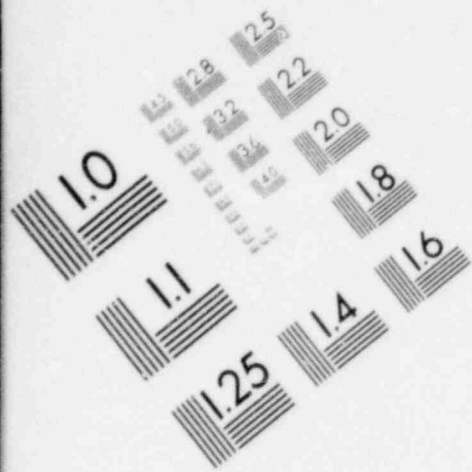


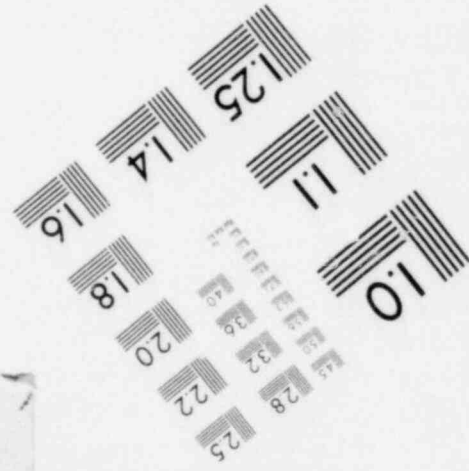
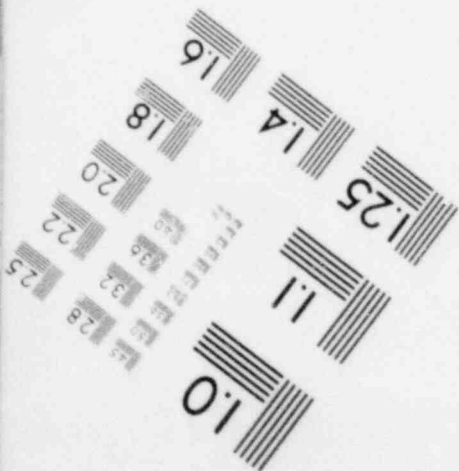
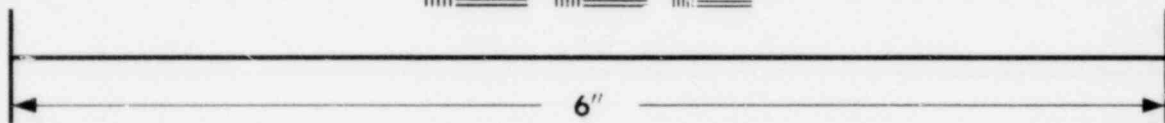
Table with 4 columns: No., Description, Date, and Status. Contains a list of items and their tracking information.

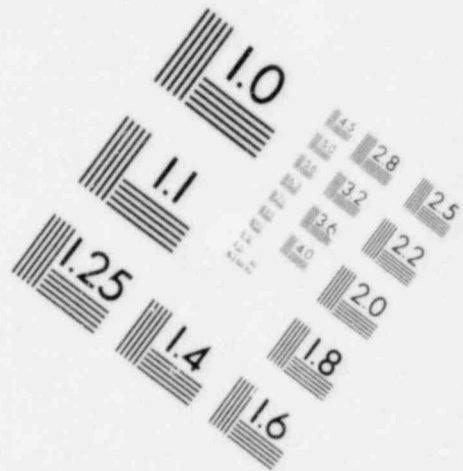
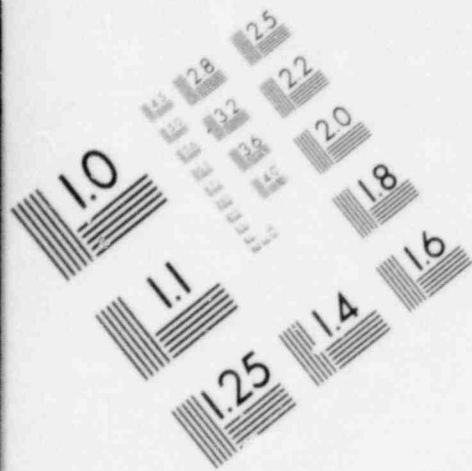
Table with 4 columns: No., Description, Date, and Status. Contains a list of items and their tracking information.

DUKE POWER COMPANY
FACILITY: [Name]
PIPING LAYOUT
SYSTEM - CA
PLAN AT ELEV 74'-0"
AUXILIARY BUILDING
DATE: [Date] DRAWING NO: MC-1403-01.4-001

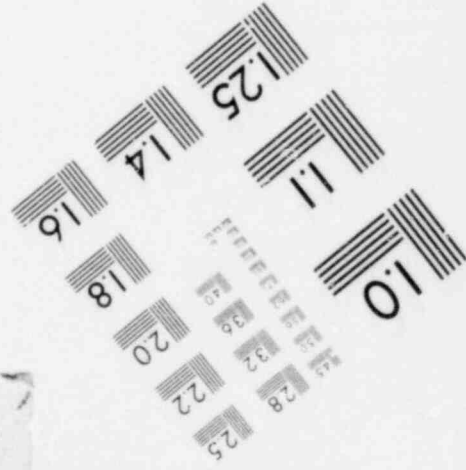
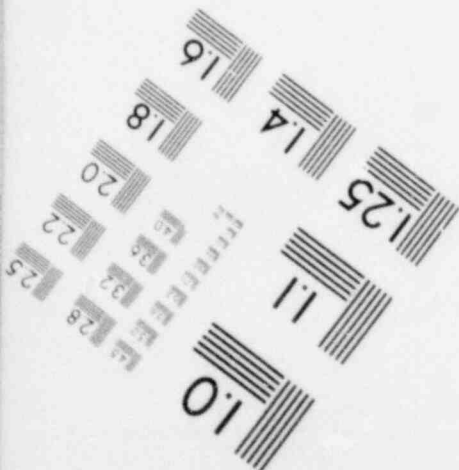
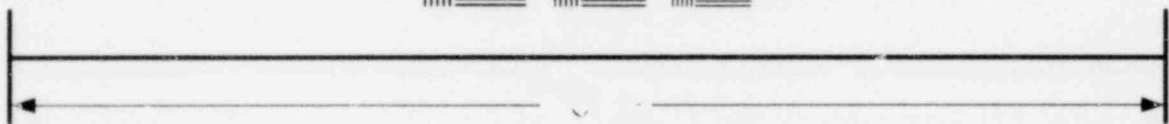


**IMAGE EVALUATION
TEST TARGET (MT-3)**

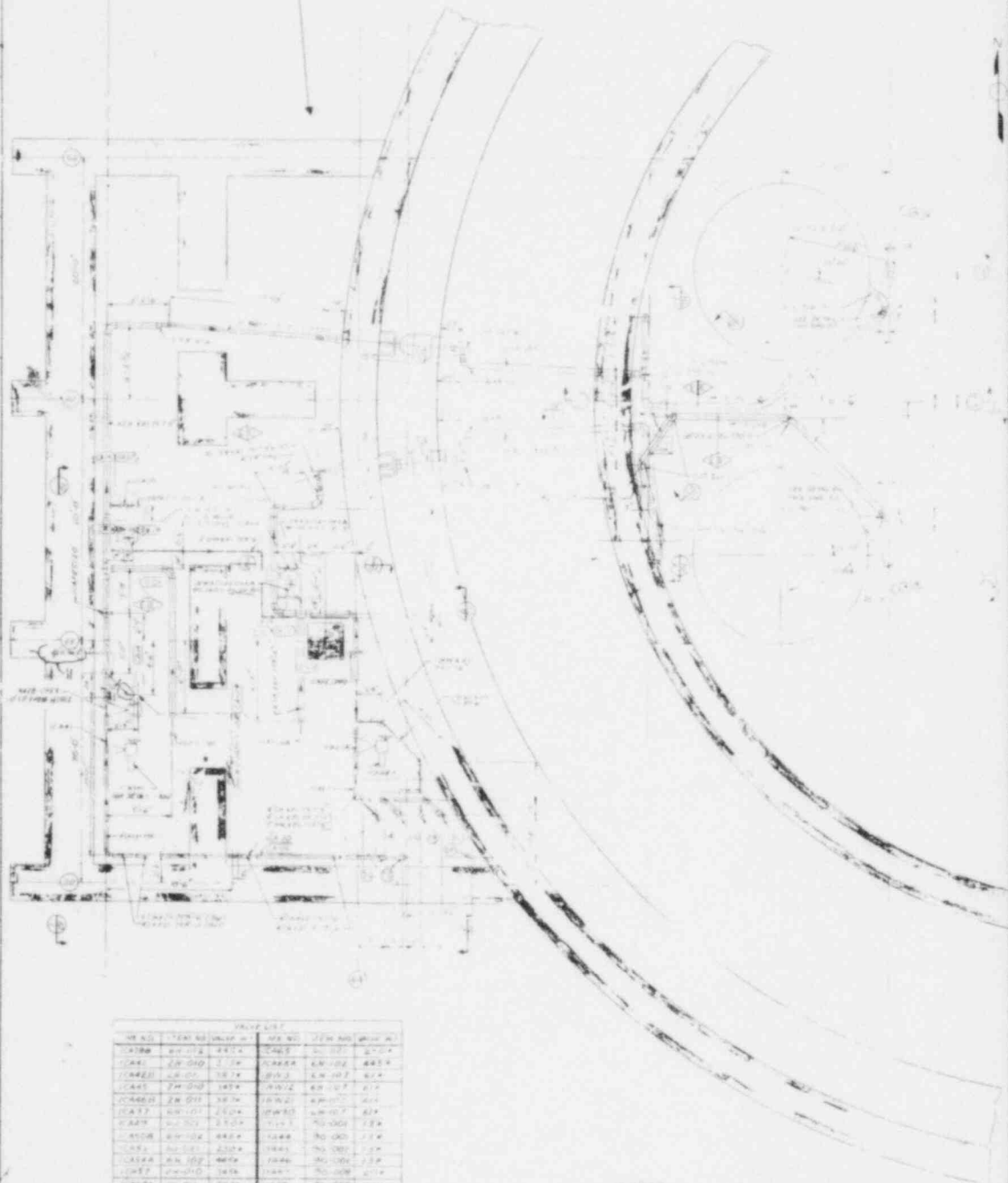




**IMAGE EVALUATION
TEST TARGET (MT-3)**



FOR BLOWUP OF THIS AREA
SEE FIG. 3.11, SHEET 8 OF 8

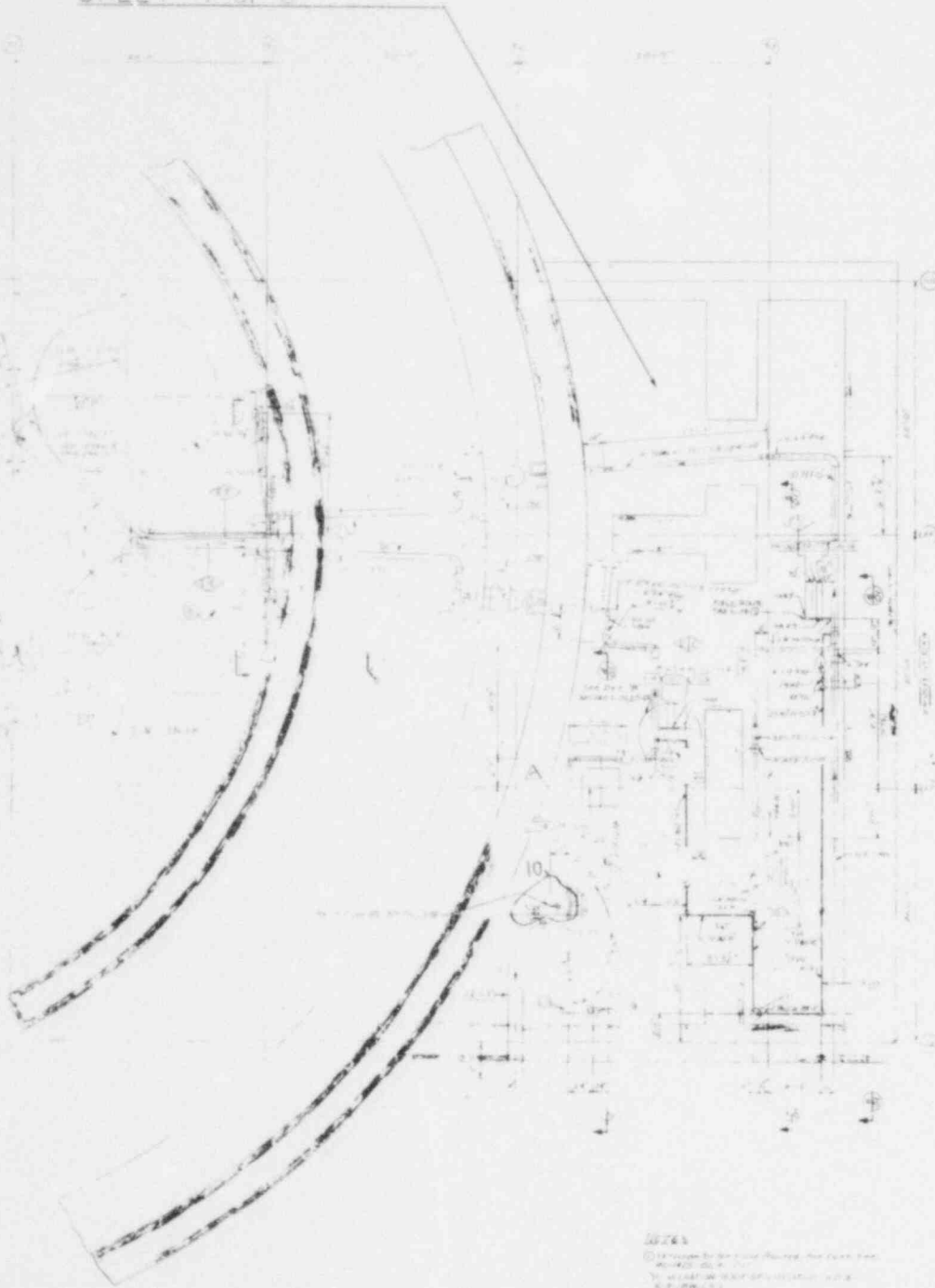


NO.	TEAM NUMBER	NO.	TEAM NUMBER
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1002	1002	1002	1002
1003	1003	1003	1003
1004	1004	1004	1004
1005	1005	1005	1005
1006	1006	1006	1006
1007	1007	1007	1007
1008	1008	1008	1008
1009	1009	1009	1009
1010	1010	1010	1010
1011	1011	1011	1011
1012	1012	1012	1012
1013	1013	1013	1013
1014	1014	1014	1014
1015	1015	1015	1015
1016	1016	1016	1016
1017	1017	1017	1017
1018	1018	1018	1018
1019	1019	1019	1019
1020	1020	1020	1020

POSTUL
FIG. 3.1
SHEET

FOR BLOWUP OF THIS AREA SEE FIG 311

SHEET 7 OF 8



NOTES
 1. THIS DRAWING IS THE PROPERTY OF THE DUKE POWER COMPANY AND IS NOT TO BE REPRODUCED OR COPIED IN ANY MANNER WITHOUT THE WRITTEN PERMISSION OF THE DUKE POWER COMPANY.
 2. THIS DRAWING IS THE PROPERTY OF THE DUKE POWER COMPANY AND IS NOT TO BE REPRODUCED OR COPIED IN ANY MANNER WITHOUT THE WRITTEN PERMISSION OF THE DUKE POWER COMPANY.
 3. THIS DRAWING IS THE PROPERTY OF THE DUKE POWER COMPANY AND IS NOT TO BE REPRODUCED OR COPIED IN ANY MANNER WITHOUT THE WRITTEN PERMISSION OF THE DUKE POWER COMPANY.
 4. THIS DRAWING IS THE PROPERTY OF THE DUKE POWER COMPANY AND IS NOT TO BE REPRODUCED OR COPIED IN ANY MANNER WITHOUT THE WRITTEN PERMISSION OF THE DUKE POWER COMPANY.
 5. THIS DRAWING IS THE PROPERTY OF THE DUKE POWER COMPANY AND IS NOT TO BE REPRODUCED OR COPIED IN ANY MANNER WITHOUT THE WRITTEN PERMISSION OF THE DUKE POWER COMPANY.
 6. THIS DRAWING IS THE PROPERTY OF THE DUKE POWER COMPANY AND IS NOT TO BE REPRODUCED OR COPIED IN ANY MANNER WITHOUT THE WRITTEN PERMISSION OF THE DUKE POWER COMPANY.
 7. THIS DRAWING IS THE PROPERTY OF THE DUKE POWER COMPANY AND IS NOT TO BE REPRODUCED OR COPIED IN ANY MANNER WITHOUT THE WRITTEN PERMISSION OF THE DUKE POWER COMPANY.
 8. THIS DRAWING IS THE PROPERTY OF THE DUKE POWER COMPANY AND IS NOT TO BE REPRODUCED OR COPIED IN ANY MANNER WITHOUT THE WRITTEN PERMISSION OF THE DUKE POWER COMPANY.
 9. THIS DRAWING IS THE PROPERTY OF THE DUKE POWER COMPANY AND IS NOT TO BE REPRODUCED OR COPIED IN ANY MANNER WITHOUT THE WRITTEN PERMISSION OF THE DUKE POWER COMPANY.
 10. THIS DRAWING IS THE PROPERTY OF THE DUKE POWER COMPANY AND IS NOT TO BE REPRODUCED OR COPIED IN ANY MANNER WITHOUT THE WRITTEN PERMISSION OF THE DUKE POWER COMPANY.
 11. THIS DRAWING IS THE PROPERTY OF THE DUKE POWER COMPANY AND IS NOT TO BE REPRODUCED OR COPIED IN ANY MANNER WITHOUT THE WRITTEN PERMISSION OF THE DUKE POWER COMPANY.
 12. THIS DRAWING IS THE PROPERTY OF THE DUKE POWER COMPANY AND IS NOT TO BE REPRODUCED OR COPIED IN ANY MANNER WITHOUT THE WRITTEN PERMISSION OF THE DUKE POWER COMPANY.
 13. THIS DRAWING IS THE PROPERTY OF THE DUKE POWER COMPANY AND IS NOT TO BE REPRODUCED OR COPIED IN ANY MANNER WITHOUT THE WRITTEN PERMISSION OF THE DUKE POWER COMPANY.
 14. THIS DRAWING IS THE PROPERTY OF THE DUKE POWER COMPANY AND IS NOT TO BE REPRODUCED OR COPIED IN ANY MANNER WITHOUT THE WRITTEN PERMISSION OF THE DUKE POWER COMPANY.

ATED BREAKS - CA SYSTEM
 REV.3 PWG
 2 OF 8

NO.	REVISIONS	CHG APPR DATE	CHK APPR DATE	INSPECTOR
1	AS SHOWN			
2	REVISED			
3	REVISED			
4	REVISED			
5	REVISED			
6	REVISED			
7	REVISED			
8	REVISED			
9	REVISED			
10	REVISED			
11	REVISED			
12	REVISED			
13	REVISED			
14	REVISED			

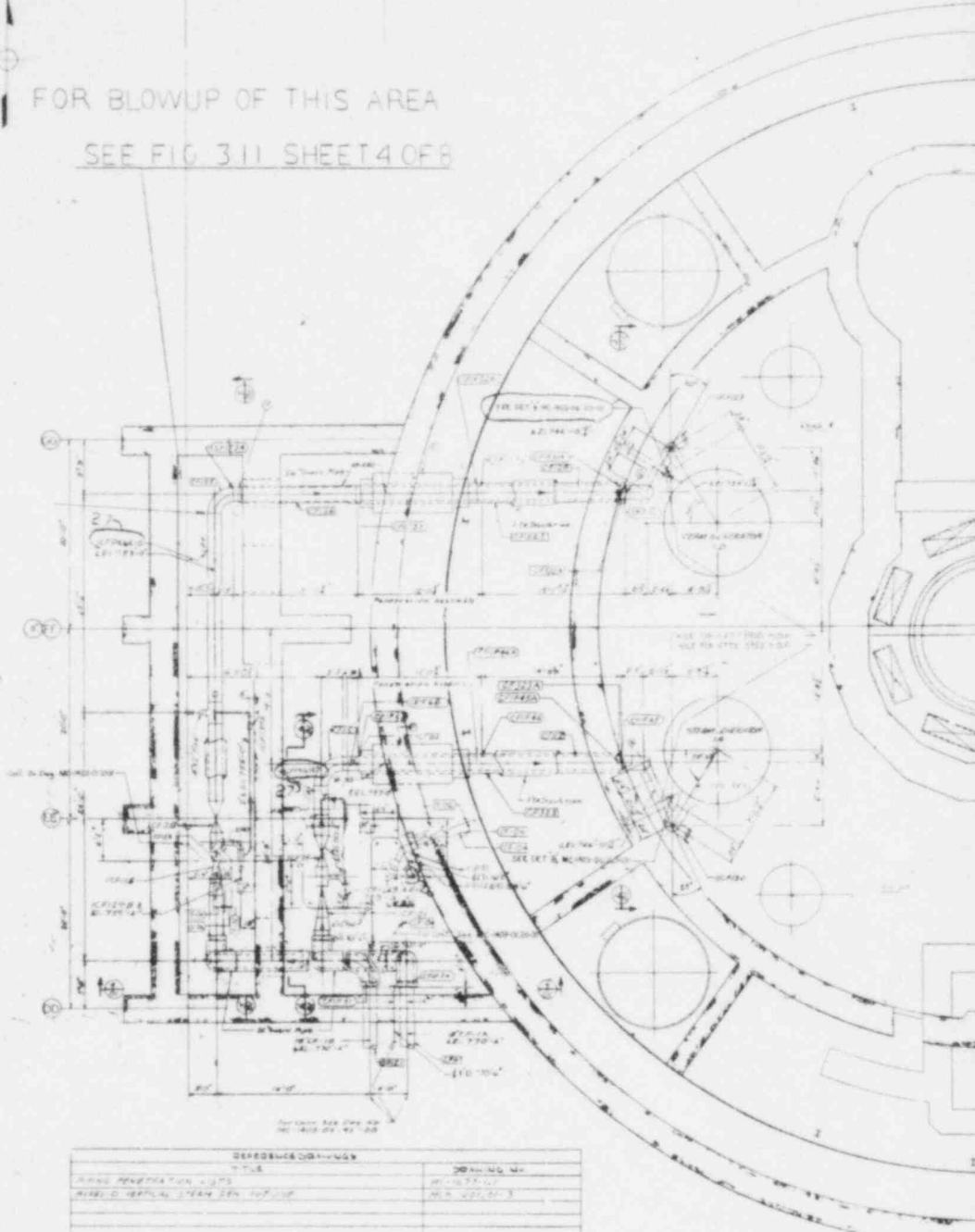
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3	3		
4	4		
5	5		
6	6		
7	7		
8	8		
9	9		
10	10		
11	11		
12	12		
13	13		
14	14		

THIS DRAWING HAS BEEN DESIGNED IN ACCORDANCE WITH
 FLOW DIAGRAM REV. NO. DATE CHANGE ORDER NO.
 15-0

NUCLEAR SAFETY RELATED
 DUKE POWER COMPANY
 MCGUIRE NUCLEAR STATION-UNIT 1
 PIPING LAYOUT
 CA DIV VA & TC SYSTEM
 PLAN
 REACTOR BUILDING

DESIGNED BY DATE CHK DATE
 DRAWN BY DATE APP DATE
 SCALE
 DWG NO: MG-1403-01, 20-00

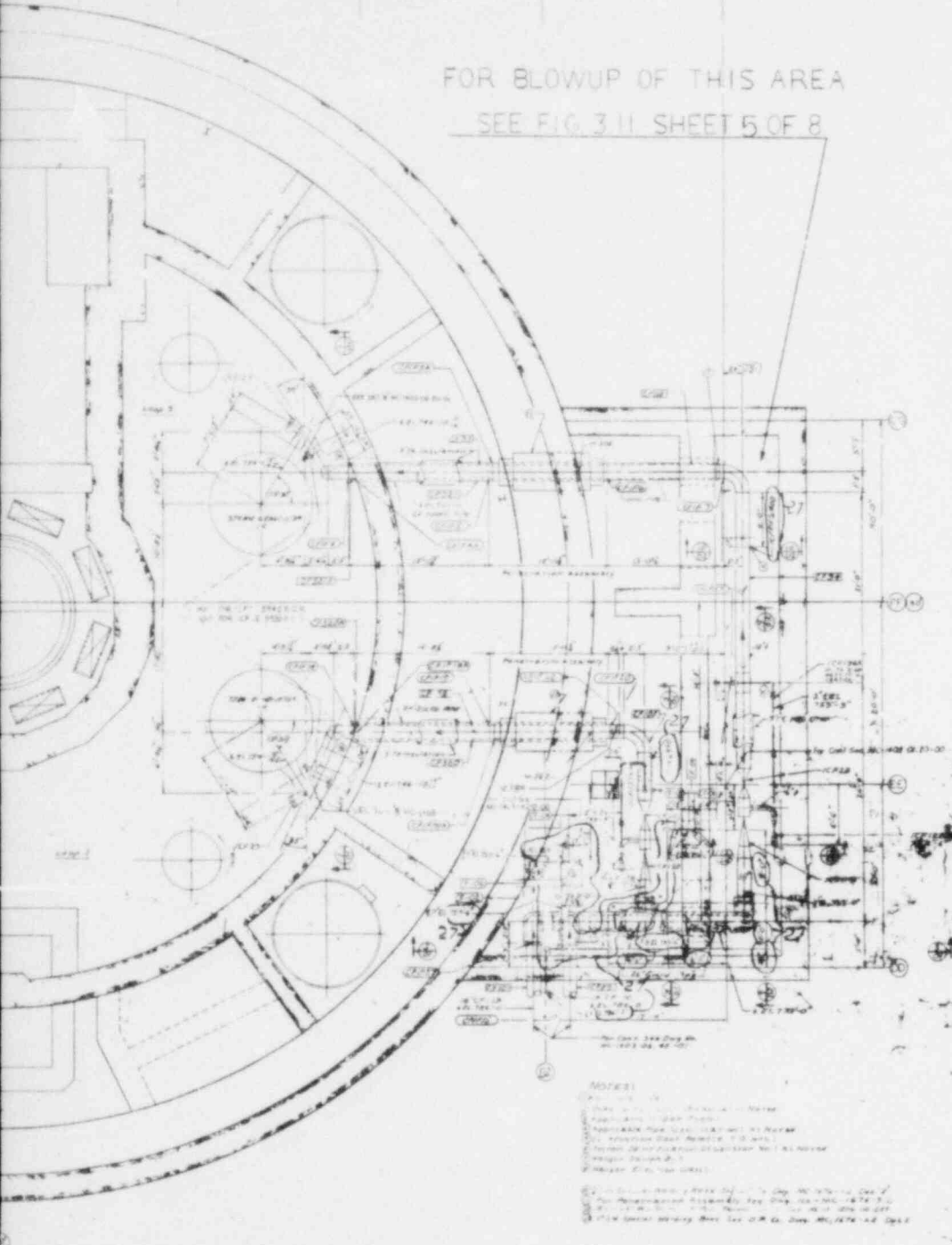
FOR BLOWUP OF THIS AREA
SEE FIG 3.11 SHEET 4 OF 8



REFERENCE DIMENSIONS							
DIMENSION				DIMENSION			
NO.	TYPE	DESCRIPTION	VALUE	NO.	TYPE	DESCRIPTION	VALUE
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02	10004	02	10004
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46	10004	46	10004
47	10004	47	10004
48	10004	48	10004
49	10004	49	10004
50	10004	50	10004

POSTULATED BREACH
FIG. 3.11 REV 3 S

FOR BLOWUP OF THIS AREA
SEE FIG. 3.11 SHEET 5 OF 8



- NOTES:
1. See Fig. 3.11, Sheet 5 of 8 for details of the system.
 2. The system is designed to operate at a pressure of 150 psig.
 3. The system is designed to operate at a temperature of 200°F.
 4. The system is designed to operate at a flow rate of 100 gpm.
 5. The system is designed to operate at a head of 100 feet.
 6. The system is designed to operate at a power of 100 kW.
 7. The system is designed to operate at a efficiency of 10%.
 8. The system is designed to operate at a reliability of 99.9%.
 9. The system is designed to operate at a safety factor of 1.5.
 10. The system is designed to operate at a design life of 10 years.

KS - CA SYSTEM
SHEET 3 OF 8

Fig. 10.4.7-5

NO.	REVISIONS	DATE	BY	CHKD	APP'D
1					
2					
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7					
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NUCLEAR SAFETY RELATED

POWER SYSTEM

MIGURE NUCLEAR STATION - UNIT 1

SPRING LAYOUT

FEEDWATER SYSTEM

PLAN -

REACTOR BUILDING

MC 40-06-20-00

20'-0"

27
ICFP16A10
EEL 753'-0"

CFI26

CFI23



POSTULATED BREAKS - CA SYSTEM
AB 750-26 JULY 1980
REF. MC-1403.06-20-00

6 FF

CA1-150
TE

Penetration As

14'-0"

2'-5 1/2"

12'-11 1/2"

Penetration As

CFI68

CFI23

ICF87

On Dwg. No. 142-0220

CA1-109
TE

27
CFI23

EEL 753'-0"

EF

CA1-120
TE

CA1-110
TE

CFE26
KFSB

ICF118

ICF129B &
EEL 752'-6"

CF-04
CF16

CF-02

18

CF-01
CF-04

For Con

EEL 760'-9"

20'-0"

DD

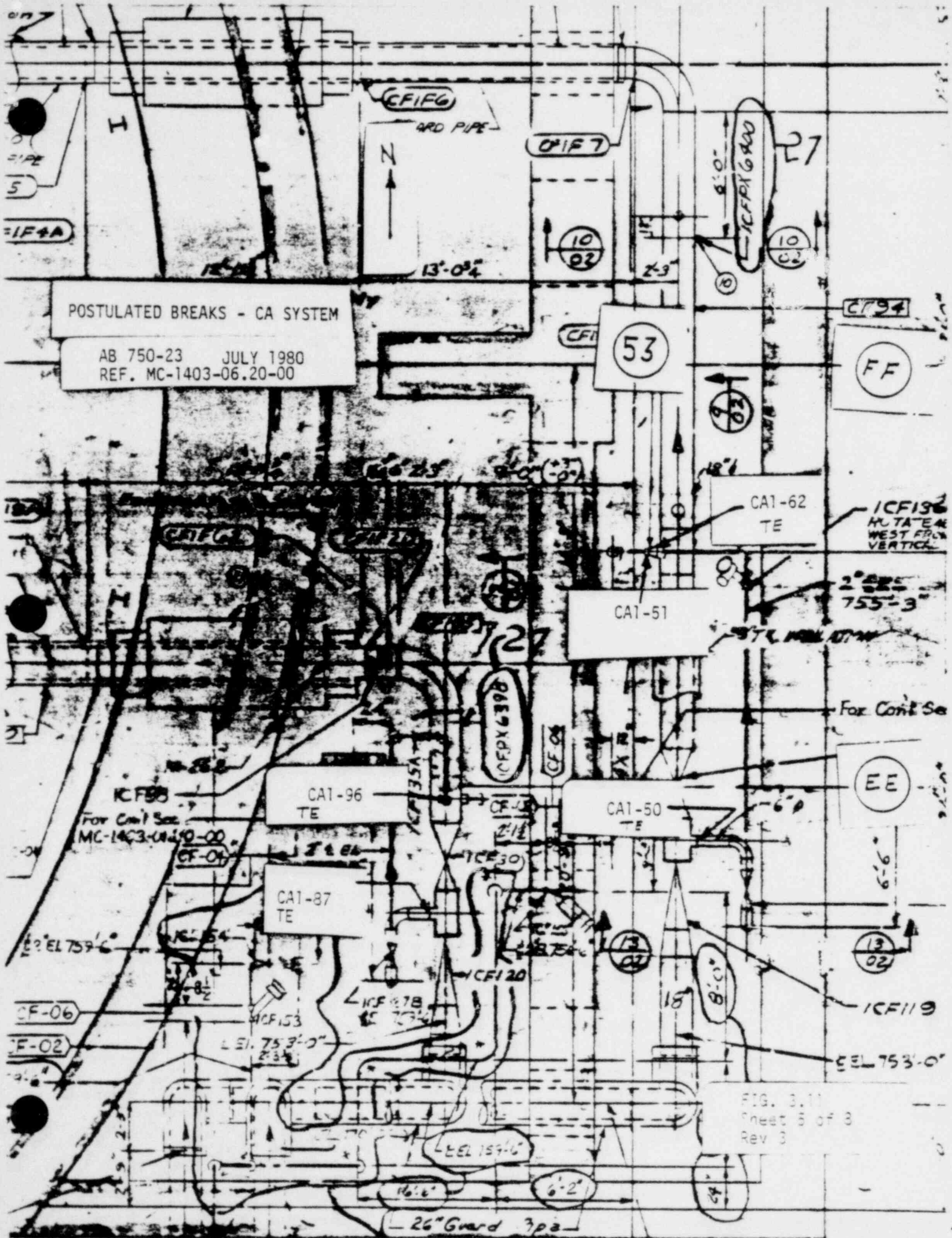
43

50

60

FIG. 3.11
Sheet 4 of 8
Rev 3

26" Guard Pipe



POSTULATED BREAKS - CA SYSTEM
 AB 750-23 JULY 1980
 REF. MC-1403-06.20-00

CAT-62
 TE

CAT-51

CAT-96
 TE

CAT-50
 TE

CAT-87
 TE

FIG. 3.11
 SHEET 3 OF 3
 REV 0

26" Guard Spa

POSTULATED BREAKS - CA SYSTEM

AB 750-23 July 1980
Ref. 1403-03.20-00

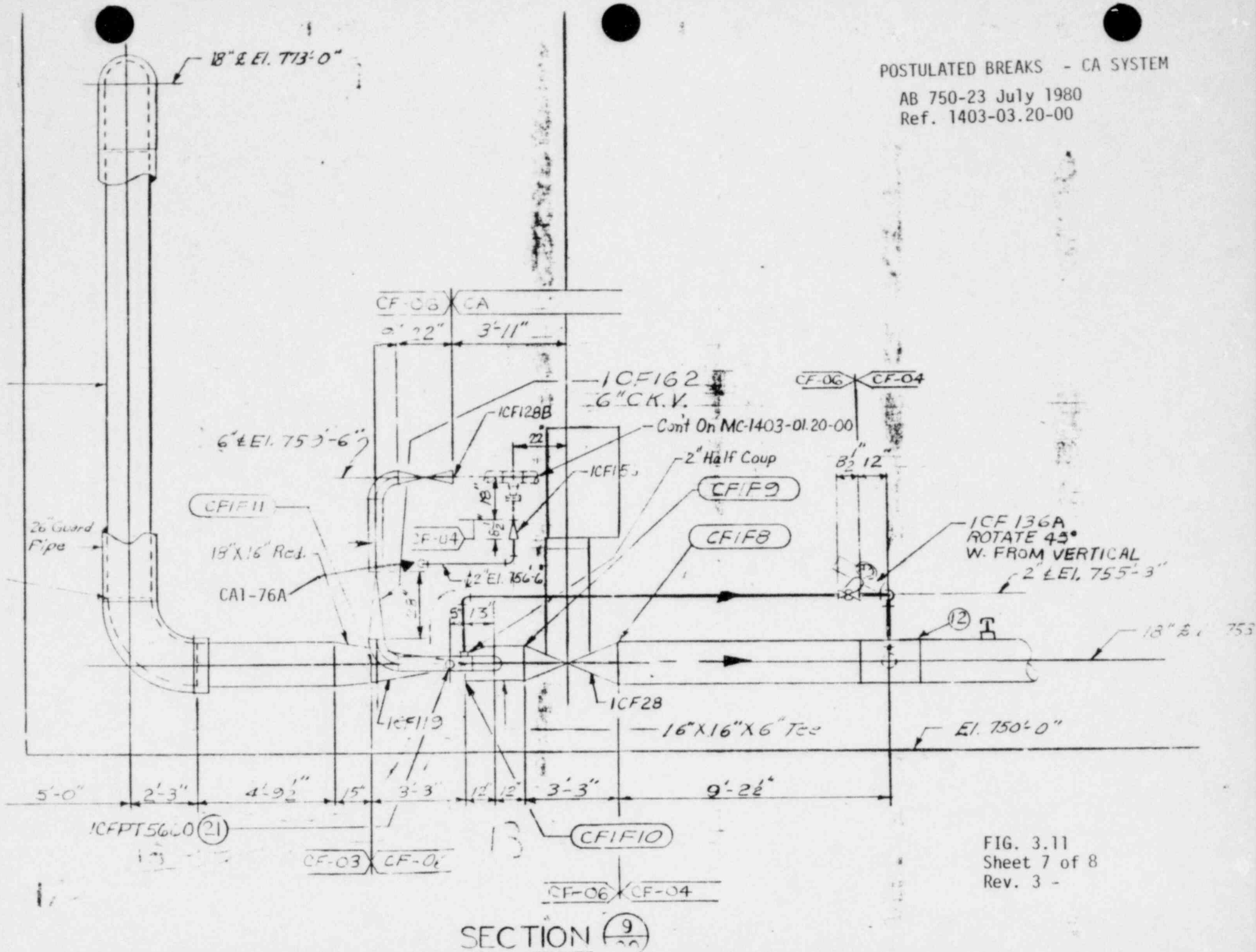


FIG. 3.11
Sheet 7 of 8
Rev. 3 -

POSTULATED BREAKS - CA SYSTEM
JULY 1980

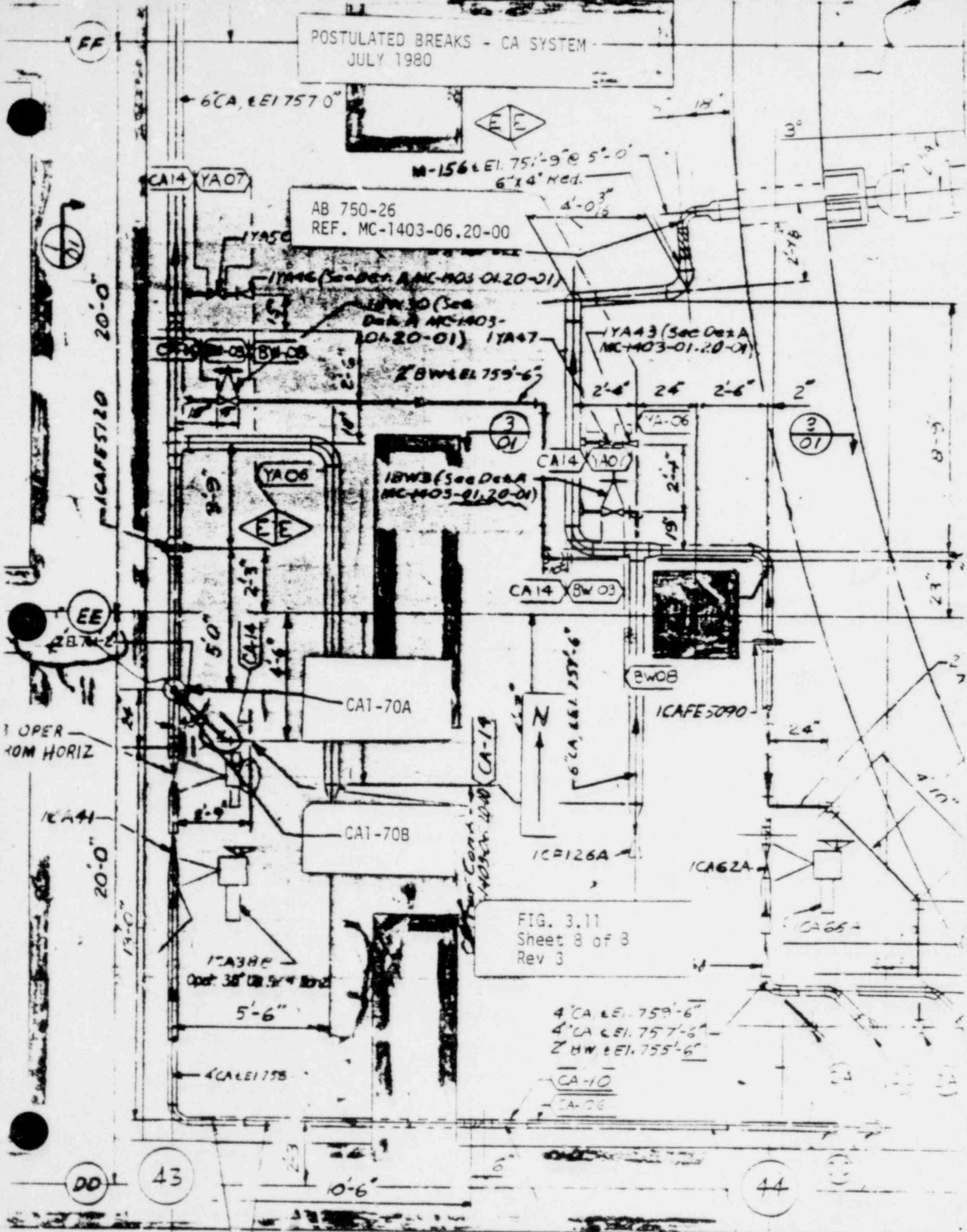


FIG. 3.11
Sheet 8 of 8
Rev 3

4" CA EEL 759'-6"
4" CA EEL 757'-6"
2" HW EEL 755'-6"

ICAF5120 20'-0"

ICAF5090

3 OPER ROOM HORIZ

43

44

POSTULATED BREAKS - CA SYSTEM
 AB 750-23 JULY 1980
 REF: MC-1403-06.20-00

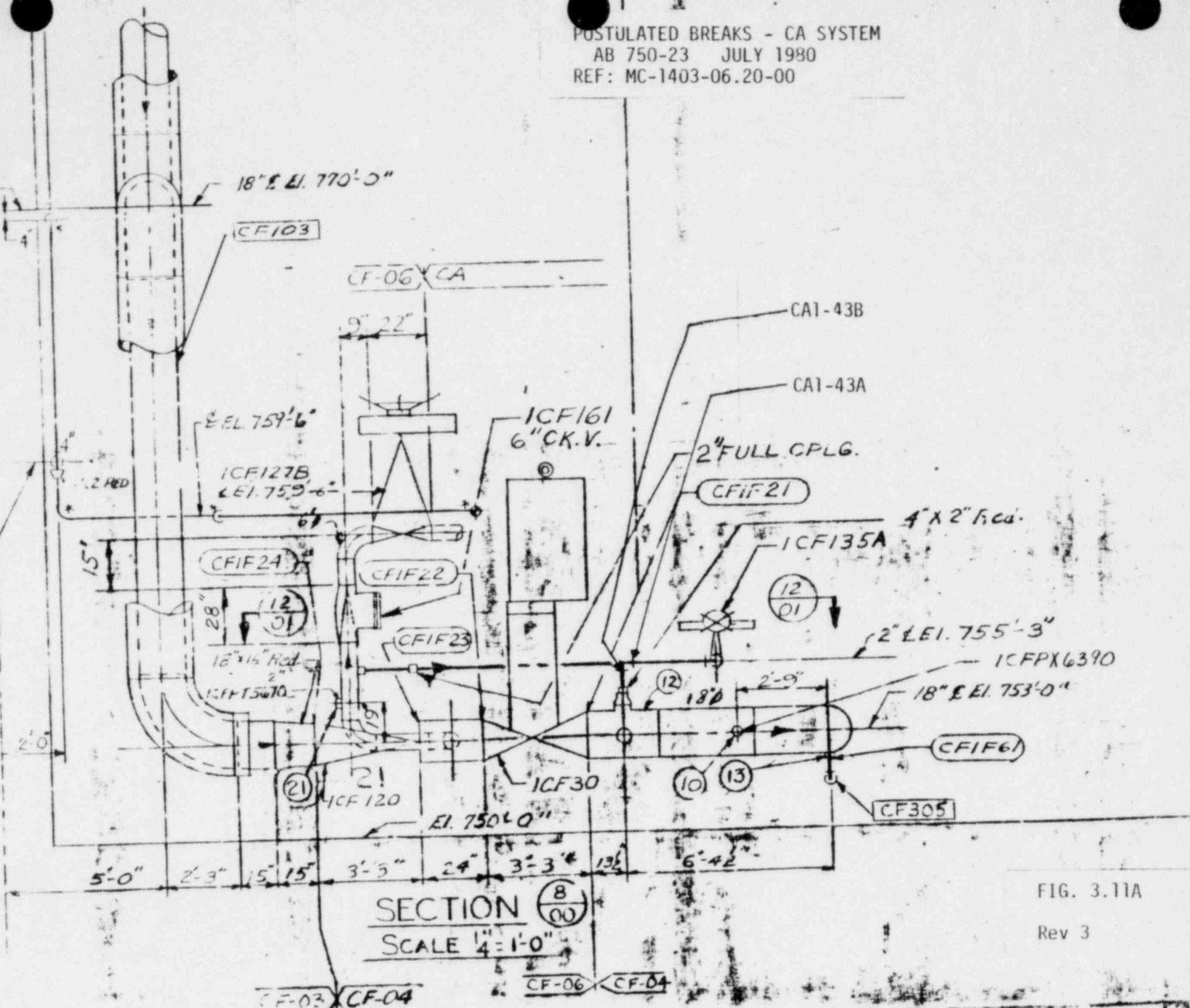
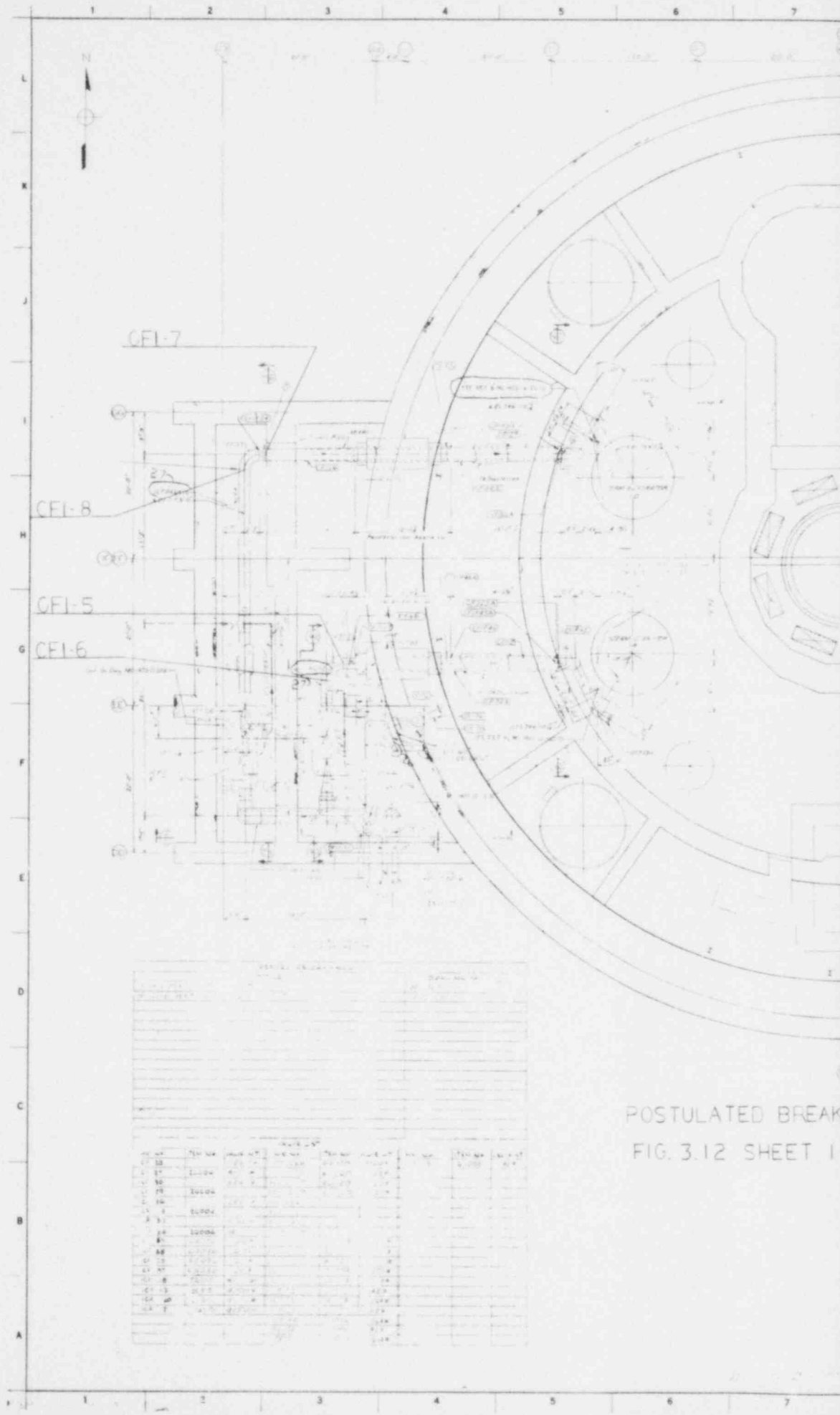
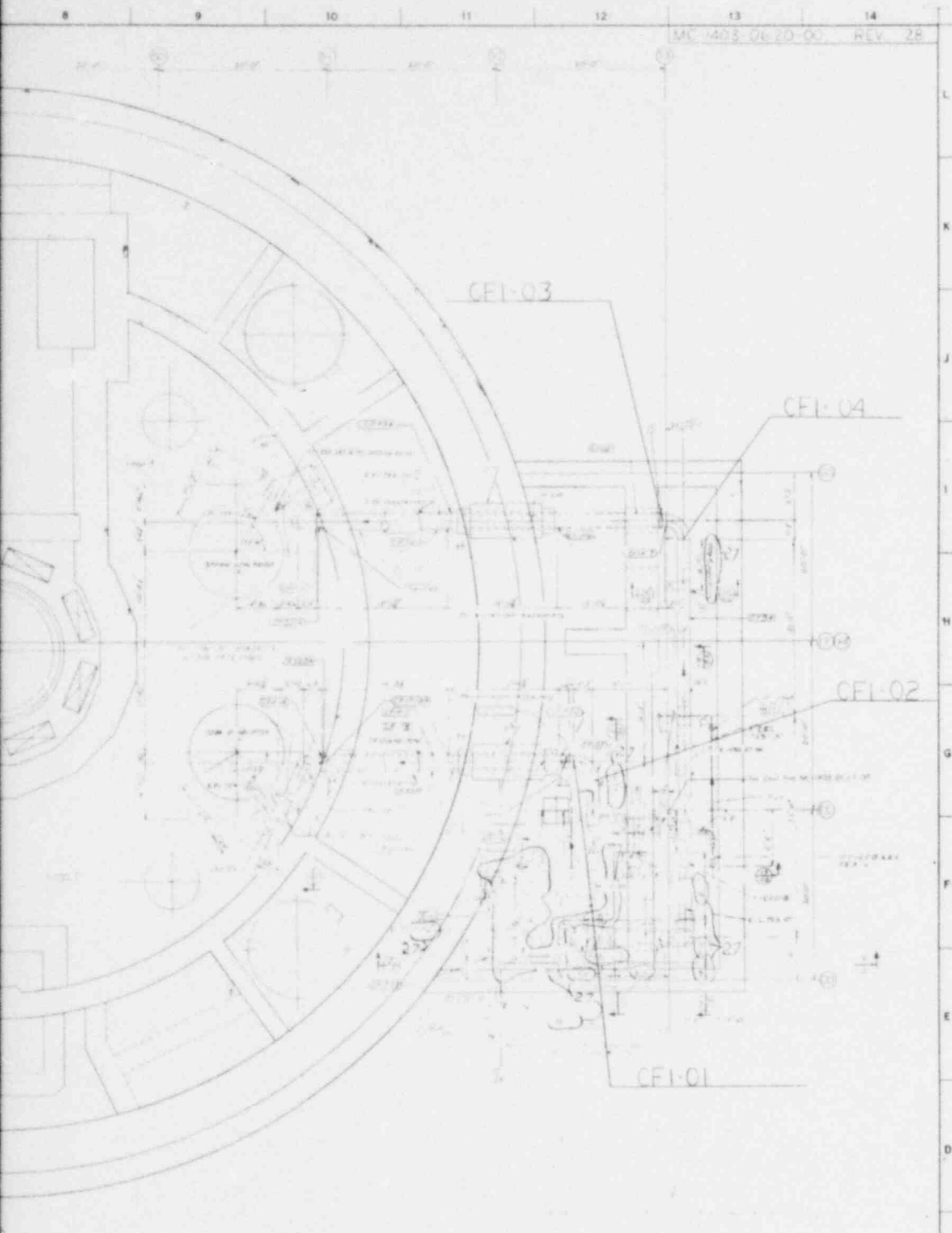


FIG. 3.11A

Rev 3



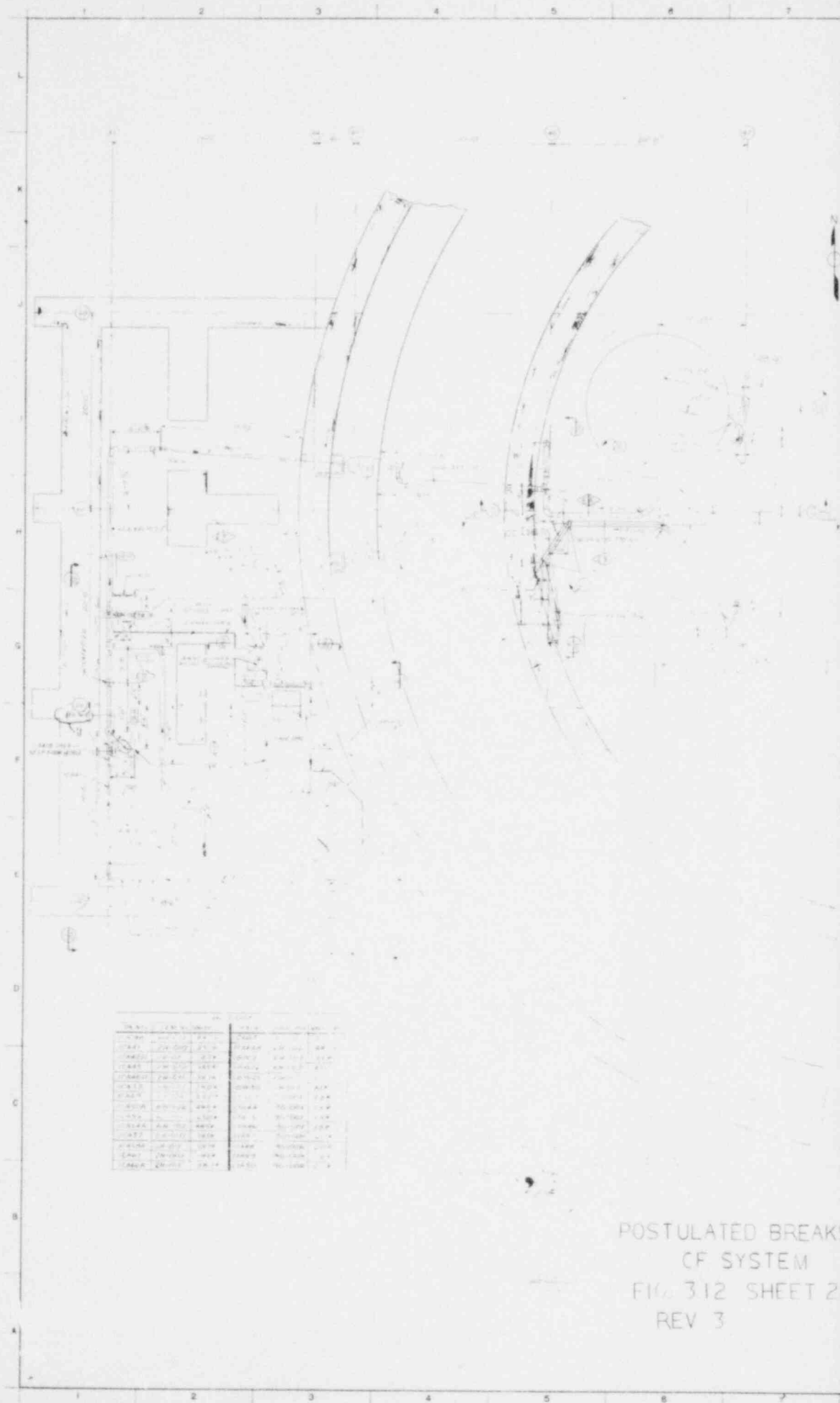
Item No.	Description	Material	Quantity	Unit	Remarks
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15
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17
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100



S IN CF SYSTEM
OF 6 REV. 3

NO.	REVISION	DATE	BY	CHKD.	APP'D.	DESCRIPTION
1						
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11						
12						
13						
14						

NUCLEAR SAFETY RELATED
 SUSS POWER COMPANY
 MIGUIRE NUCLEAR STATION - UNIT 1
 WING LAYOUT
 TRITON WATER (3) SYSTEM
 PLAN
 REACTOR BUILDING



ITEM NO.	DESCRIPTION	QTY	UNIT	PRICE	TOTAL
1001	PIPE 24" DIA	100	FT	1.50	150.00
1002	PIPE 20" DIA	200	FT	1.20	240.00
1003	PIPE 18" DIA	300	FT	1.00	300.00
1004	PIPE 16" DIA	400	FT	.80	320.00
1005	PIPE 14" DIA	500	FT	.60	300.00
1006	PIPE 12" DIA	600	FT	.40	240.00
1007	PIPE 10" DIA	700	FT	.30	210.00
1008	PIPE 8" DIA	800	FT	.20	160.00
1009	PIPE 6" DIA	900	FT	.15	135.00
1010	PIPE 4" DIA	1000	FT	.10	100.00
1011	FLANGE 24" DIA	10	EA	15.00	150.00
1012	FLANGE 20" DIA	20	EA	12.00	240.00
1013	FLANGE 18" DIA	30	EA	10.00	300.00
1014	FLANGE 16" DIA	40	EA	8.00	320.00
1015	FLANGE 14" DIA	50	EA	6.00	300.00
1016	FLANGE 12" DIA	60	EA	4.00	240.00
1017	FLANGE 10" DIA	70	EA	3.00	210.00
1018	FLANGE 8" DIA	80	EA	2.00	160.00
1019	FLANGE 6" DIA	90	EA	1.50	135.00
1020	FLANGE 4" DIA	100	EA	1.00	100.00

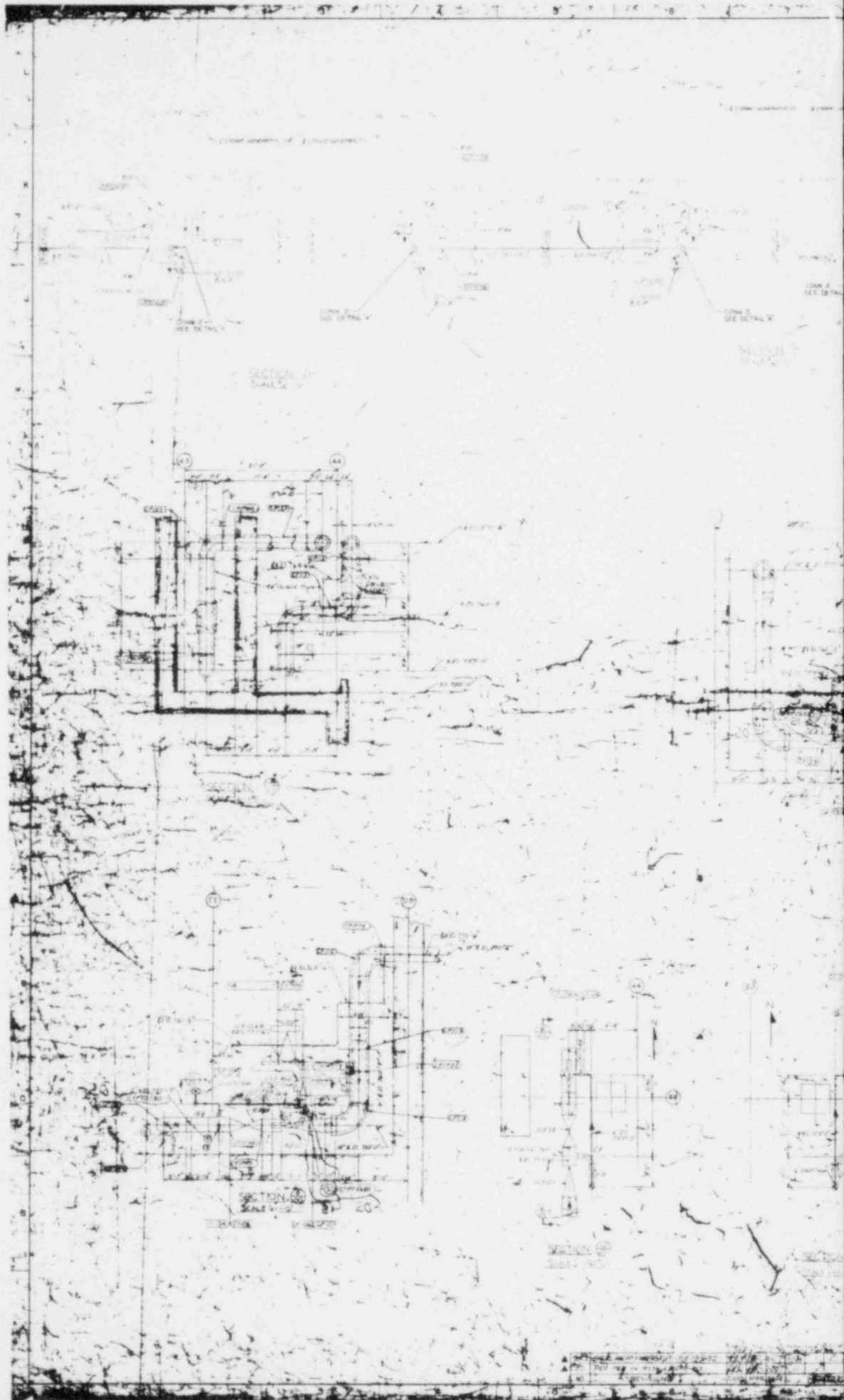
POSTULATED BREAKS
 CF SYSTEM
 FIG. 312 SHEET 2
 REV 3

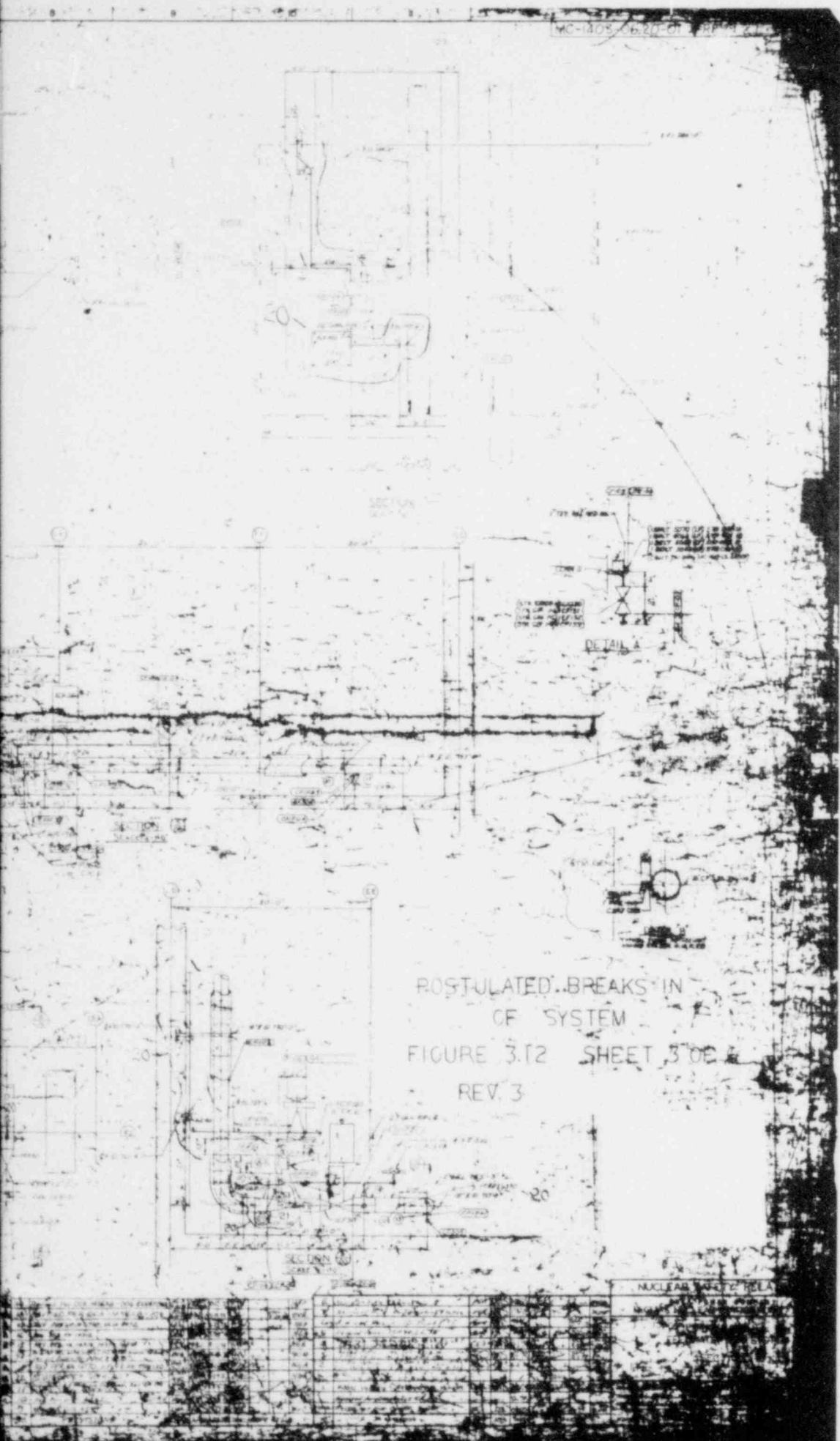


IN
OF 6

NO.	REVISIONS	CHG. APPR.	DATE	BY	CHKD.	DATE	BY	INSPECTED
1								
2								
3								
4								
5								
6								
7								
8								
9								
10								
11								
12								

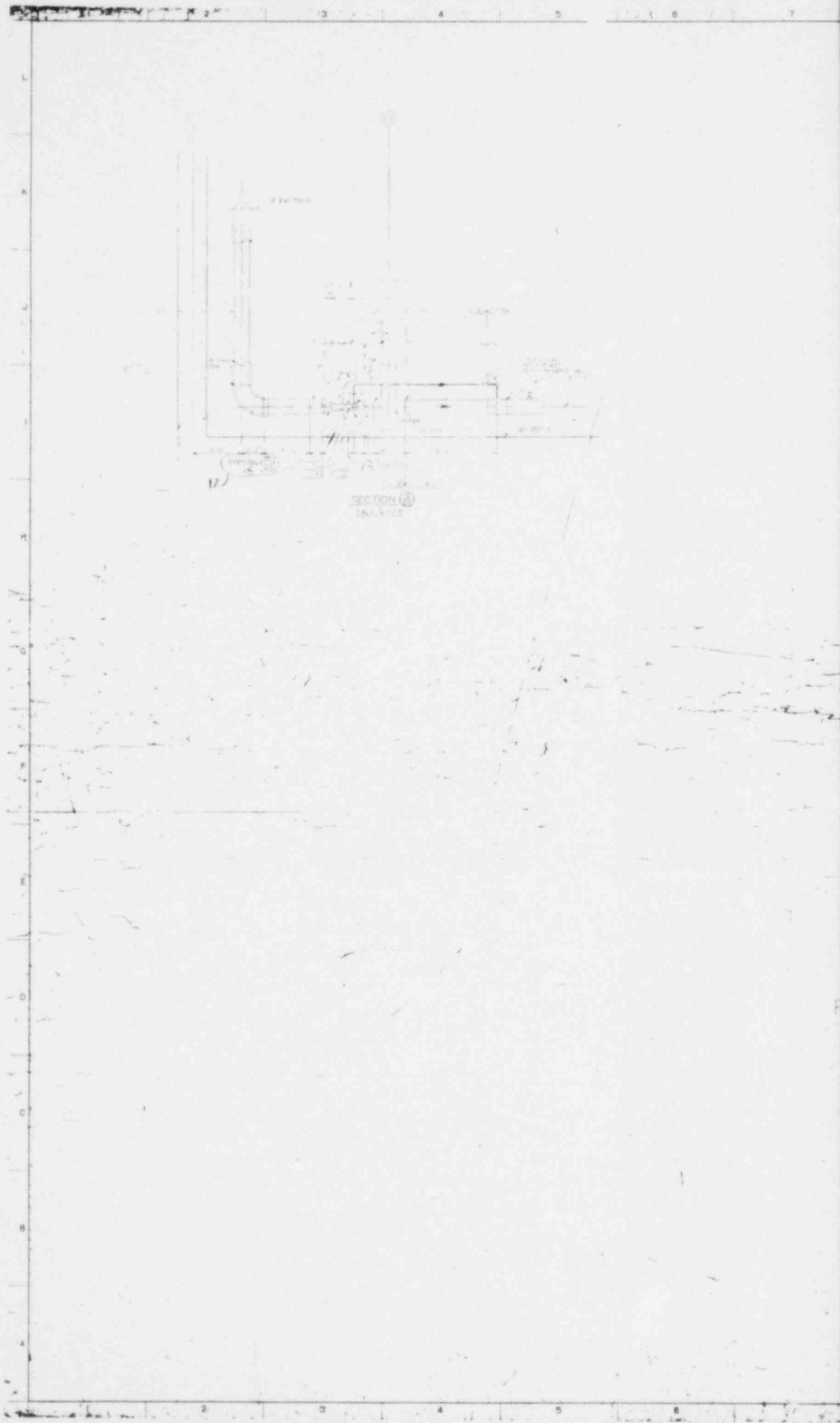
THIS Dwg. HAS BEEN REVISIONED IN AN EMERGENCY. V.P.			
FLOW DIAGRAM	REV. NO.	DATE	ISSUED BY
NUCLEAR SAFETY RELATED			
DUKE POWER COMPANY			
MOORE NUO EAR STATION-UNIT 1			
PIPING LAYOUT			
CA-100 VA-5 TO SYSTEM			
PLAN			
REACTOR BUILDING			
DESIGNER	C.L.E.	APP.	DATE
DRAWN	DATE	APP.	DATE
CHECKED	DATE	APP.	DATE
SCALE			
Dwg. NO. MG-1403-01, 20-00			





PROPOSED BREAKS IN
 OF SYSTEM
 FIGURE 3.12 SHEET 3 OF 4
 REV. 3

NUCLEAR SAFETY REL.





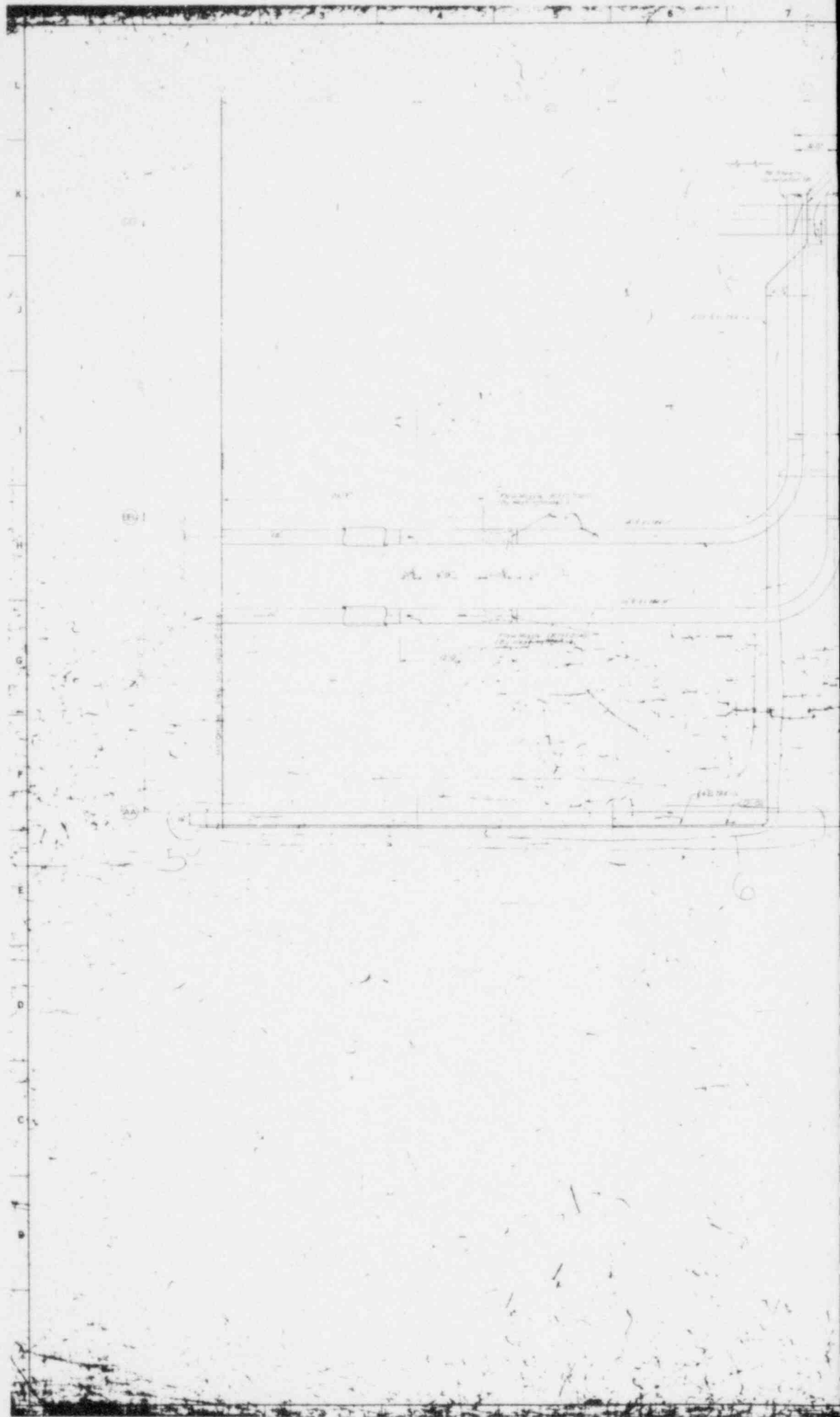
POSTULATED BREAKS IN CF SYSTEM

FIG. 3.12 SHEET 4 OF 6

REV. 3

NO.	REVISIONS	DATE	BY	CHKD
1	REVISED PER INSTRUCTIONS			
2	REVISED PER INSTRUCTIONS			
3	REVISED PER INSTRUCTIONS			
4	REVISED PER INSTRUCTIONS			
5	REVISED PER INSTRUCTIONS			
6	REVISED PER INSTRUCTIONS			
7	REVISED PER INSTRUCTIONS			
8	REVISED PER INSTRUCTIONS			
9	REVISED PER INSTRUCTIONS			
10	REVISED PER INSTRUCTIONS			
11	REVISED PER INSTRUCTIONS			
12	REVISED PER INSTRUCTIONS			
13	REVISED PER INSTRUCTIONS			
14	REVISED PER INSTRUCTIONS			
15	REVISED PER INSTRUCTIONS			
16	REVISED PER INSTRUCTIONS			
17	REVISED PER INSTRUCTIONS			
18	REVISED PER INSTRUCTIONS			
19	REVISED PER INSTRUCTIONS			
20	REVISED PER INSTRUCTIONS			

DUKE POWER COMPANY
 LENOIR, NORTH CAROLINA
 FEDERAL ENERGY REGULATORY COMMISSION
 FORM NO. 1 (REV. 10-1-73)





POSTULATED BREAKS IN CF SYSTEM

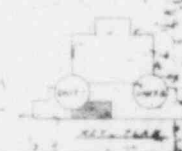
FIG. 3.12 SHEET 5 OF 6

REV. 3

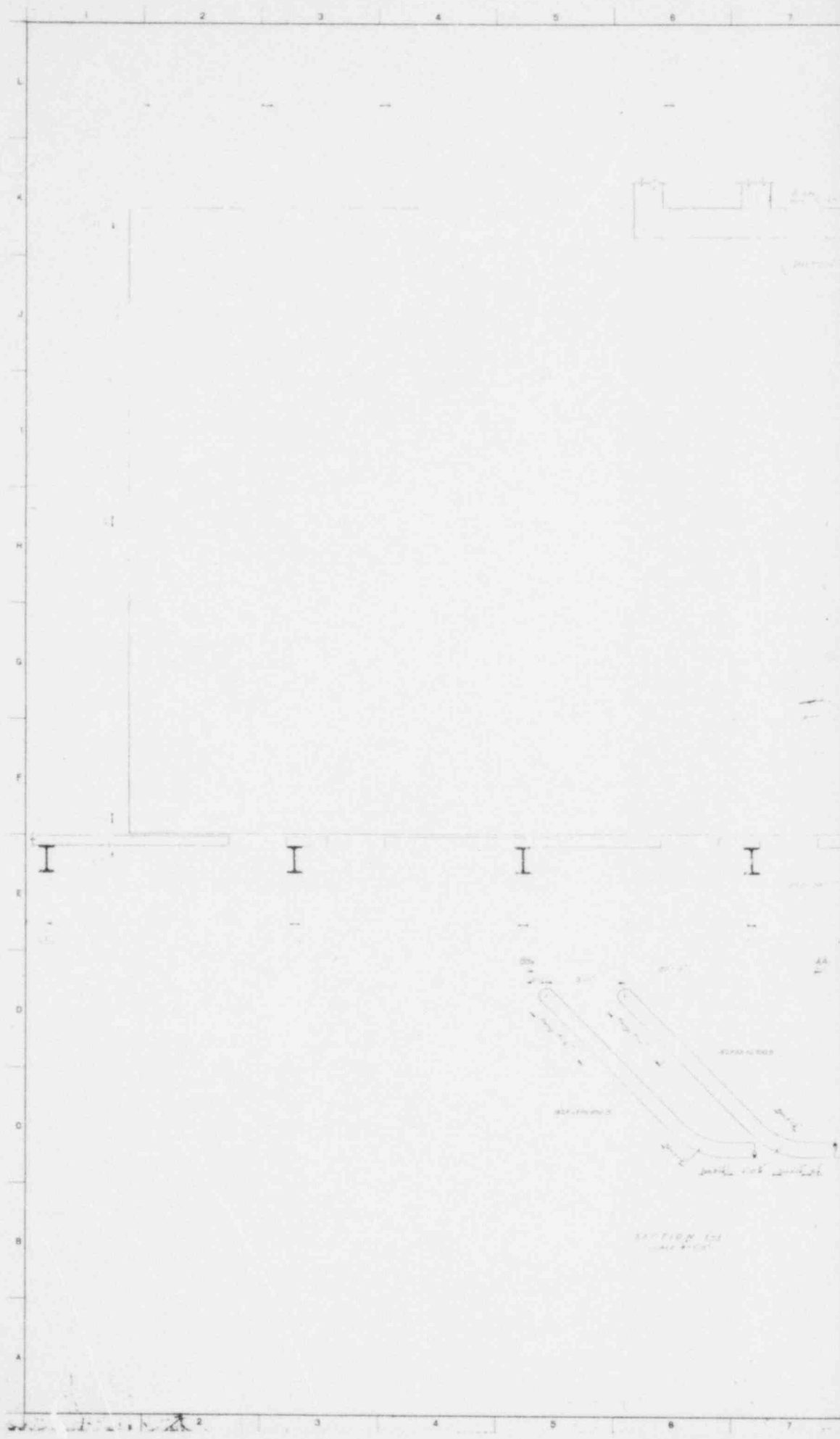
NOTE:
For General Notes See Sheet 4 of 6

- REFERENCES:
- 1. See Design Specification, Section 3.0.0.0.0.
 - 2. See Design Criteria for this Pipe & System.
 - 3. Design & Construction Manual, Volume 1.
 - 4. Construction Code, 1967.
 - 5. Construction Code, 1967.

▶ FULLY ENGINEERED

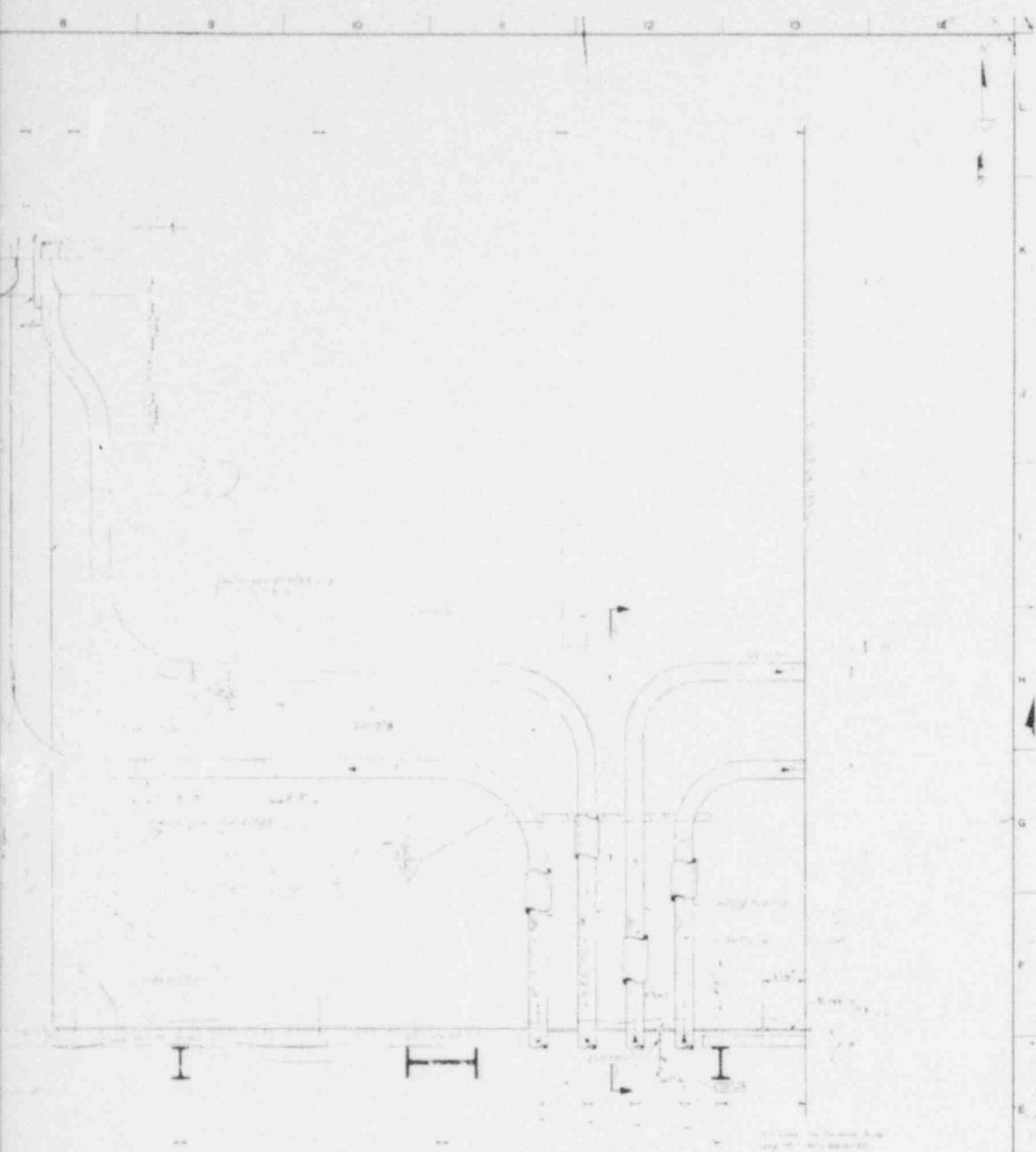


NUCLEAR SAFETY RELATED				SAGE POWER CORP MAGLURE NUCLEAR STATION - DENTON, TEXAS	
NO.	DESCRIPTION	DATE	BY	REVISION	REVISION
1	ISSUED FOR CONSTRUCTION	10/1/67	J. W. ...		
2	REVISED TO SHOW BREAK	10/1/67	J. W. ...		
3	REVISED TO SHOW BREAK	10/1/67	J. W. ...		
4	REVISED TO SHOW BREAK	10/1/67	J. W. ...		
5	REVISED TO SHOW BREAK	10/1/67	J. W. ...		
6	REVISED TO SHOW BREAK	10/1/67	J. W. ...		
7	REVISED TO SHOW BREAK	10/1/67	J. W. ...		
8	REVISED TO SHOW BREAK	10/1/67	J. W. ...		
9	REVISED TO SHOW BREAK	10/1/67	J. W. ...		
10	REVISED TO SHOW BREAK	10/1/67	J. W. ...		



SECTION AA

SECTION BB



POSTULATED BREAKS IN CF SYSTEM

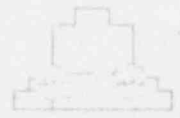
FIGURE 3.12 SHEET 6 OF 6

REV. 3

NOTE:
 "BRK 1, BRK 2, BRK 3, & BRK 4"

1. This drawing is a schematic diagram of the CF system showing the location of postulated breaks. It is not intended to be used for design purposes. All dimensions are in feet.

→ FLOW DIRECTION

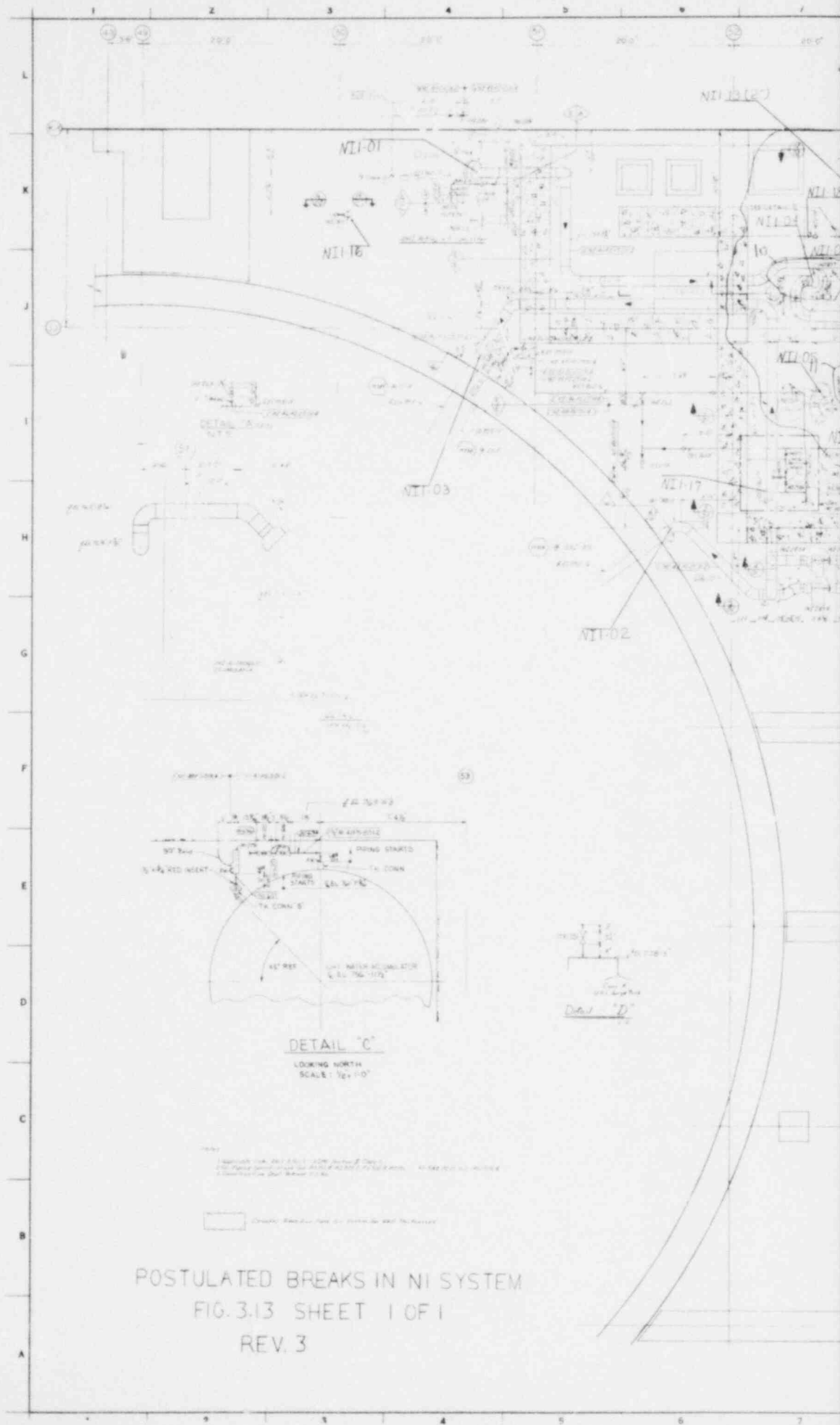


REVISIONS					SURE POWER COMPANY	
NO.	DESCRIPTION	DATE	BY	CHKD.	DESIGNED	DRAWN
1	INITIAL DESIGN	12/15/88	J. L.
2	REVISED DESIGN	12/20/88	J. L.
3	REVISED DESIGN	12/20/88	J. L.
4	REVISED DESIGN	12/20/88	J. L.
5	REVISED DESIGN	12/20/88	J. L.
6	REVISED DESIGN	12/20/88	J. L.
7	REVISED DESIGN	12/20/88	J. L.
8	REVISED DESIGN	12/20/88	J. L.
9	REVISED DESIGN	12/20/88	J. L.
10	REVISED DESIGN	12/20/88	J. L.
11	REVISED DESIGN	12/20/88	J. L.
12	REVISED DESIGN	12/20/88	J. L.
13	REVISED DESIGN	12/20/88	J. L.
14	REVISED DESIGN	12/20/88	J. L.
15	REVISED DESIGN	12/20/88	J. L.
16	REVISED DESIGN	12/20/88	J. L.
17	REVISED DESIGN	12/20/88	J. L.
18	REVISED DESIGN	12/20/88	J. L.
19	REVISED DESIGN	12/20/88	J. L.
20	REVISED DESIGN	12/20/88	J. L.

CONFIDENTIAL

FORM LAYOUT
 OF CF
 BOX OF AN & SECTION X
 FROM THE DISTING

403 0545 00



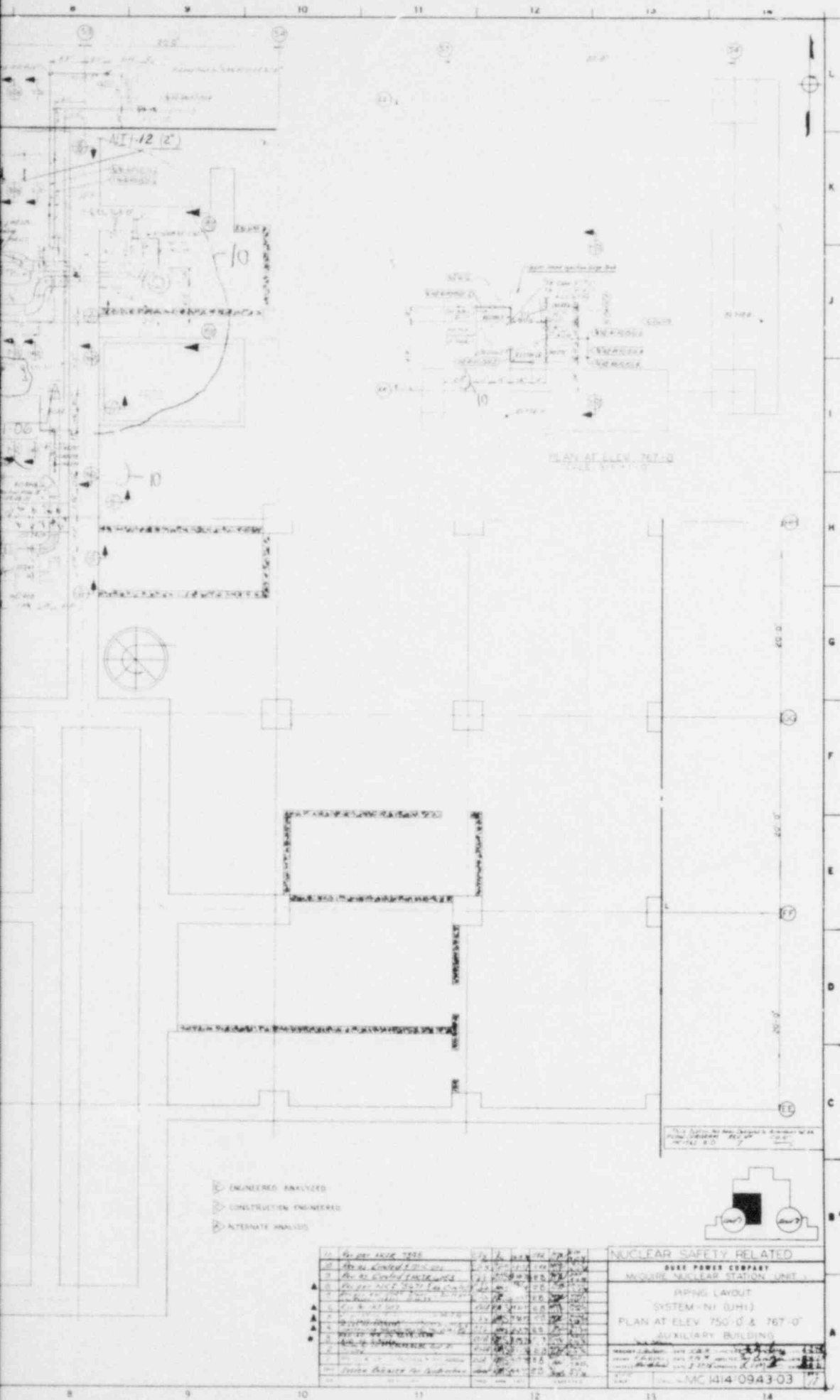
DETAIL "C"

LOOKING NORTH
SCALE: 1/2" = 1'-0"

1. APPROVED FOR RELEASE BY NSA ON 05-08-2014 pursuant to E.O. 13526, ALL INFORMATION CONTAINED HEREIN IS UNCLASSIFIED EXCEPT WHERE SHOWN OTHERWISE.

CLASSIFIED BY: [redacted] ON: [redacted] FOR: [redacted]

POSTULATED BREAKS IN NI SYSTEM
FIG. 3.13 SHEET 1 OF 1
REV. 3



- ▷ ENGINEER ANALYZED
- ▷ CONSTRUCTION ENGINEER
- ▷ ALTERNATE ANALYSIS

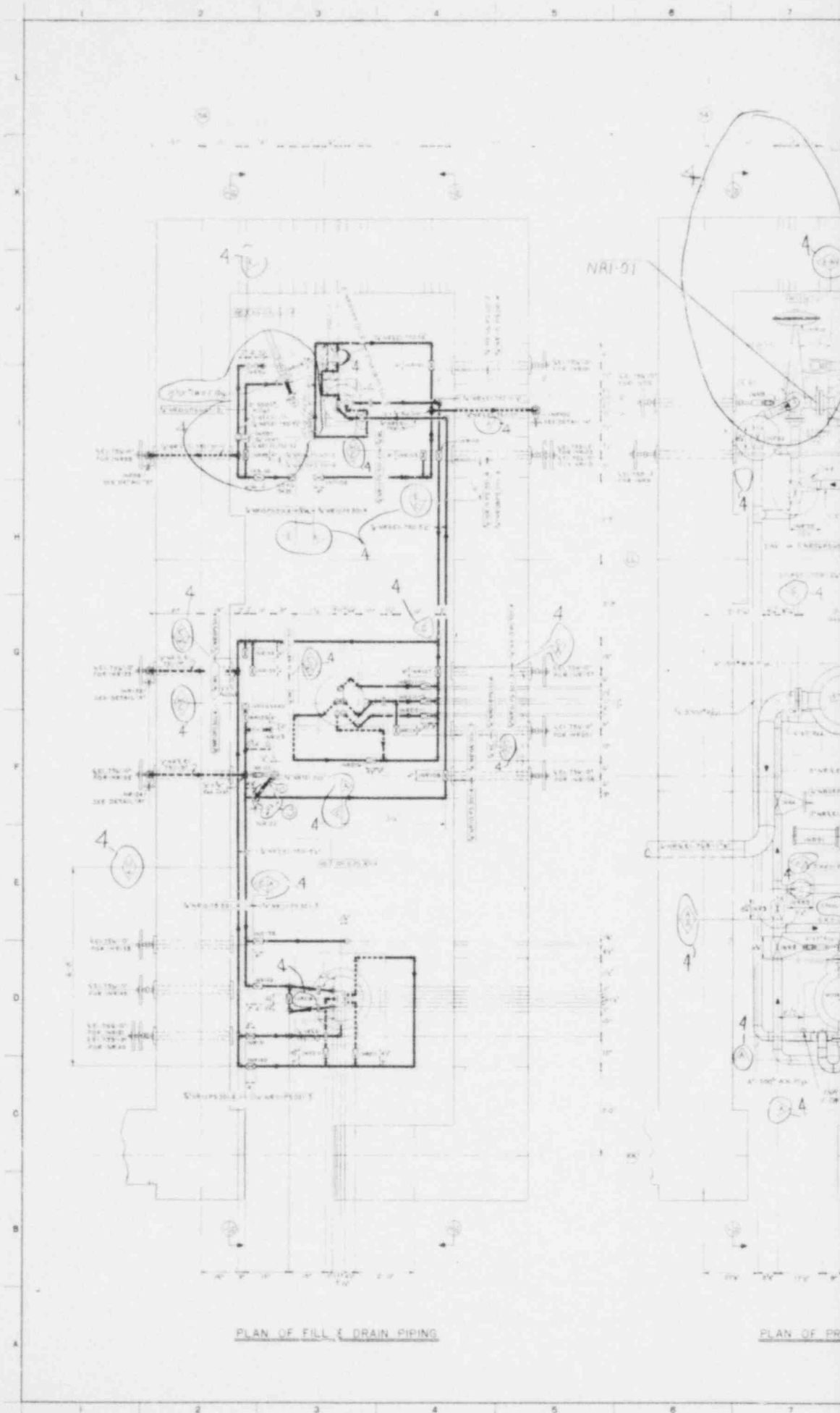
NO.	DESCRIPTION	DATE	BY	CHKD.	APP.
1	As per 4414-1000	11/15/77	J.P.	J.P.	J.P.
2	As per 4414-1000	11/15/77	J.P.	J.P.	J.P.
3	As per 4414-1000	11/15/77	J.P.	J.P.	J.P.
4	As per 4414-1000	11/15/77	J.P.	J.P.	J.P.
5	As per 4414-1000	11/15/77	J.P.	J.P.	J.P.
6	As per 4414-1000	11/15/77	J.P.	J.P.	J.P.
7	As per 4414-1000	11/15/77	J.P.	J.P.	J.P.
8	As per 4414-1000	11/15/77	J.P.	J.P.	J.P.
9	As per 4414-1000	11/15/77	J.P.	J.P.	J.P.
10	As per 4414-1000	11/15/77	J.P.	J.P.	J.P.
11	As per 4414-1000	11/15/77	J.P.	J.P.	J.P.
12	As per 4414-1000	11/15/77	J.P.	J.P.	J.P.
13	As per 4414-1000	11/15/77	J.P.	J.P.	J.P.
14	As per 4414-1000	11/15/77	J.P.	J.P.	J.P.

NUCLEAR SAFETY RELATED

DUKE POWER COMPANY
MOORE NUCLEAR STATION (UNIT 1)

PIPING LAYOUT
SYSTEM-NI (UHI)
PLAN AT ELEV. 750'-0" & 767'-0"
AUXILIARY BUILDING

MOORE NUCLEAR STATION
DUKE POWER COMPANY
MOORE, NORTH CAROLINA
PROJECT NO. 4414-1000
DRAWING NO. 4414-10943-03



PLAN OF FILL & DRAIN PIPING

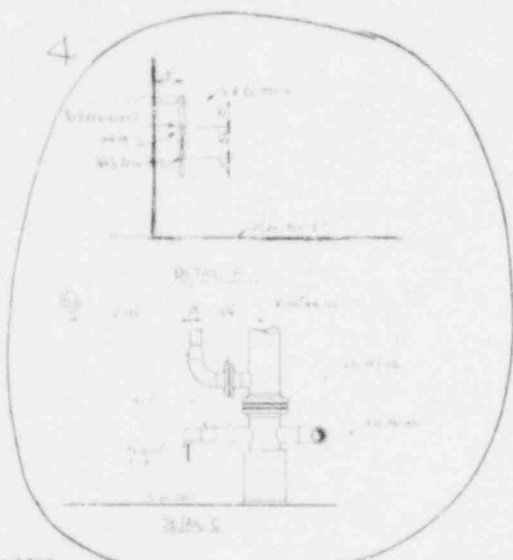
PLAN OF PRO

POSTULATED BREAKS IN NR SYSTEM

FIG. 3.14 SHEET 1 OF 1
REV. 3



DETAIL "A"



NOTES

1. ALL DRAIN PIPING TO THE WASTE TANK WHEN CONNECTED TO THE DRAINAGE SYSTEM SHALL BE SUPPORTED UNLESS OTHERWISE NOTED TO THE CONTRARY.
2. WHEREVER DRAIN PIPING IS TO BE INSTALLED IN FLOOR SLABS, THE SLABS ARE TO BE REINFORCED TO CARRY THE PIPE.
3. APPLICABLE CODES: ASME, CC, CLASS 2 & 3, ASME, B31.1.
4. FOR PIPE SPECIFICATIONS SEE: PG-00-00, PG-00-01, PG-00-02.

E- FULL ENGINEERED
A- ALTERNATE ANALYSIS
C- CONSTRUCTION

THIS DRAWING HAS BEEN DESIGNED IN ACCORDANCE WITH
FLOW DIAGRAM NO. _____ REVISION NO. _____ CHANGE ORDER NO. _____
BY: _____ DATE: _____
CHECKED BY: _____ DATE: _____

VALVE LIST										
VALVE	SIZE	TYPE	NO.	LOCATION	OPERATOR	OPERATION	STATUS	DATE	REVISION	REMARKS
NR1	2"	200-2	219	DRUM	MAN	---	---	---	---	---
NR2	2"	200-2	219	MAN	---	---	---	---	---	---
NR3	2"	200-2	219	DRUM	MAN	---	---	---	---	---
NR4	2"	200-2	219	MAN	---	---	---	---	---	---
NR5	2"	200-2	219	DRUM	MAN	---	---	---	---	---
NR6	2"	200-2	219	DRUM	MAN	---	---	---	---	---
NR7	2"	200-2	219	DRUM	MAN	---	---	---	---	---
NR8	2"	200-2	219	DRUM	MAN	---	---	---	---	---
NR9	2"	200-2	219	DRUM	MAN	---	---	---	---	---
NR10	2"	200-2	219	DRUM	MAN	---	---	---	---	---
NR11	2"	200-2	219	DRUM	MAN	---	---	---	---	---
NR12	2"	200-2	219	DRUM	MAN	---	---	---	---	---
NR13	2"	200-2	219	DRUM	MAN	---	---	---	---	---
NR14	2"	200-2	219	DRUM	MAN	---	---	---	---	---
NR15	2"	200-2	219	DRUM	MAN	---	---	---	---	---
NR16	2"	200-2	219	DRUM	MAN	---	---	---	---	---
NR17	2"	200-2	219	DRUM	MAN	---	---	---	---	---
NR18	2"	200-2	219	DRUM	MAN	---	---	---	---	---
NR19	2"	200-2	219	DRUM	MAN	---	---	---	---	---
NR20	2"	200-2	219	DRUM	MAN	---	---	---	---	---
NR21	2"	200-2	219	DRUM	MAN	---	---	---	---	---
NR22	2"	200-2	219	DRUM	MAN	---	---	---	---	---
NR23	2"	200-2	219	DRUM	MAN	---	---	---	---	---
NR24	2"	200-2	219	DRUM	MAN	---	---	---	---	---
NR25	2"	200-2	219	DRUM	MAN	---	---	---	---	---
NR26	2"	200-2	219	DRUM	MAN	---	---	---	---	---
NR27	2"	200-2	219	DRUM	MAN	---	---	---	---	---
NR28	2"	200-2	219	DRUM	MAN	---	---	---	---	---
NR29	2"	200-2	219	DRUM	MAN	---	---	---	---	---
NR30	2"	200-2	219	DRUM	MAN	---	---	---	---	---

NUCLEAR SAFETY RELATED

DUKE POWER COMPANY
MCQUEE NUCLEAR STATION UNIT 2

PIPING LAYOUT SYS.
NR
PLAN AT ELEVATION 750'-0"
AUXILIARY BUILDING

DRAWN BY: _____ DATE: _____
CHECKED BY: _____ DATE: _____
SCALE: _____
DWG NO. MC-1414-1843-00

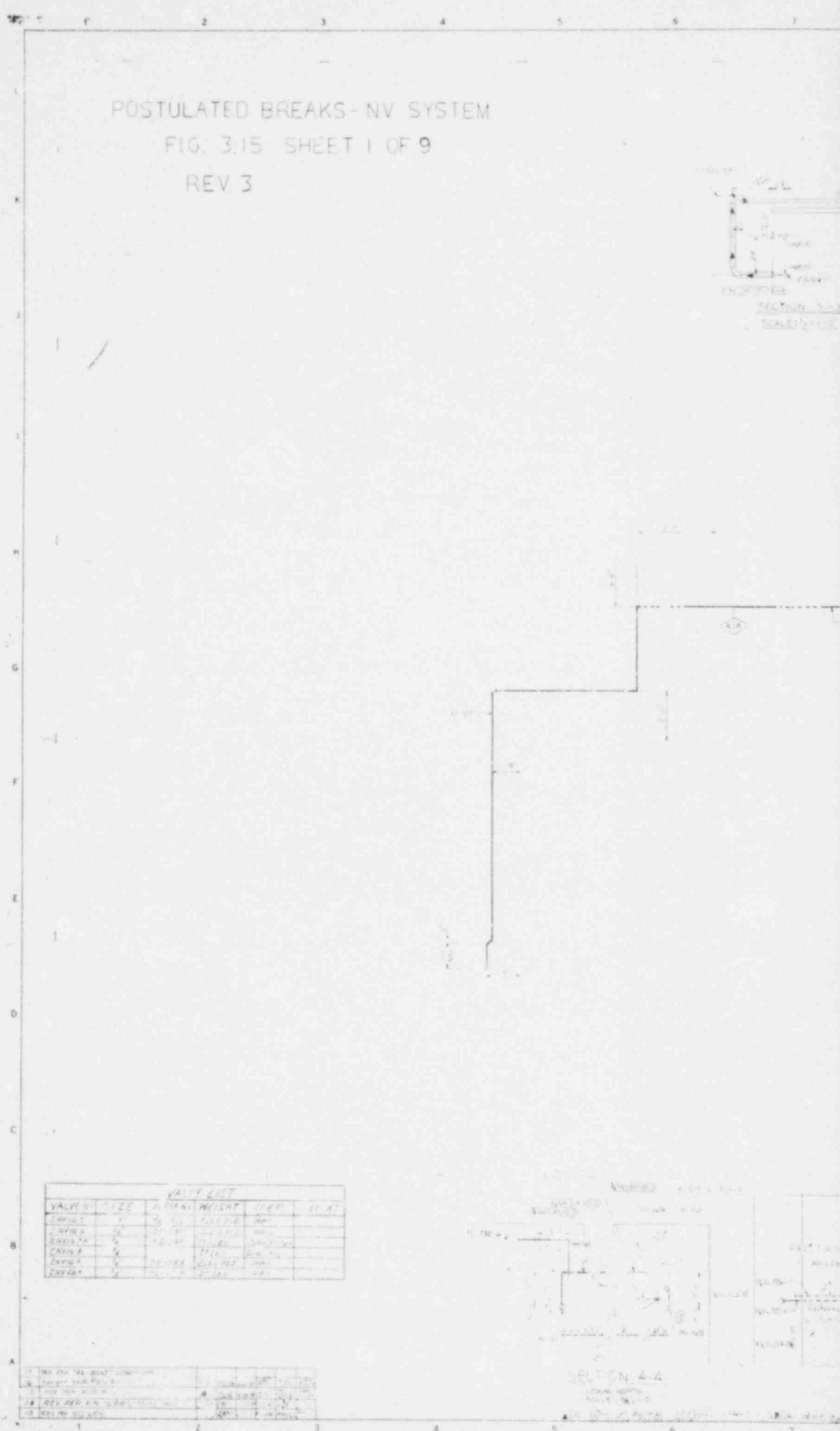
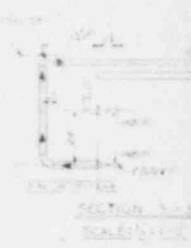
NO.	REVISIONS	DATE	BY	CHKD.	APP.	REMARKS

LESS PIPING

POSTULATED BREAKS - NV SYSTEM

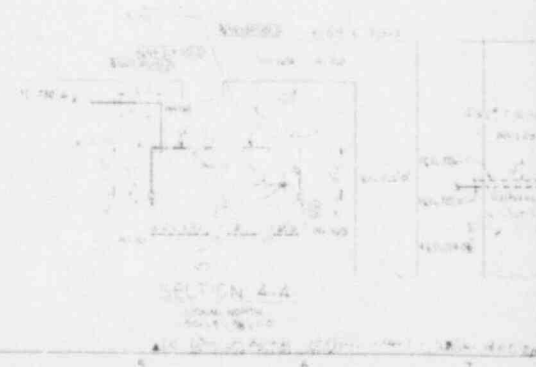
FIG. 3.15 SHEET 1 OF 9

REV. 3



VALVE LIST

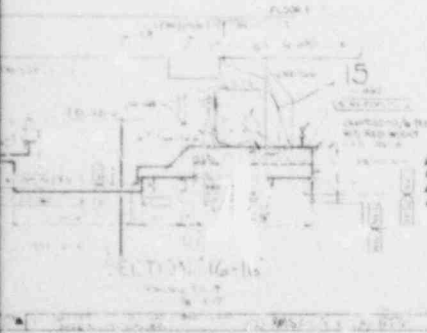
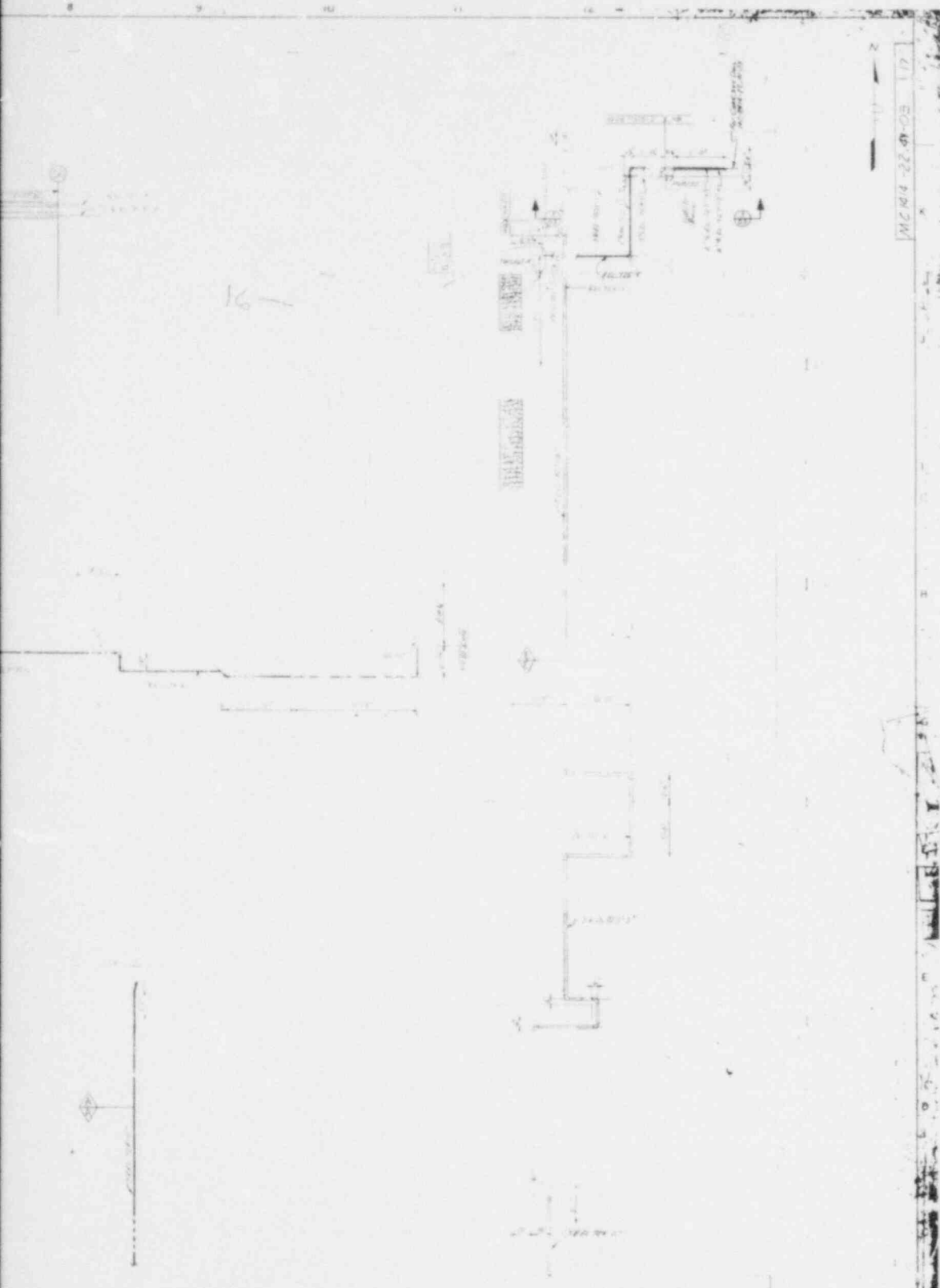
VALVE NO.	SIZE	ALPHA	HEIGHT	CLASS	TYPE
VALVE 1	2"	10	10'-0"	100	100
VALVE 2	2"	10	10'-0"	100	100
VALVE 3	2"	10	10'-0"	100	100
VALVE 4	2"	10	10'-0"	100	100
VALVE 5	2"	10	10'-0"	100	100
VALVE 6	2"	10	10'-0"	100	100
VALVE 7	2"	10	10'-0"	100	100
VALVE 8	2"	10	10'-0"	100	100



1	REV. 01	AS-BUILT	10/10/00
2	REV. 02	ADD VALVE 1	11/10/00
3	REV. 03	ADD VALVE 2	12/10/00
4	REV. 04	ADD VALVE 3	01/11/01
5	REV. 05	ADD VALVE 4	02/11/01

SECTION 2-4

MCN4-22-47-03 17



FIELD ENGINEER
A. J. MALLOTT

NO.	DESCRIPTION	DATE	BY	CHKD.
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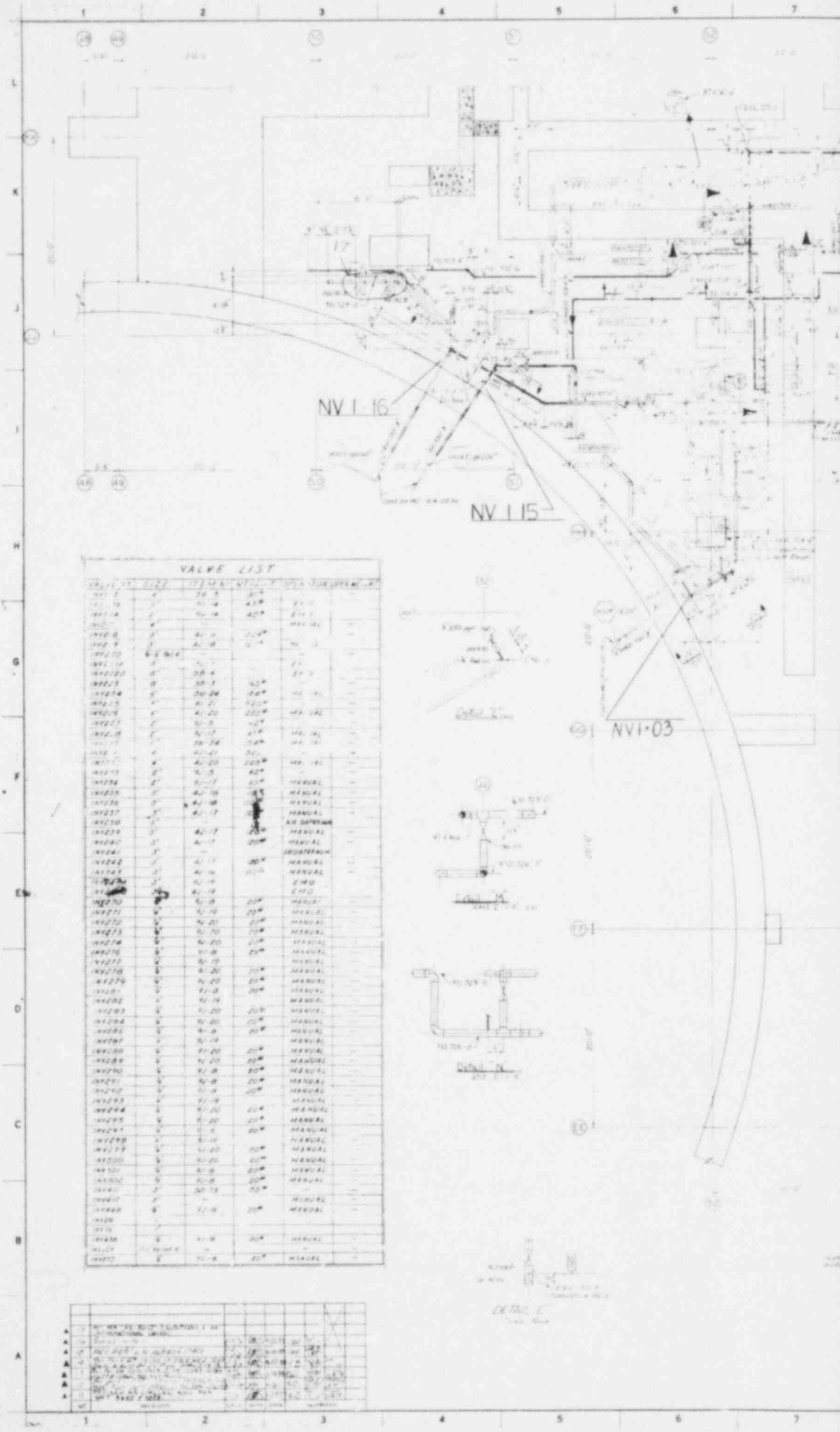
EGLE POWER COMPANY
MOORE NUCLEAR STATION - UNITS 1 & 2

PIPING LAYOUT
SYSTEM - NV
PLAN AT ELEVATION 716'0"

REVISIONS

NO.	DATE	DESCRIPTION	BY	CHKD.
1
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DATE: 10/11/67

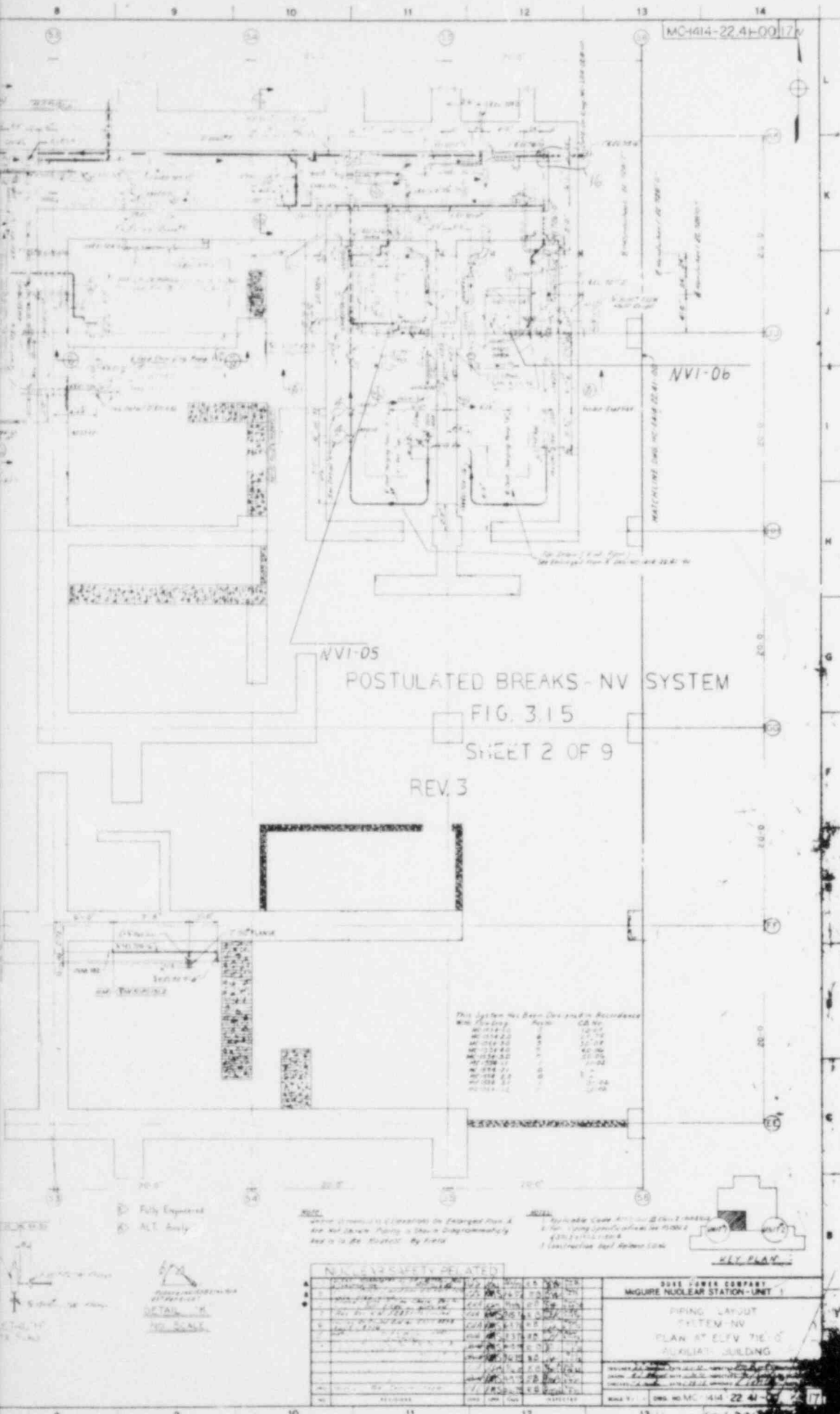


VALVE LIST

VALVE NO.	SIZE	TYPE	LOCATION	OPERATION
NV I-01	2"	20"	30"	MANUAL
NV I-02	2"	20"	30"	MANUAL
NV I-03	2"	20"	30"	MANUAL
NV I-04	2"	20"	30"	MANUAL
NV I-05	2"	20"	30"	MANUAL
NV I-06	2"	20"	30"	MANUAL
NV I-07	2"	20"	30"	MANUAL
NV I-08	2"	20"	30"	MANUAL
NV I-09	2"	20"	30"	MANUAL
NV I-10	2"	20"	30"	MANUAL
NV I-11	2"	20"	30"	MANUAL
NV I-12	2"	20"	30"	MANUAL
NV I-13	2"	20"	30"	MANUAL
NV I-14	2"	20"	30"	MANUAL
NV I-15	2"	20"	30"	MANUAL
NV I-16	2"	20"	30"	MANUAL
NV I-17	2"	20"	30"	MANUAL
NV I-18	2"	20"	30"	MANUAL
NV I-19	2"	20"	30"	MANUAL
NV I-20	2"	20"	30"	MANUAL
NV I-21	2"	20"	30"	MANUAL
NV I-22	2"	20"	30"	MANUAL
NV I-23	2"	20"	30"	MANUAL
NV I-24	2"	20"	30"	MANUAL
NV I-25	2"	20"	30"	MANUAL
NV I-26	2"	20"	30"	MANUAL
NV I-27	2"	20"	30"	MANUAL
NV I-28	2"	20"	30"	MANUAL
NV I-29	2"	20"	30"	MANUAL
NV I-30	2"	20"	30"	MANUAL
NV I-31	2"	20"	30"	MANUAL
NV I-32	2"	20"	30"	MANUAL
NV I-33	2"	20"	30"	MANUAL
NV I-34	2"	20"	30"	MANUAL
NV I-35	2"	20"	30"	MANUAL
NV I-36	2"	20"	30"	MANUAL
NV I-37	2"	20"	30"	MANUAL
NV I-38	2"	20"	30"	MANUAL
NV I-39	2"	20"	30"	MANUAL
NV I-40	2"	20"	30"	MANUAL
NV I-41	2"	20"	30"	MANUAL
NV I-42	2"	20"	30"	MANUAL
NV I-43	2"	20"	30"	MANUAL
NV I-44	2"	20"	30"	MANUAL
NV I-45	2"	20"	30"	MANUAL
NV I-46	2"	20"	30"	MANUAL
NV I-47	2"	20"	30"	MANUAL
NV I-48	2"	20"	30"	MANUAL
NV I-49	2"	20"	30"	MANUAL
NV I-50	2"	20"	30"	MANUAL

NO.	DESCRIPTION	DATE	BY
1
2
3
4
5
6
7

DETAIL E



POSTULATED BREAKS - NV SYSTEM

FIG. 3.15

SHEET 2 OF 9

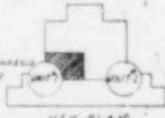
REV. 3

This system has been designed in accordance with the following:

Material	Grade	Quantity
MC-4414-22.41-0017	304 SS	12.15
MC-4414-22.41-0018	304 SS	1.75
MC-4414-22.41-0019	304 SS	12.15
MC-4414-22.41-0020	304 SS	4.00
MC-4414-22.41-0021	304 SS	12.15
MC-4414-22.41-0022	304 SS	12.15
MC-4414-22.41-0023	304 SS	1.75
MC-4414-22.41-0024	304 SS	1.75
MC-4414-22.41-0025	304 SS	1.75

NOTE: Where shown in elevation on enlarged view, the size and shape of piping is shown diagrammatically and is to be checked by field.

NOTE: See General Note A-11-1 for details. See also General Note A-11-2 for details. See also General Note A-11-3 for details. See also General Note A-11-4 for details. See also General Note A-11-5 for details.

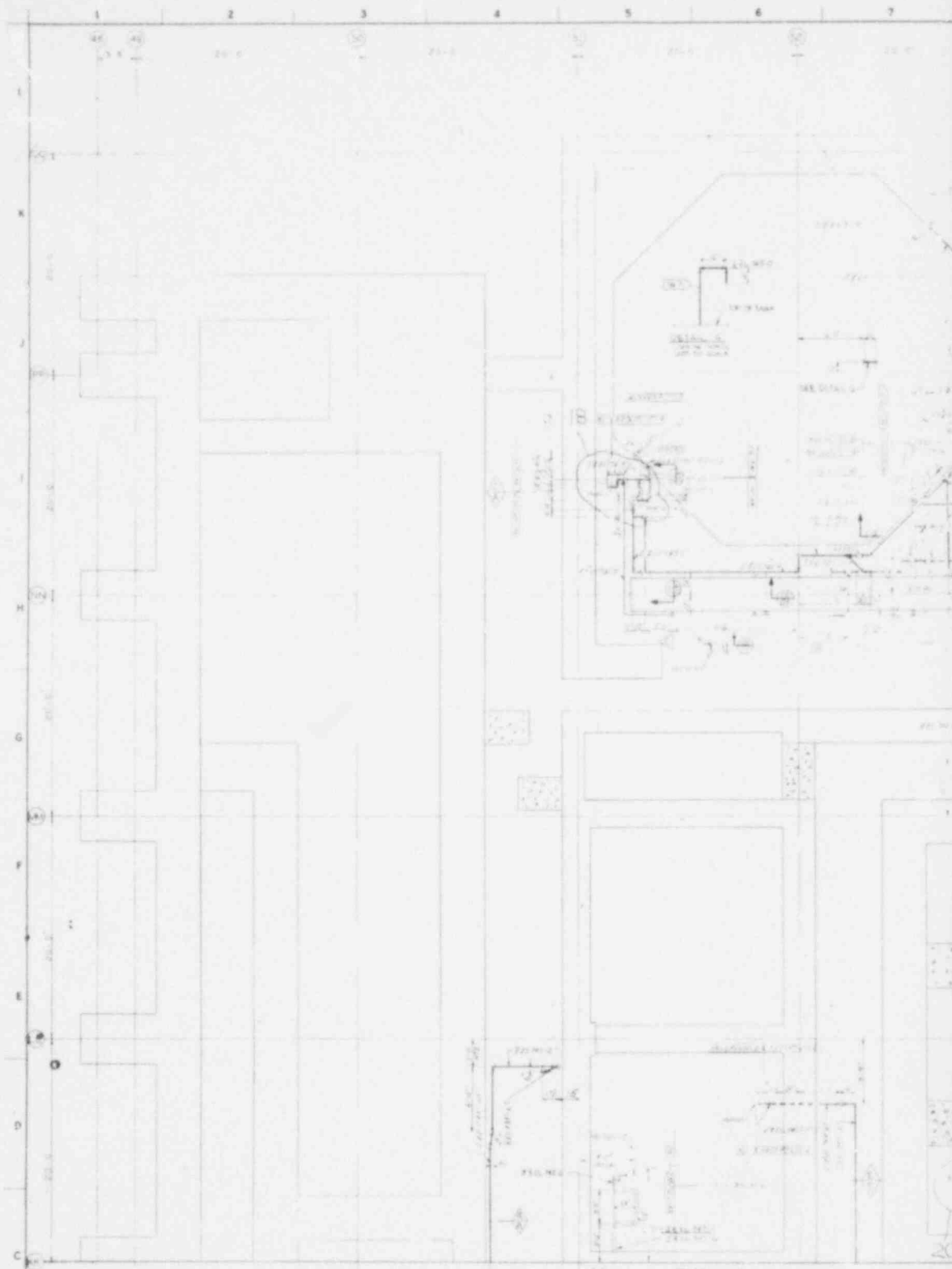


NUCLEAR SAFETY RELATED			
NO.	DESCRIPTION	QTY	GRADE
1	MC-4414-22.41-0017	12.15	304 SS
2	MC-4414-22.41-0018	1.75	304 SS
3	MC-4414-22.41-0019	12.15	304 SS
4	MC-4414-22.41-0020	4.00	304 SS
5	MC-4414-22.41-0021	12.15	304 SS
6	MC-4414-22.41-0022	12.15	304 SS
7	MC-4414-22.41-0023	1.75	304 SS
8	MC-4414-22.41-0024	1.75	304 SS
9	MC-4414-22.41-0025	1.75	304 SS

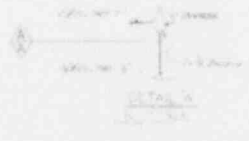
BOSS POWER COMPANY
MIGURE NUCLEAR STATION - UNIT 1

PIPING LAYOUT
SYSTEM - NV
PLAN AT ELEV. 716.0
AUXILIARY BUILDING

DATE: 7/1/68
DRAWN BY: [Signature]
CHECKED BY: [Signature]

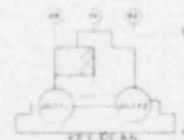
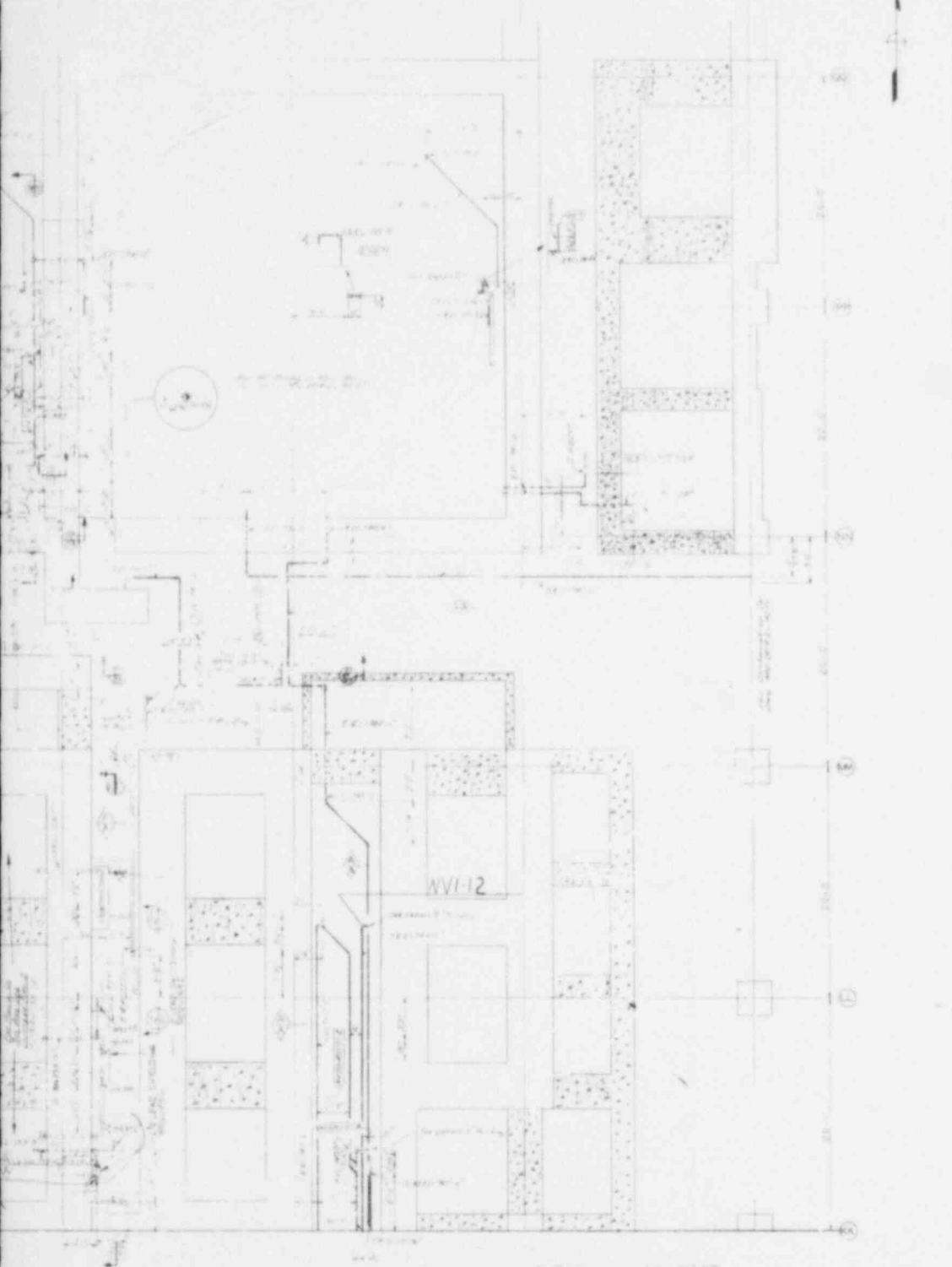


POSTULATED BREAKS IN
 NV SYSTEM
 FIG 3.15 SHEET 3 OF 9
 REV 3



1. DATE: 08/11/1981
 2. DRAWN BY: [illegible]
 3. CHECKED BY: [illegible]
 4. APPROVED BY: [illegible]

MC-114 2241-0120



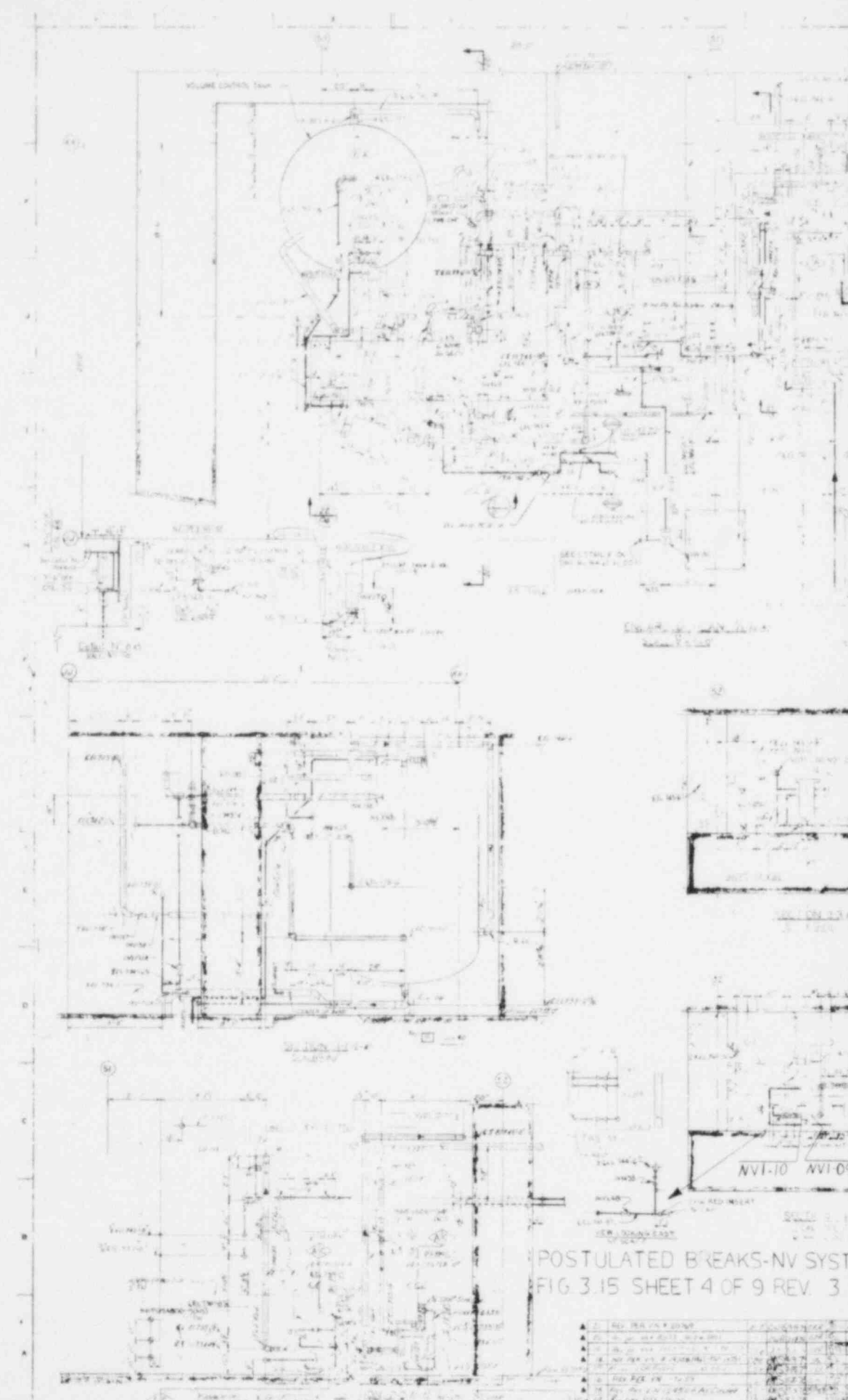
NUCLEAR SAFETY RELATED

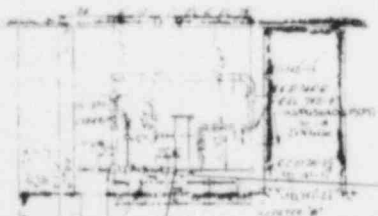
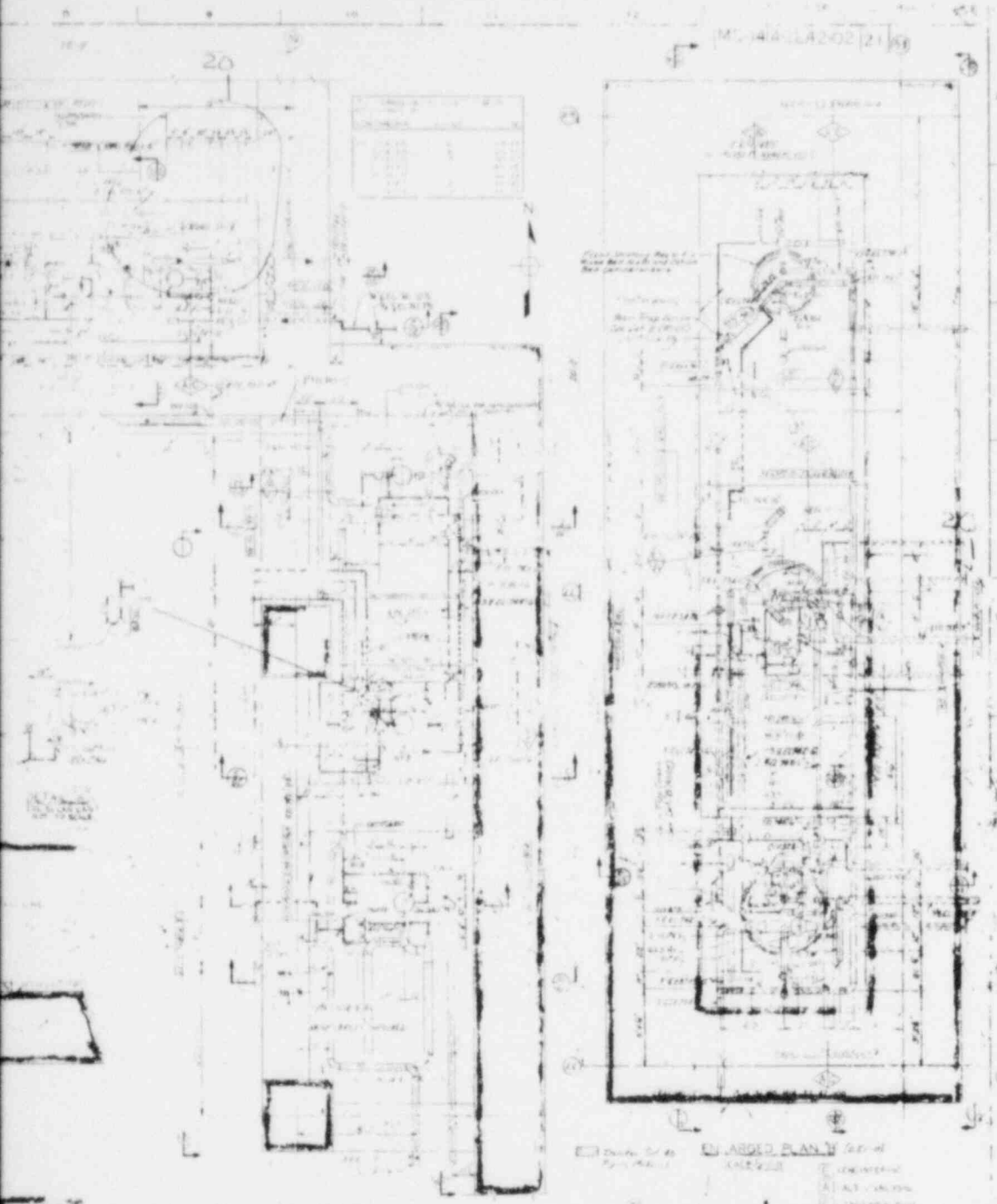
DUKE POWER COMPANY
 WILSON NUCLEAR STATION - UNITS 1 & 2

SYSTEM LAYOUT
 SYSTEM RX
 PLAN & ELEVATION 733'-0"
 AUXILIARY BUILDING

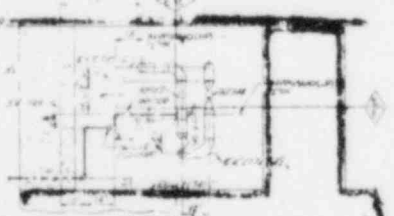
NO.	REVISION	DATE	BY	CHKD.
10	ADD CORRECTIVE WORK TO PLAN	11/15/73
9
8
7
6
5
4
3
2
1

MC-114 2241-0120





NVI-11



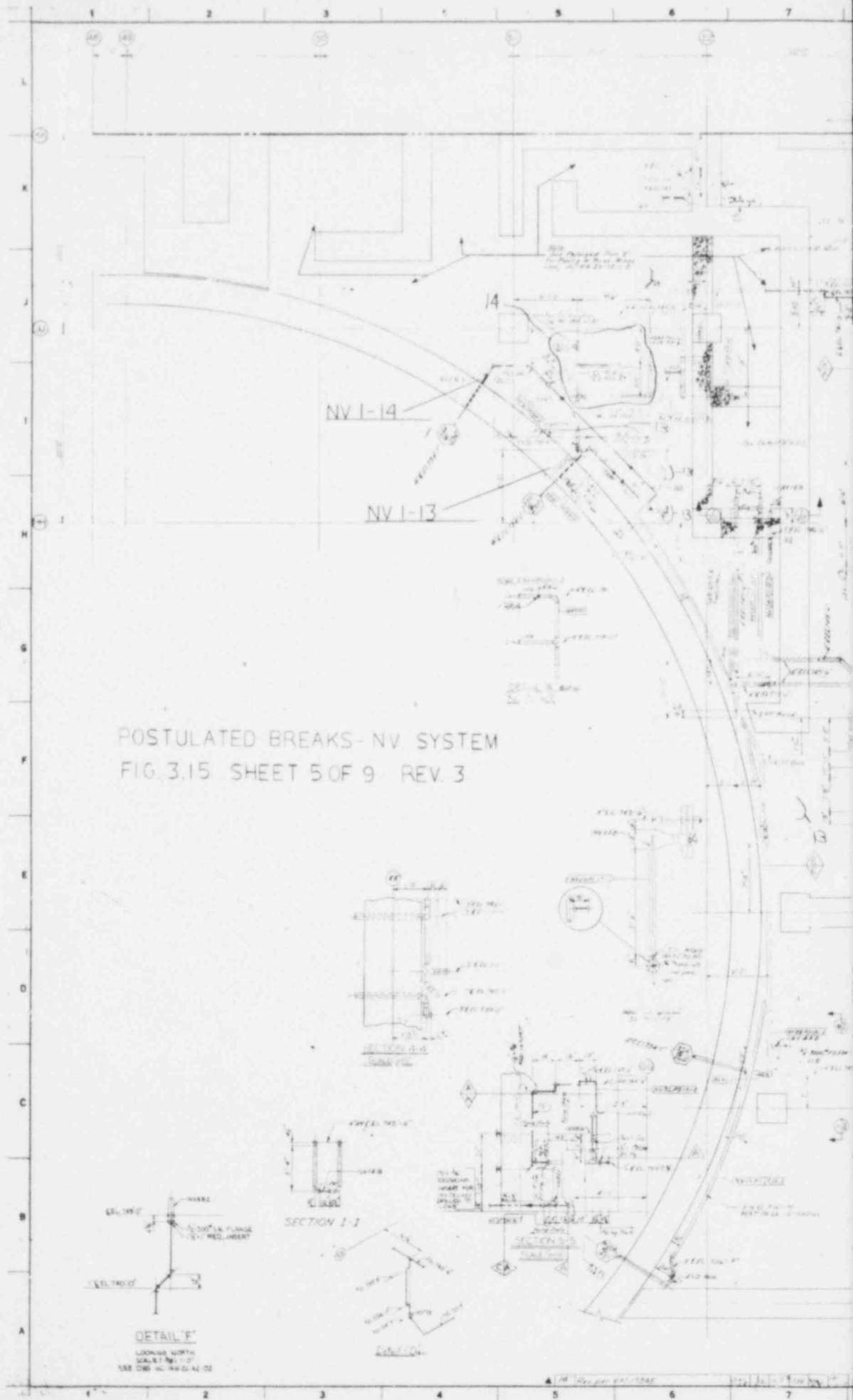
NO.	DESCRIPTION	DATE	BY	CHKD.
11	REVISED TO SHOW CHANGES TO PIPING AND INSTRUMENTATION	10/15/68	J. W. [unclear]	[unclear]
10	REVISED TO SHOW CHANGES TO PIPING AND INSTRUMENTATION	10/15/68	J. W. [unclear]	[unclear]
9	REVISED TO SHOW CHANGES TO PIPING AND INSTRUMENTATION	10/15/68	J. W. [unclear]	[unclear]
8	REVISED TO SHOW CHANGES TO PIPING AND INSTRUMENTATION	10/15/68	J. W. [unclear]	[unclear]
7	REVISED TO SHOW CHANGES TO PIPING AND INSTRUMENTATION	10/15/68	J. W. [unclear]	[unclear]
6	REVISED TO SHOW CHANGES TO PIPING AND INSTRUMENTATION	10/15/68	J. W. [unclear]	[unclear]
5	REVISED TO SHOW CHANGES TO PIPING AND INSTRUMENTATION	10/15/68	J. W. [unclear]	[unclear]
4	REVISED TO SHOW CHANGES TO PIPING AND INSTRUMENTATION	10/15/68	J. W. [unclear]	[unclear]
3	REVISED TO SHOW CHANGES TO PIPING AND INSTRUMENTATION	10/15/68	J. W. [unclear]	[unclear]
2	REVISED TO SHOW CHANGES TO PIPING AND INSTRUMENTATION	10/15/68	J. W. [unclear]	[unclear]
1	REVISED TO SHOW CHANGES TO PIPING AND INSTRUMENTATION	10/15/68	J. W. [unclear]	[unclear]

SECTION 8-6 (SIP 3)
SCALE 1/4" = 1'-0"

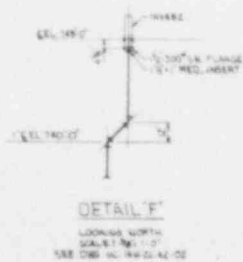
DUKE POWER COMPANY
MELHRE NUCLEAR STATION UNIT

PIPING LAYOUT
SYSTEM NV
PLANS & SECTIONS ELEVATION 73'
AUXILIARY BUILDING

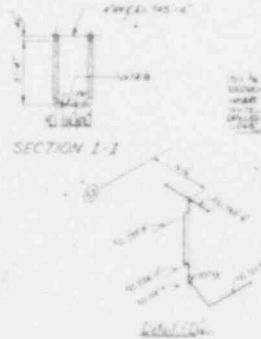
DATE: 10/15/68
DWG. NO. MC 44-224202

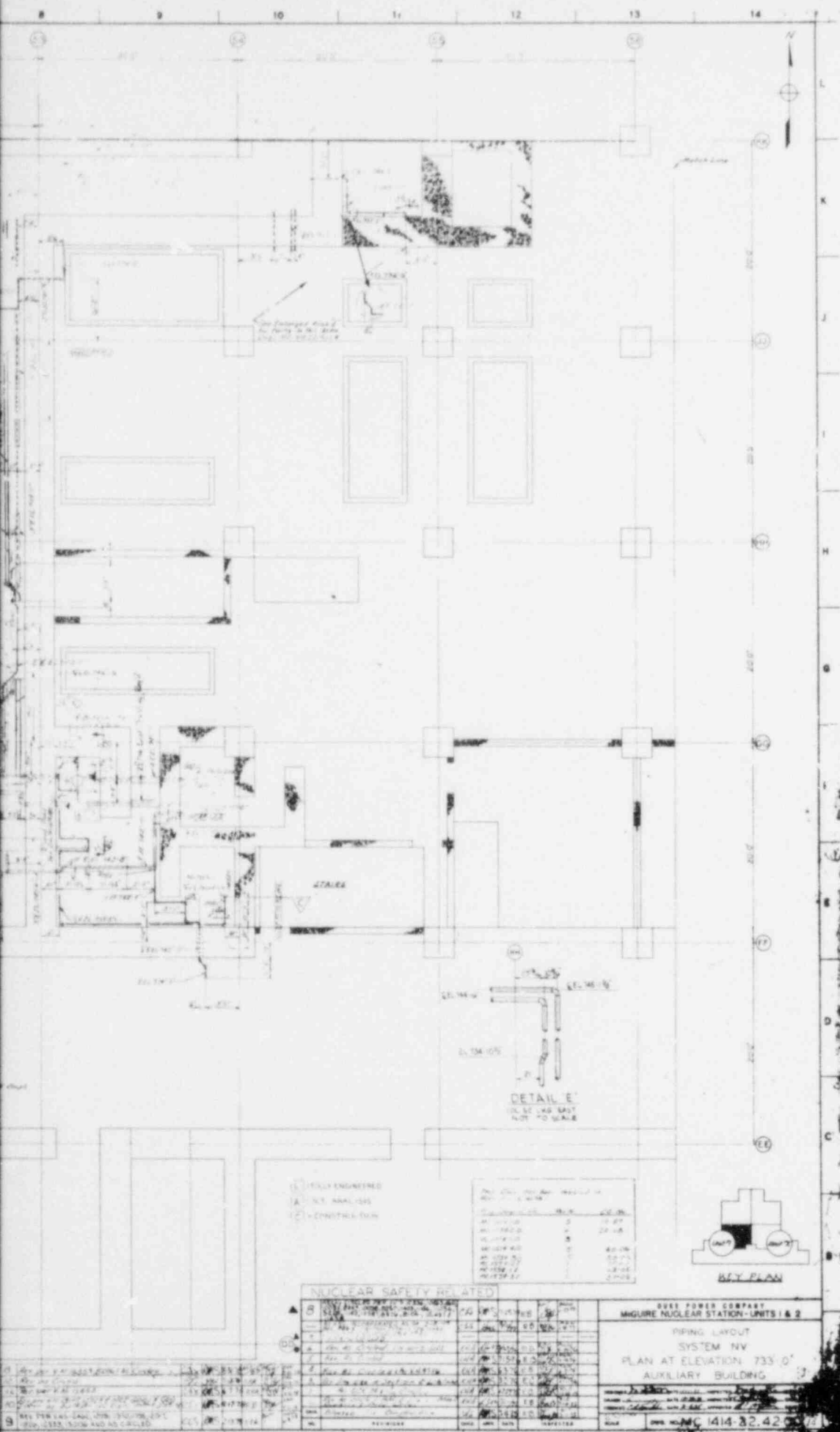


POSTULATED BREAKS - NV SYSTEM
 FIG. 3.15 SHEET 5 OF 9 REV. 3



SECTION 1-1

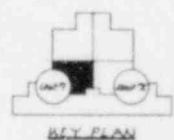




DETAIL E
PIPE LAP JOINT
1/2" = 1'0" SCALE

- (1) FULLY ENGINEERED
- (A) NOT APPROVED
- (2) ELECTRICAL WORK

NO.	DATE	DESCRIPTION	BY	CHKD.
1	10/1/68	ISSUED FOR PERMITS
2	10/15/68	REVISED PER COMMENTS
3	10/20/68	REVISED PER COMMENTS
4	10/25/68	REVISED PER COMMENTS
5	11/5/68	REVISED PER COMMENTS
6	11/15/68	REVISED PER COMMENTS
7	11/25/68	REVISED PER COMMENTS
8	12/5/68	REVISED PER COMMENTS
9	12/15/68	REVISED PER COMMENTS
10	12/25/68	REVISED PER COMMENTS



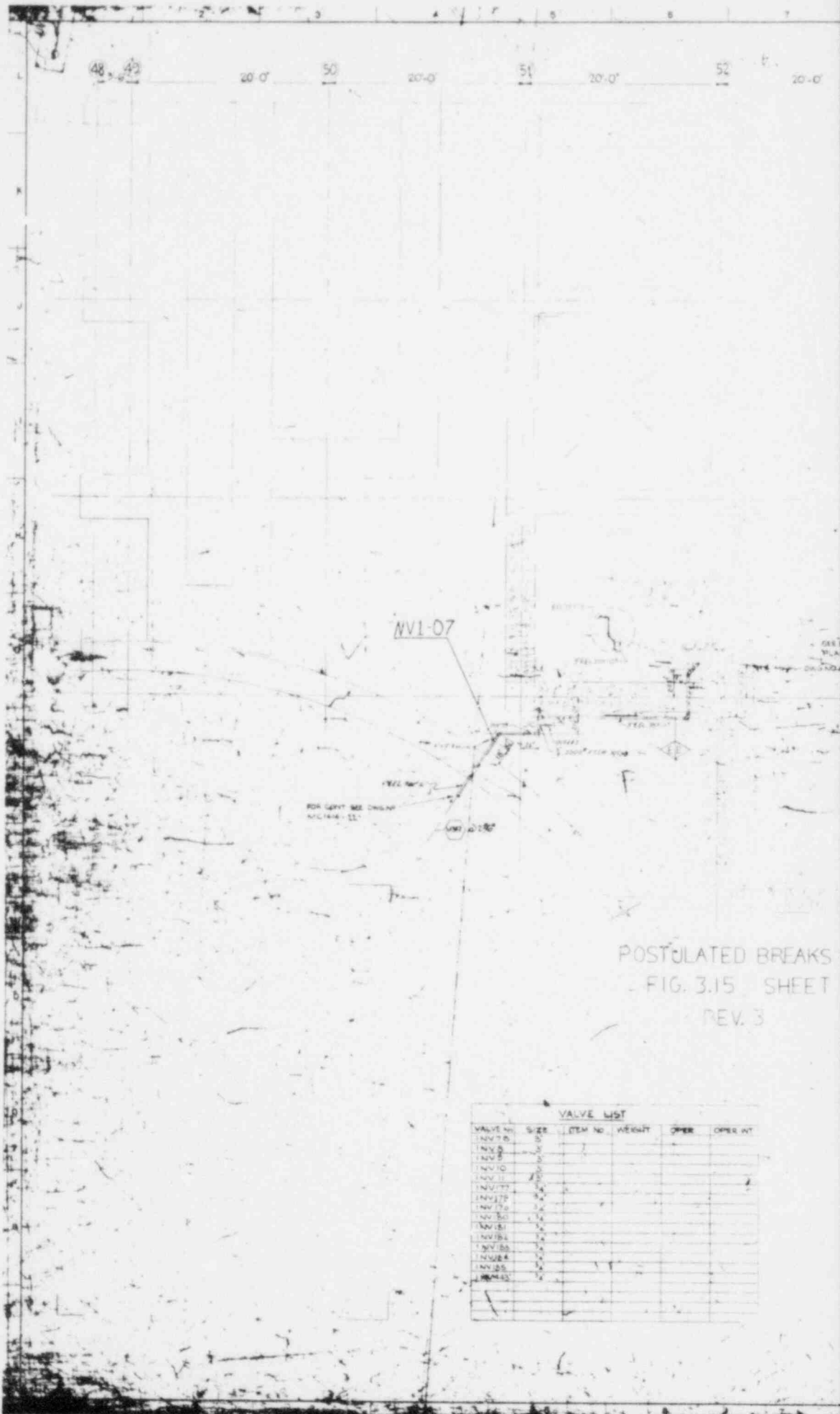
NUCLEAR SAFETY RELATED

NO.	DATE	DESCRIPTION	BY	CHKD.
1	10/1/68	ISSUED FOR PERMITS
2	10/15/68	REVISED PER COMMENTS
3	10/20/68	REVISED PER COMMENTS
4	10/25/68	REVISED PER COMMENTS
5	11/5/68	REVISED PER COMMENTS
6	11/15/68	REVISED PER COMMENTS
7	11/25/68	REVISED PER COMMENTS
8	12/5/68	REVISED PER COMMENTS
9	12/15/68	REVISED PER COMMENTS
10	12/25/68	REVISED PER COMMENTS

DUKE POWER COMPANY
MIGURE NUCLEAR STATION - UNITS 1 & 2

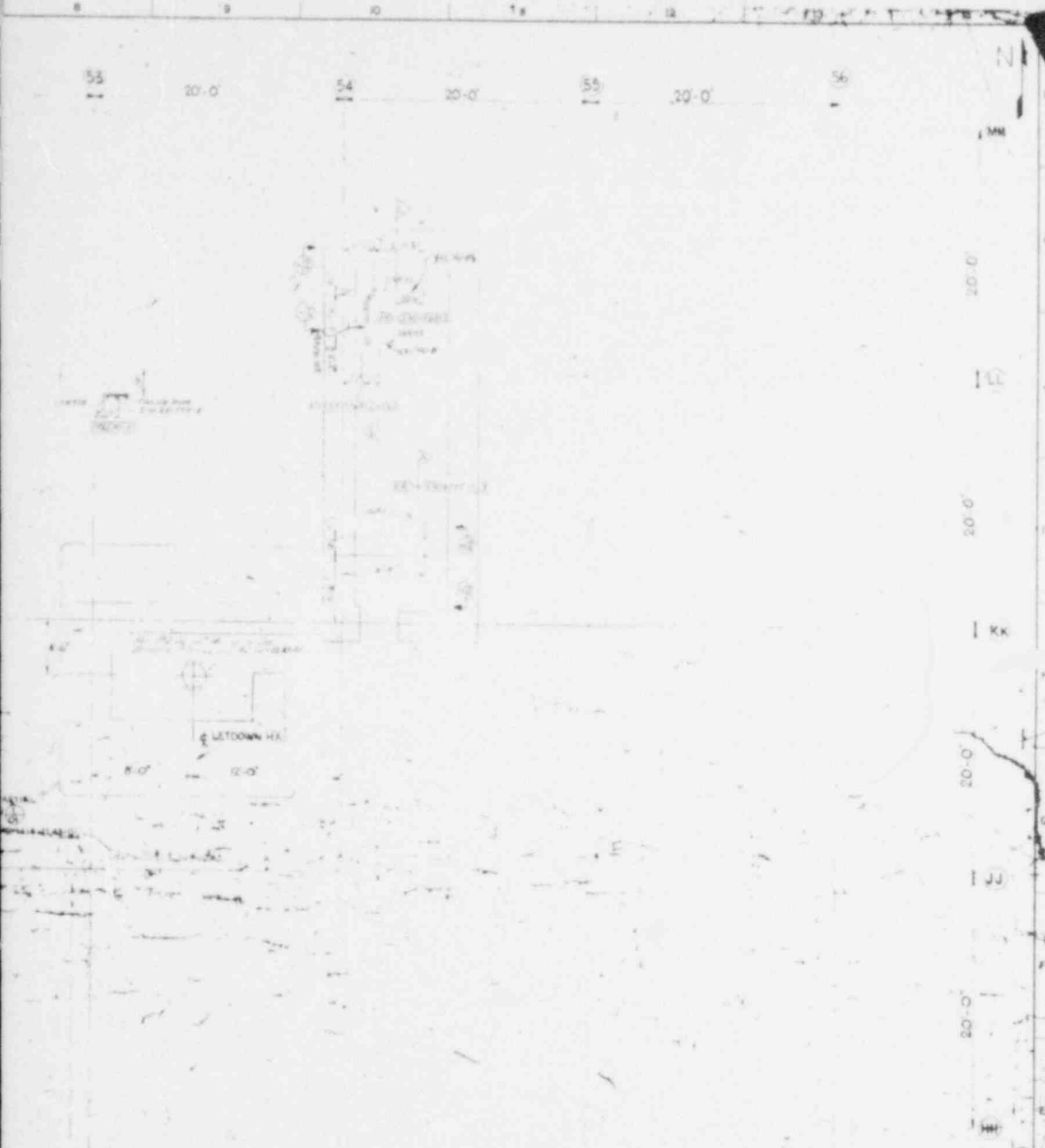
PIPING LAYOUT
SYSTEM NV
PLAN AT ELEVATION 733.0'
AUXILIARY BUILDING

Scale: ONE INCH = 14'4.2"



POSTULATED BREAKS
 FIG. 3.15 SHEET
 REV. 3

VALVE LIST					
VALVE NO.	SIZE	ITEM NO.	WEIGHT	OPER.	OPER. WT.
INV 7B	2				
INV 8	2				
INV 9	2				
INV 10	2				
INV 11	2				
INV 17	2				
INV 17D	2				
INV 18	2				
INV 18A	2				
INV 18B	2				
INV 18C	2				
INV 18D	2				
INV 18E	2				
INV 18F	2				
INV 18G	2				
INV 18H	2				
INV 18I	2				
INV 18J	2				
INV 18K	2				
INV 18L	2				
INV 18M	2				
INV 18N	2				
INV 18O	2				
INV 18P	2				
INV 18Q	2				
INV 18R	2				
INV 18S	2				
INV 18T	2				
INV 18U	2				
INV 18V	2				
INV 18W	2				
INV 18X	2				
INV 18Y	2				
INV 18Z	2				



NV SYSTEM
6 OF 9



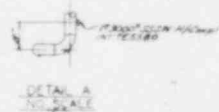
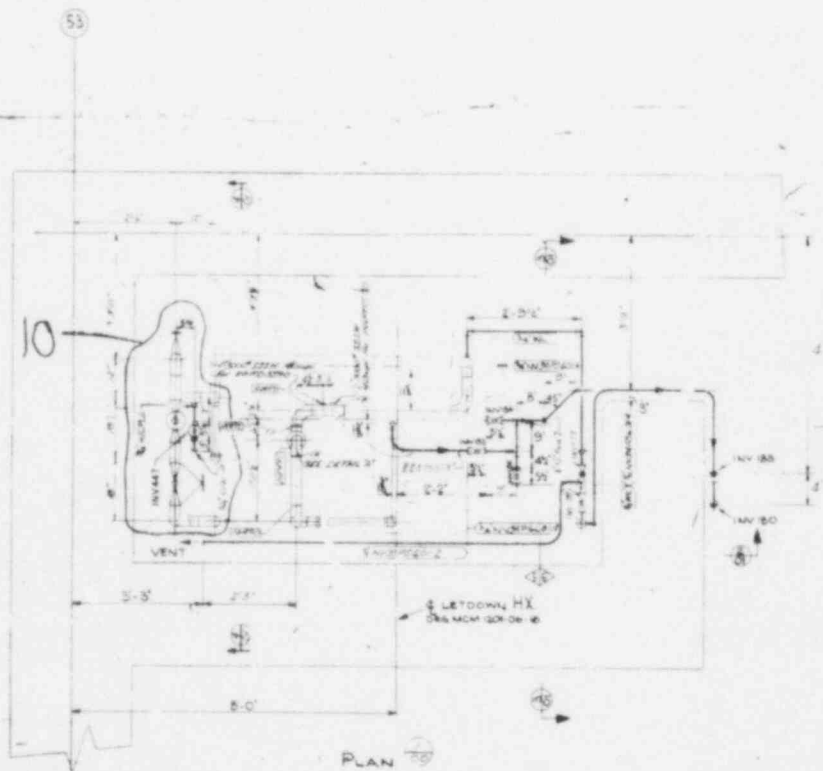
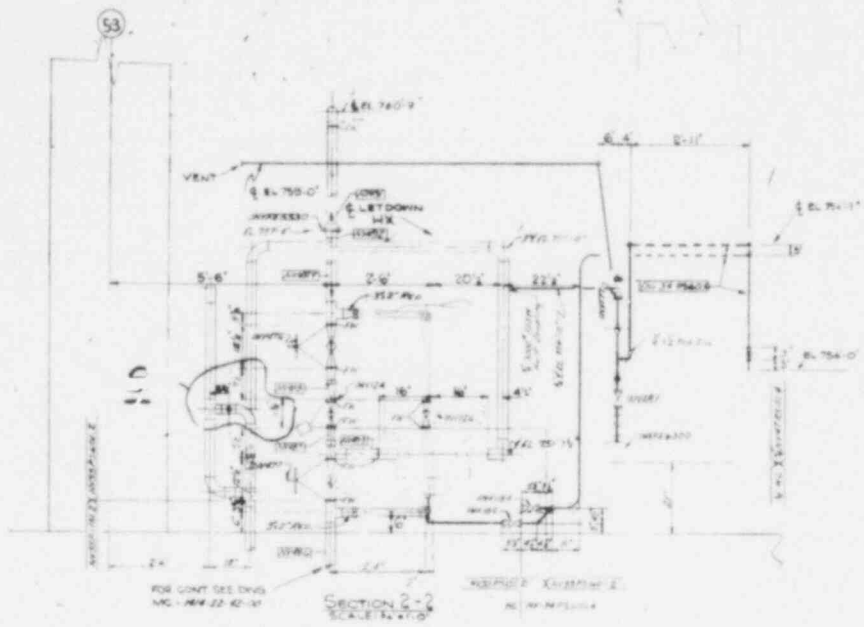
FULL ENGINEER
A. M. MULLIN

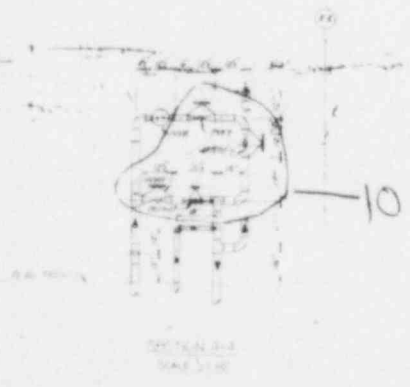
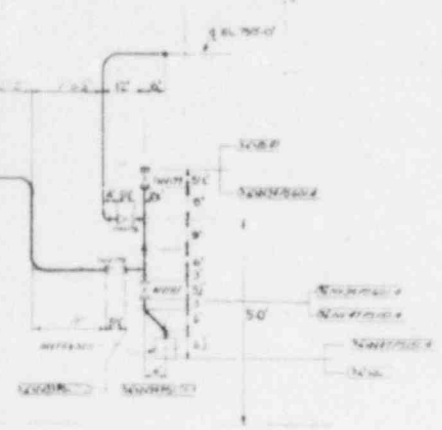
- NOTES:
1. APPLICABLE CODES ARE AS SHOWN ON SHEET 1 OF 9.
 2. FOR PIPING SPECIFICATIONS SEE SHEET 1 OF 9.
 3. CONSTRUCTION DEPT. RELEASE 50140

CHECKED FOR SAFETY BY		DATE	
1
2
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4
5
6
7
8
9
10

3. REF. FOR 1000 GPM
ANNUAL NUCLEAR PLANT
PIPING LAYOUT
SYSTEM - NV
ELEV. 50'-0" PLAN AND SECTION
AUXILIARY BUILDING







POSTULATED BREAKS IN NV SYSTEM

FIG. 3.15 SHEET 7 OF 9

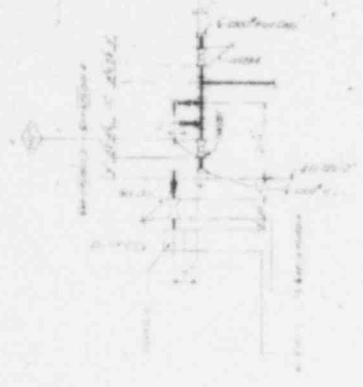
REV. 3

DESIGNED BY: FULL ENGINEERS

NO.	DESCRIPTION	DATE	BY	CHKD.
1	ISSUED FOR CONSTRUCTION	12/15/68	J. J. [unclear]	[unclear]
2	REVISED FOR [unclear]	1/15/69	[unclear]	[unclear]
3	REVISED FOR [unclear]	2/15/69	[unclear]	[unclear]
4	REVISED FOR [unclear]	3/15/69	[unclear]	[unclear]

NO.	REVISIONS	DATE	BY	CHKD.
1	ISSUED FOR CONSTRUCTION	12/15/68	J. J. [unclear]	[unclear]
2	REVISED FOR [unclear]	1/15/69	[unclear]	[unclear]
3	REVISED FOR [unclear]	2/15/69	[unclear]	[unclear]
4	REVISED FOR [unclear]	3/15/69	[unclear]	[unclear]

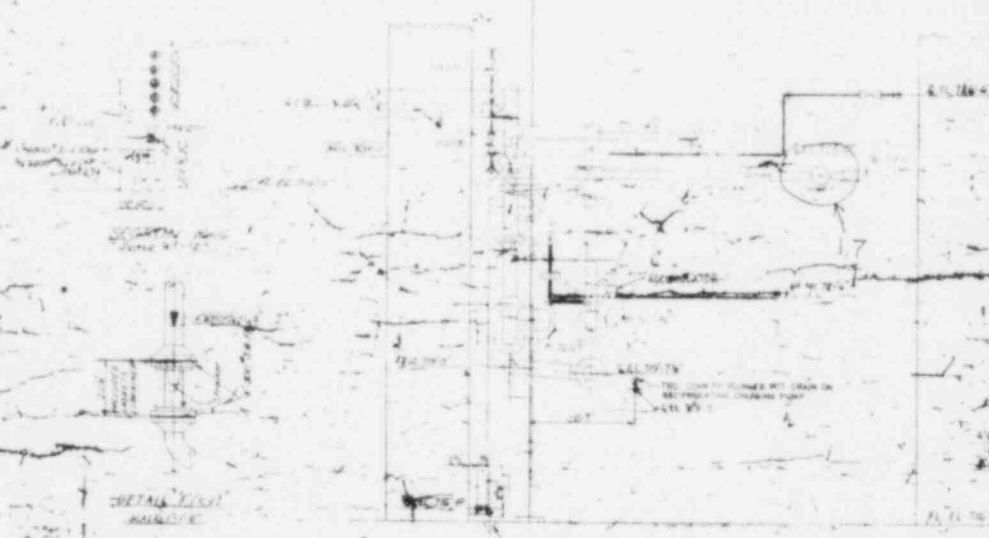
NUCLEAR SAFETY RELATED
 DUKES POWER COMPANY
 MCQUEEN NUCLEAR STATION UNIT
 PIPING LAYOUT
 SYSTEM NV
 PLAN AT ELEVATION
 AUXILIARY 101



SECTION A-A
Scale 1/2" = 1"

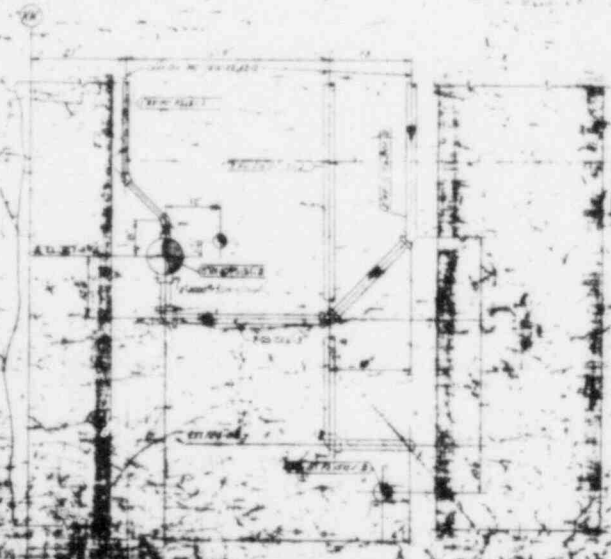


SECTION B-B
Scale 1/2" = 1"

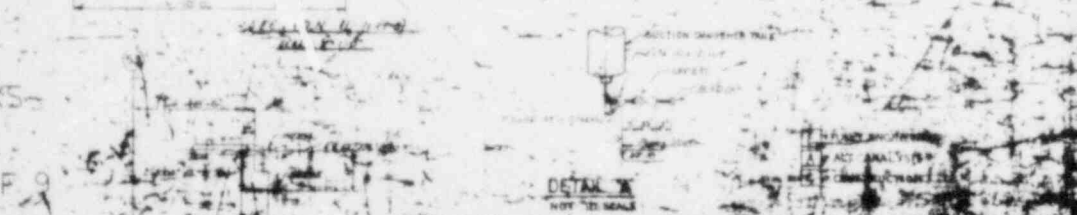
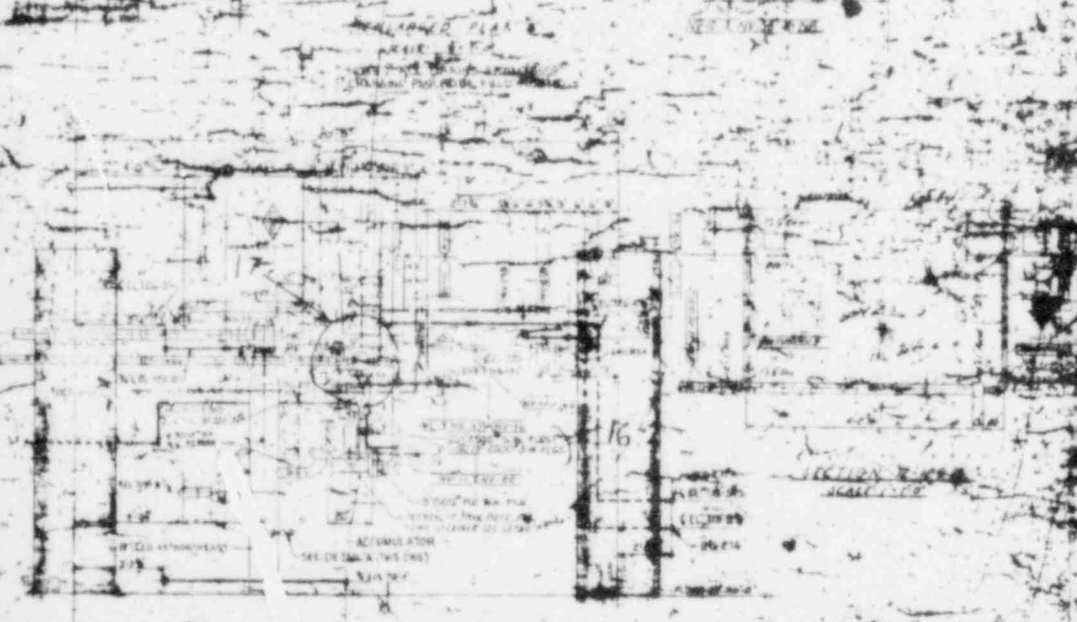
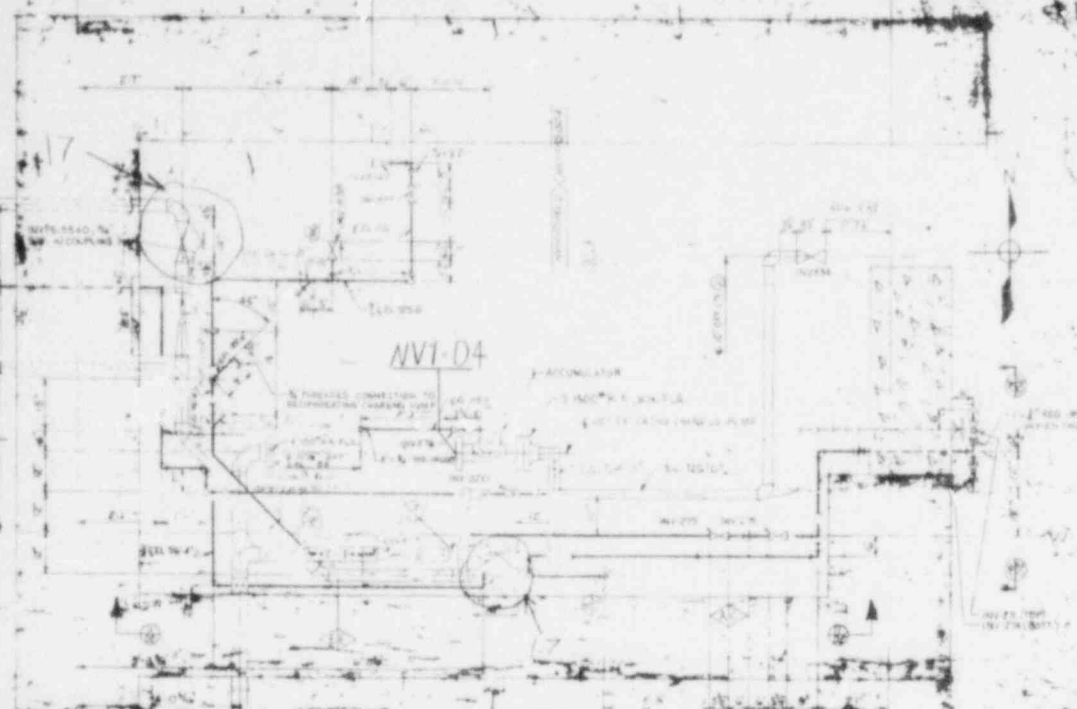


DETAIL 1
Scale 1/2" = 1"

SECTION C-C
Scale 1/2" = 1"



POSTULATED BR
= NV SYSTEM
FIG. 215 SHEET
REV. 7



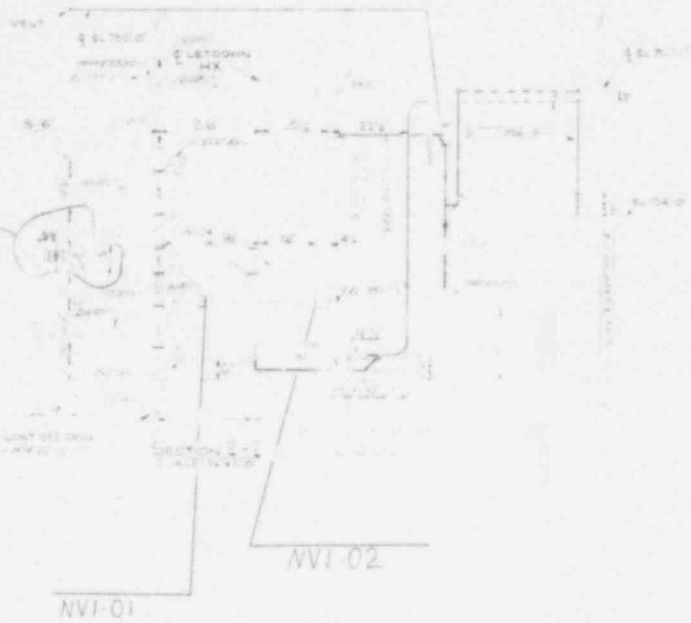
AKS
OF 9

NO.	DESCRIPTION	QTY	UNIT	REMARKS
1	VALVE BODY	1	PC	
2	VALVE STEM	1	PC	
3	VALVE SEAT	1	PC	
4	VALVE GASKET	1	PC	
5	VALVE NUT	1	PC	
6	VALVE WASHER	1	PC	
7	VALVE O-RING	1	PC	
8	VALVE PIN	1	PC	
9	VALVE SPRING	1	PC	
10	VALVE COIL SPRING	1	PC	
11	VALVE BALL	1	PC	
12	VALVE BALL SEAT	1	PC	
13	VALVE BALL SPRING	1	PC	
14	VALVE BALL PIN	1	PC	
15	VALVE BALL WASHER	1	PC	
16	VALVE BALL NUT	1	PC	
17	VALVE BALL GASKET	1	PC	
18	VALVE BALL O-RING	1	PC	
19	VALVE BALL PIN WASHER	1	PC	
20	VALVE BALL PIN NUT	1	PC	
21	VALVE BALL PIN WASHER	1	PC	
22	VALVE BALL PIN NUT	1	PC	
23	VALVE BALL PIN WASHER	1	PC	
24	VALVE BALL PIN NUT	1	PC	
25	VALVE BALL PIN WASHER	1	PC	
26	VALVE BALL PIN NUT	1	PC	
27	VALVE BALL PIN WASHER	1	PC	
28	VALVE BALL PIN NUT	1	PC	
29	VALVE BALL PIN WASHER	1	PC	
30	VALVE BALL PIN NUT	1	PC	
31	VALVE BALL PIN WASHER	1	PC	
32	VALVE BALL PIN NUT	1	PC	
33	VALVE BALL PIN WASHER	1	PC	
34	VALVE BALL PIN NUT	1	PC	
35	VALVE BALL PIN WASHER	1	PC	
36	VALVE BALL PIN NUT	1	PC	
37	VALVE BALL PIN WASHER	1	PC	
38	VALVE BALL PIN NUT	1	PC	
39	VALVE BALL PIN WASHER	1	PC	
40	VALVE BALL PIN NUT	1	PC	
41	VALVE BALL PIN WASHER	1	PC	
42	VALVE BALL PIN NUT	1	PC	
43	VALVE BALL PIN WASHER	1	PC	
44	VALVE BALL PIN NUT	1	PC	
45	VALVE BALL PIN WASHER	1	PC	
46	VALVE BALL PIN NUT	1	PC	
47	VALVE BALL PIN WASHER	1	PC	
48	VALVE BALL PIN NUT	1	PC	
49	VALVE BALL PIN WASHER	1	PC	
50	VALVE BALL PIN NUT	1	PC	

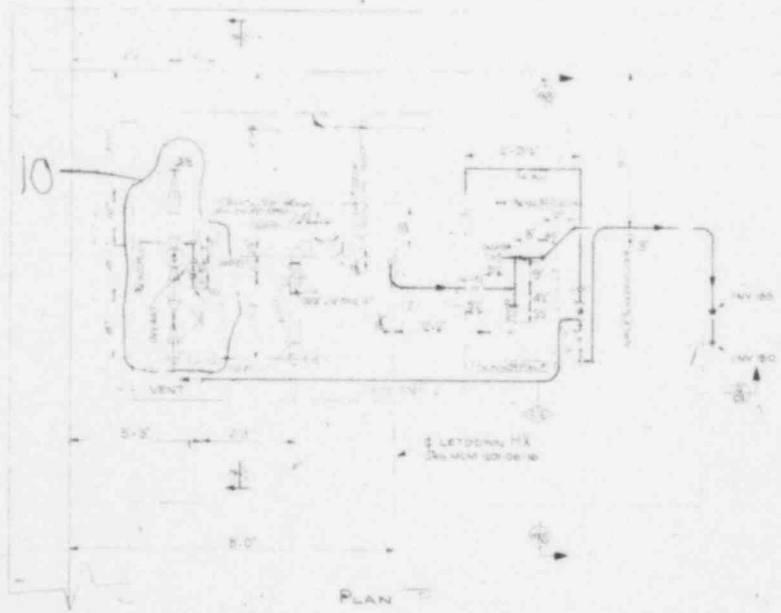
53

KK

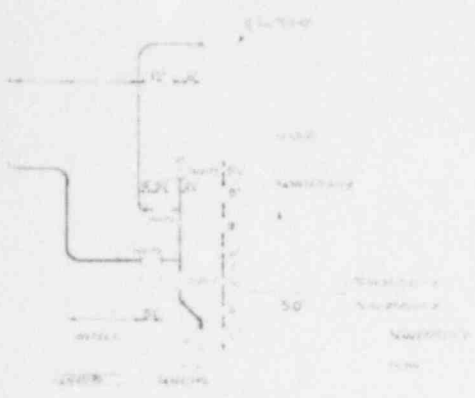
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10



PLAN

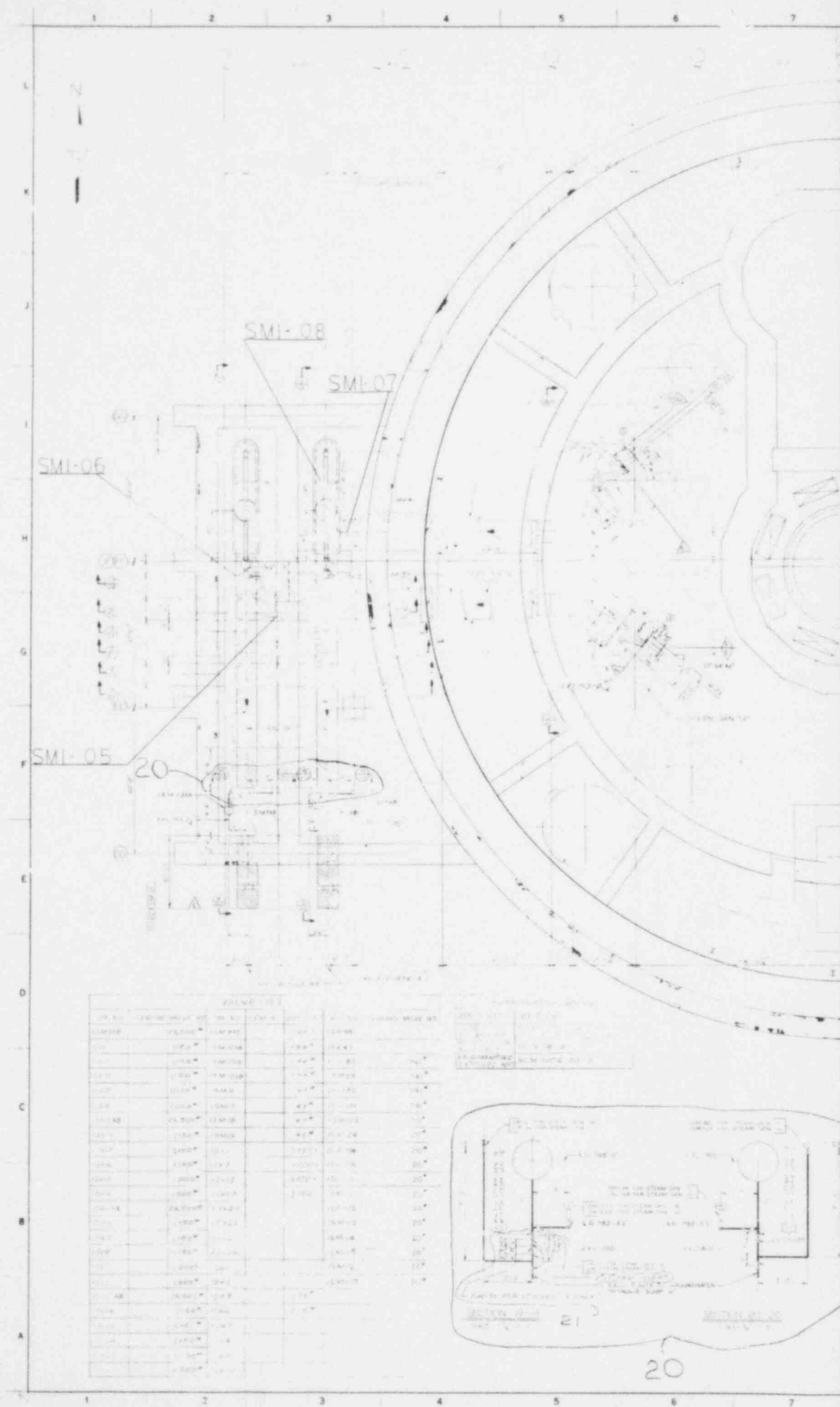


POSTULATED BREAKS IN NV SYSTEM
 FIG. 3.15 SHEET 9 OF 9 REV. 3

FALTY ENGINEERS

DUKE POWER COMPANY	
NO.	REV.
1	AS SHOWN
2	REVISED
3	REVISED
4	REVISED
5	REVISED
6	REVISED
7	REVISED
8	REVISED
9	REVISED
10	REVISED
11	REVISED
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93	REVISED
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98	REVISED
99	REVISED
100	REVISED

DUKE POWER COMPANY
 M.C.
 SYSTEM NV
 PLAN AT ELEVATION
 DWG. NO. 2450110



SMI-05

SMI-06

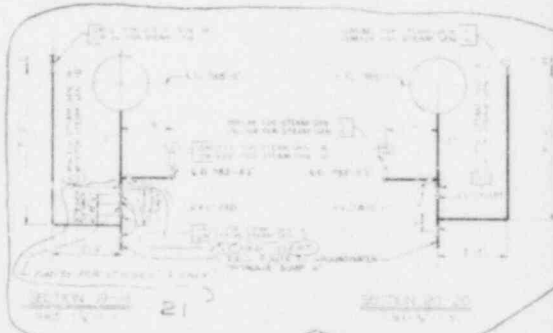
SMI-08

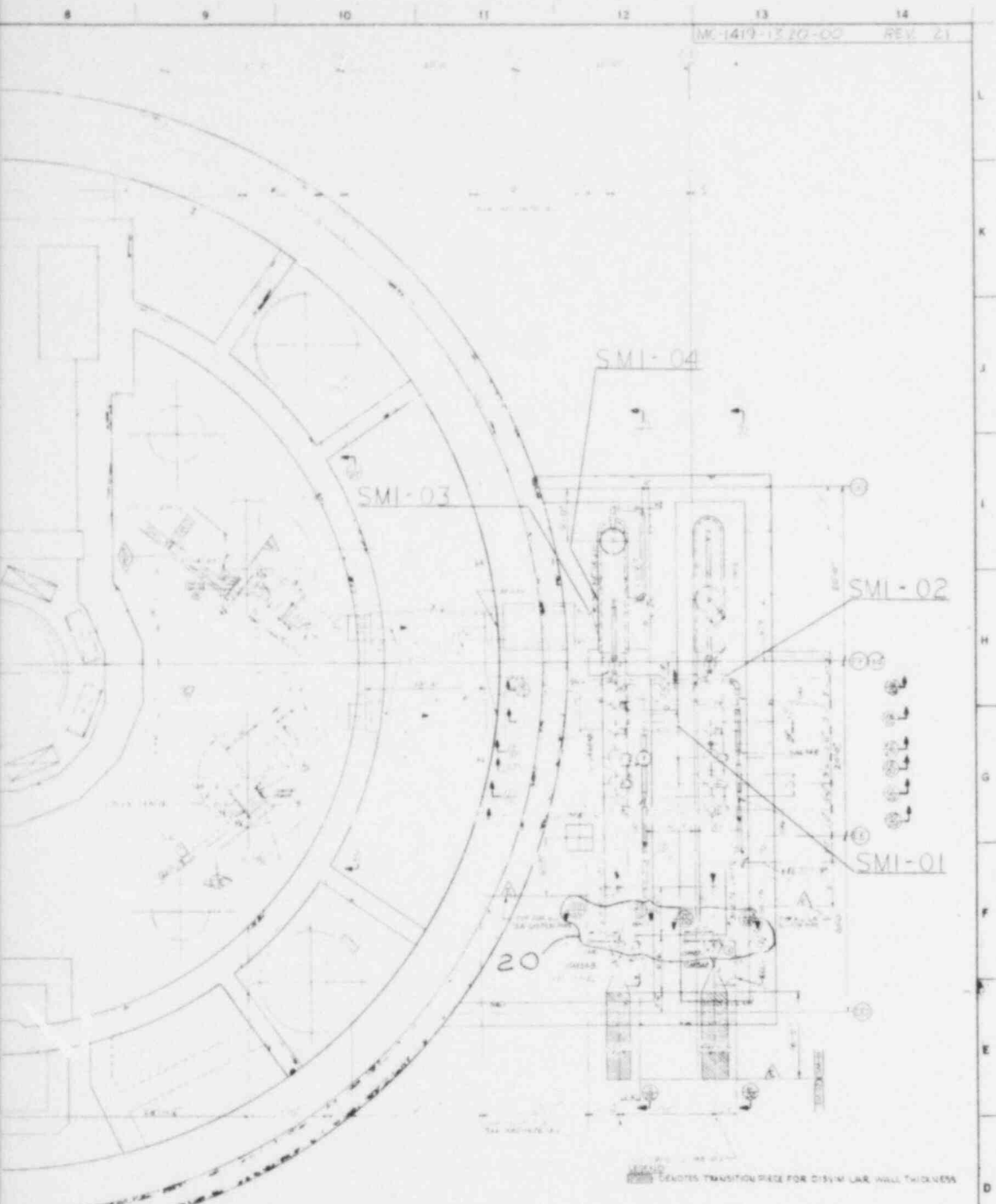
SMI-07

20

20

NO.	DESCRIPTION	UNIT	QTY	UNIT PRICE	TOTAL
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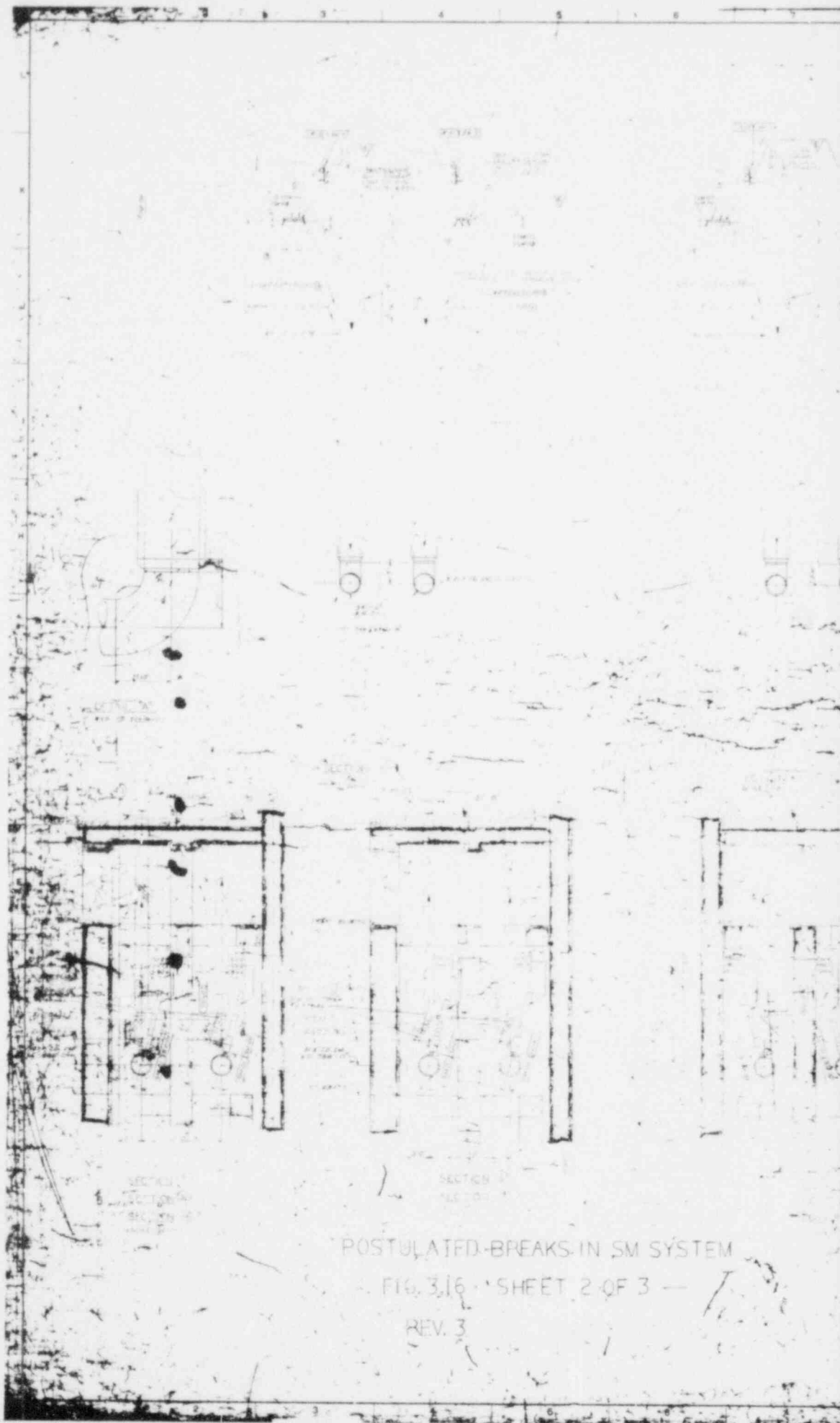


POSTULATED BREAKS SM SYSTEM
 FIG. 3.16 SHEET 10F3
 REV. 3

- 1. 1/2" (12.7mm) WELD
- 2. 1/2" (12.7mm) WELD
- 3. 1/2" (12.7mm) WELD
- 4. 1/2" (12.7mm) WELD
- 5. 1/2" (12.7mm) WELD
- 6. 1/2" (12.7mm) WELD
- 7. 1/2" (12.7mm) WELD
- 8. 1/2" (12.7mm) WELD
- 9. 1/2" (12.7mm) WELD
- 10. 1/2" (12.7mm) WELD
- 11. 1/2" (12.7mm) WELD
- 12. 1/2" (12.7mm) WELD
- 13. 1/2" (12.7mm) WELD
- 14. 1/2" (12.7mm) WELD
- 15. 1/2" (12.7mm) WELD
- 16. 1/2" (12.7mm) WELD
- 17. 1/2" (12.7mm) WELD
- 18. 1/2" (12.7mm) WELD
- 19. 1/2" (12.7mm) WELD
- 20. 1/2" (12.7mm) WELD

THIS DWG. HAS BEEN DESIGNED IN ACCORDANCE WITH		
FLOW DIAGRAM	REV. NO.	CHANGE ORDER NO.
MC-1419-13	21	MC-1419-13-001
MC-1419-13	20	MC-1419-13-000
MC-1419-13	19	MC-1419-13-000
DESIGNED BY: [REDACTED]		
CHECKED BY: [REDACTED]		
APPROVED BY: [REDACTED]		
DATE: [REDACTED]		
SCALE: [REDACTED]		
SHEET NO. [REDACTED]		
TOTAL SHEETS: [REDACTED]		
PROJECT: [REDACTED]		
DRAWING NO. MC-1419-13 20 00 21		

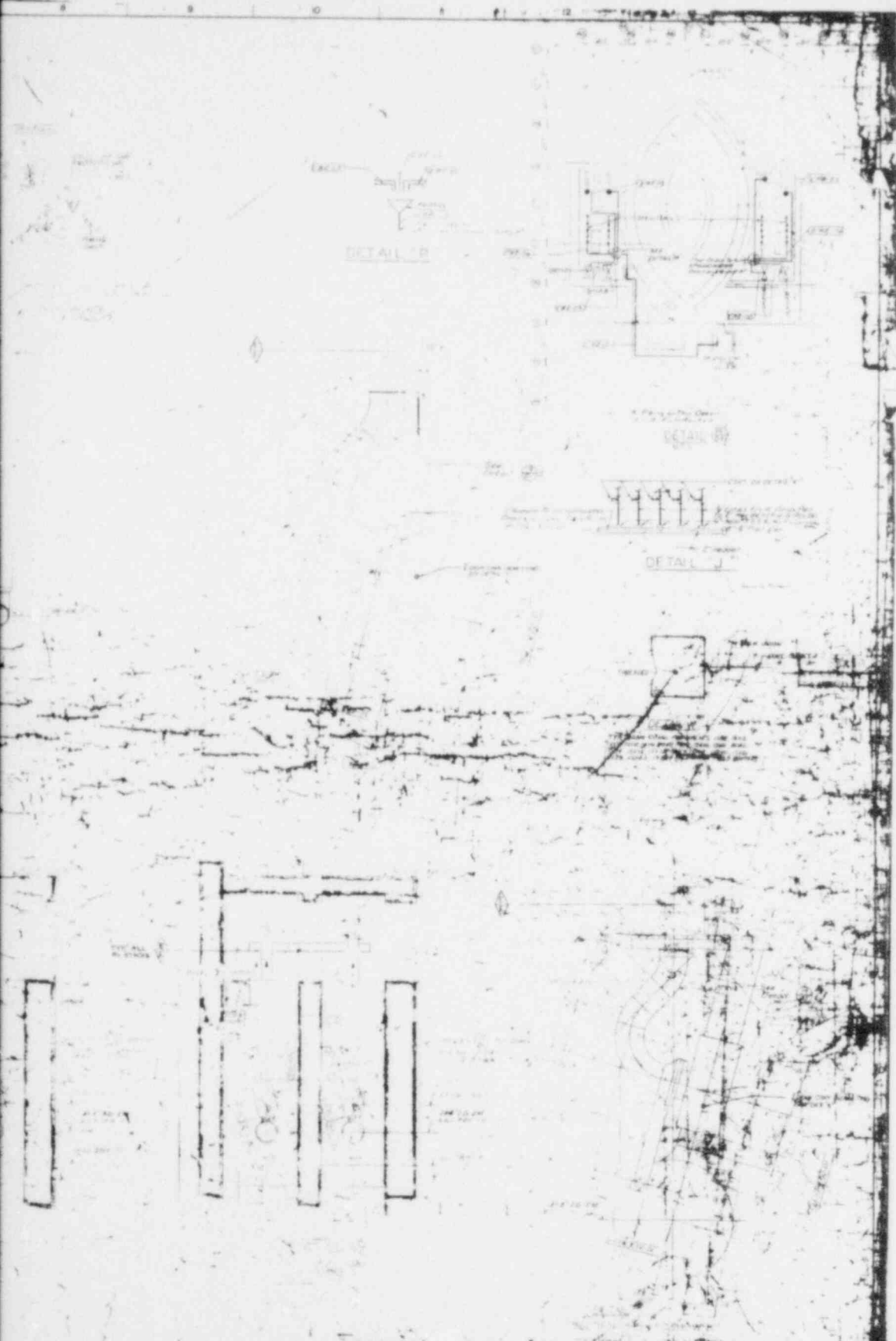
NO.	DESCRIPTION	DATE	BY	CHKD.	APP.
1	ISSUED FOR CONSTRUCTION	10/1/00	[REDACTED]	[REDACTED]	[REDACTED]
2	ISSUED FOR CONSTRUCTION	10/1/00	[REDACTED]	[REDACTED]	[REDACTED]
3	ISSUED FOR CONSTRUCTION	10/1/00	[REDACTED]	[REDACTED]	[REDACTED]
4	ISSUED FOR CONSTRUCTION	10/1/00	[REDACTED]	[REDACTED]	[REDACTED]
5	ISSUED FOR CONSTRUCTION	10/1/00	[REDACTED]	[REDACTED]	[REDACTED]
6	ISSUED FOR CONSTRUCTION	10/1/00	[REDACTED]	[REDACTED]	[REDACTED]
7	ISSUED FOR CONSTRUCTION	10/1/00	[REDACTED]	[REDACTED]	[REDACTED]
8	ISSUED FOR CONSTRUCTION	10/1/00	[REDACTED]	[REDACTED]	[REDACTED]
9	ISSUED FOR CONSTRUCTION	10/1/00	[REDACTED]	[REDACTED]	[REDACTED]
10	ISSUED FOR CONSTRUCTION	10/1/00	[REDACTED]	[REDACTED]	[REDACTED]
11	ISSUED FOR CONSTRUCTION	10/1/00	[REDACTED]	[REDACTED]	[REDACTED]
12	ISSUED FOR CONSTRUCTION	10/1/00	[REDACTED]	[REDACTED]	[REDACTED]
13	ISSUED FOR CONSTRUCTION	10/1/00	[REDACTED]	[REDACTED]	[REDACTED]
14	ISSUED FOR CONSTRUCTION	10/1/00	[REDACTED]	[REDACTED]	[REDACTED]
15	ISSUED FOR CONSTRUCTION	10/1/00	[REDACTED]	[REDACTED]	[REDACTED]
16	ISSUED FOR CONSTRUCTION	10/1/00	[REDACTED]	[REDACTED]	[REDACTED]
17	ISSUED FOR CONSTRUCTION	10/1/00	[REDACTED]	[REDACTED]	[REDACTED]
18	ISSUED FOR CONSTRUCTION	10/1/00	[REDACTED]	[REDACTED]	[REDACTED]
19	ISSUED FOR CONSTRUCTION	10/1/00	[REDACTED]	[REDACTED]	[REDACTED]
20	ISSUED FOR CONSTRUCTION	10/1/00	[REDACTED]	[REDACTED]	[REDACTED]



POSTULATED BREAKS IN SM SYSTEM

FIG. 3.16 SHEET 2 OF 3

REV. 3



REVISIONS		DATE		BY		CHECKED	
1	AS SHOWN						
2							
3							
4							
5							
6							
7							
8							
9							
10							

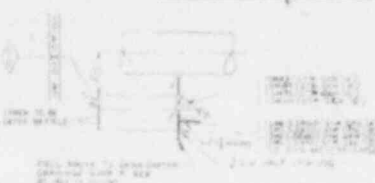
TRING BAY
 SH SYSTEM
 SECTIONS
 FACTORY BLDG

POSTULATED BREAKS - SM SYSTEM

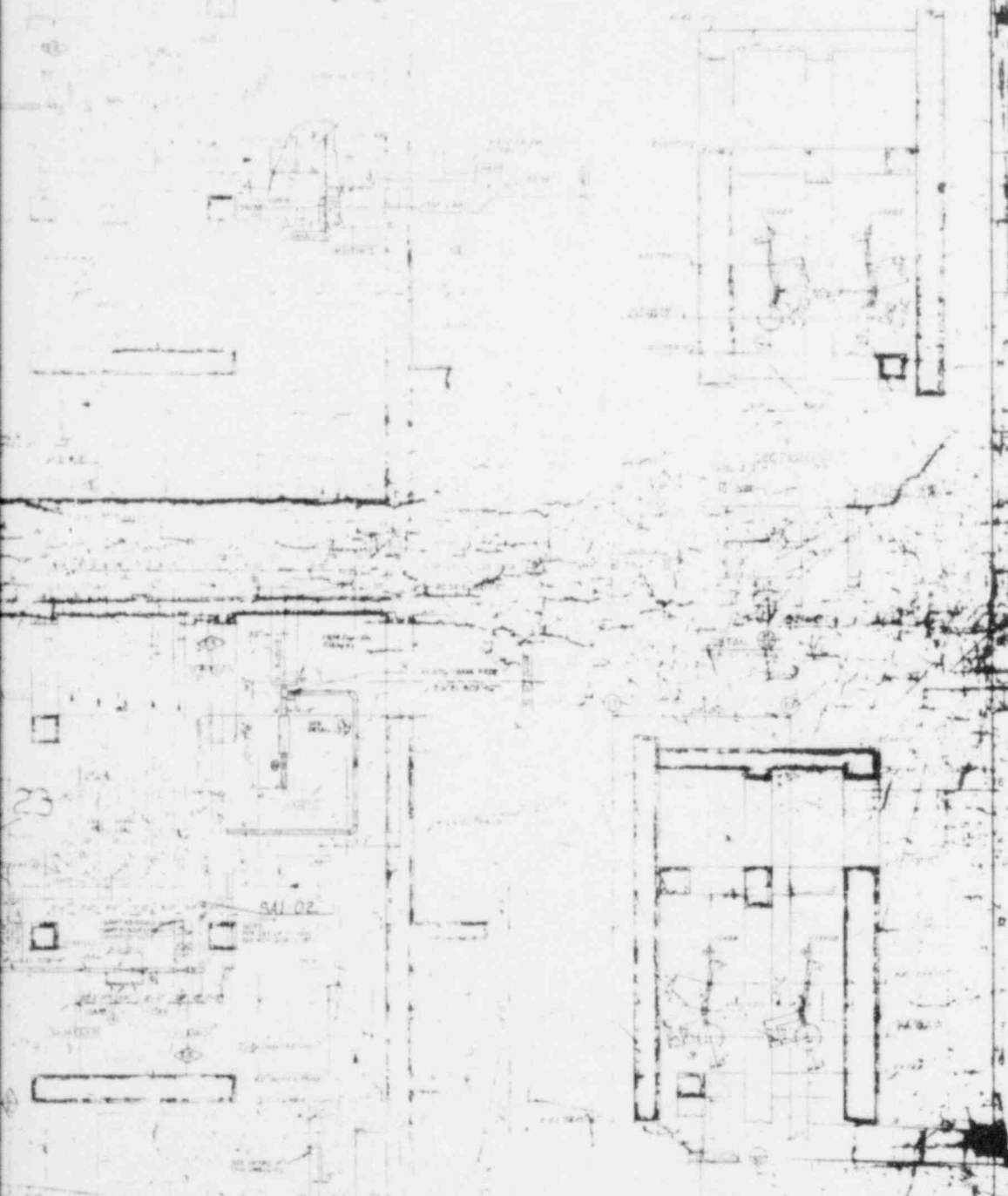
FIG. 3.16 SHEET 3 OF 3

REV. 3





DETAIL 2



NUCLEAR SAFETY RELATED
SHELL POWER BOARD
INGUIRE NUCLEAR STATION UNIT
PIPING LAYOUT
SM SYSTEM
SECTIONS
REACTOR

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4.0 SYSTEMS AND EQUIPMENT EVALUATION

4.1 General

Protection is provided to those plant structures and equipment essential to insure that a safe plant shutdown can be accomplished and maintained, that consequences of the postulated pipe rupture can be mitigated, and that the offsite dose consequences are less than the 10CFR100 allowable limits for a rupture of the piping system under consideration. This requirement is met by preventing unacceptable levels of damage to essential structures and equipment by one or more of the following acceptable methods, listed in their order of preference. Choice of method gives due consideration to other plant design and system requirements.

- a) Separation and remote location of fluid system piping from essential structures and equipment.
- b) Structural enclosure of the fluid system piping or alternatively, enclosure of the essential equipment.
- c) Provision of system-redundant design features separated, or otherwise protected, from the effects of the postulated pipe rupture; or additional protection features such as restraints and barriers.
- d) Design of essential structures and equipment to withstand the effects of the postulated pipe rupture.
- e) Addition of guard piping for the main purpose of diverting or restricting blowdown flow.
- f) In areas where none of the above can be met, or where unacceptable, more severe problems may be created, augmented inservice

inspection may be used on a case by case basis with NRC approval to reduce the probability of failure to acceptable levels and not postulate the failure.

4.2 Protection From Rupture of High-Energy Piping Systems

4.2.1 High Energy Piping Systems Physically Separated From Essential Systems and Structures

For the purpose of assuring compliance with the separation requirements of Section 4.1, breaks are postulated in accordance with Section 3.2; and it is clearly demonstrated by review of piping layout that the effects of pipe whip, jet impingement, compartment pressurization, environmental conditions, and flooding associated with the escape of the contained fluids cannot cause any damage to essential structures and systems. In this regard, the following assumptions are utilized in the evaluation of damage to essential structure and systems.

a) Whipping Pipe and Jet Impingement Interactions

- 1) A whipping pipe or jet is assumed not to cause failure of other pipes of equal or greater size and equal or greater thickness. Smaller and thinner pipes are assumed to encounter unacceptable damage upon impact. A whipping pipe or jet is considered capable of developing through-wall leakage cracks in larger nominal pipe sizes with thinner wall thicknesses, except where experimental or analytical data for the expected range of impact energies demonstrate the capability to withstand the impact without failure.

- 2) For purposes of evaluating the mechanical integrity effects of the impact of a whipping pipe or jet on piping components (i.e., pumps, valves, flanges, etc.), piping component pressure boundaries are considered to be pipe of the same size and wall thickness as the main process piping to which the components are attached. Bodies of valves (including check valves) are assumed not to fail upon impact by a smaller whipping pipe or jet from a smaller pipe.
 - 3) An active component is assumed to be incapable of performing its active function following impact by a whipping pipe or jet, unless proved otherwise by detailed impact analysis. External valve operators are assumed to be incapable of performing their intended function following impact by a whipping pipe or jet.
 - 4) Electrical and instrument cables are assumed severed upon impact by a whipping pipe or jet, unless it can be demonstrated otherwise.
 - 5) If pipe supports are damaged by whipping pipes or jets, the target pipe supported by the hangers is assumed to remain supported upon the loss of one hanger.
- b) Environmental Interactions
- 1) Structures are assumed failed if their design pressure is exceeded, unless integrity can be demonstrated.

- 2) Electrical cabling and instrument wiring are assumed to be severed upon exceeding the environmental qualifications of the cabling or wiring, unless it can be demonstrated otherwise.
- 3) An active component is assumed incapable of performing its active function upon experiencing environmental conditions exceeding the environmental qualifications of the active component.

4.2.2 Piping Systems Enclosed Within Structures

For the purpose of assuring compliance with the requirements of Section 4.1 for piping systems within enclosures, breaks are postulated in accordance with Section 3.2, and it is demonstrated by analysis that such enclosure is adequately designed to prevent damage to essential structures and equipment from the effects of pipe whip, jet impingement, pressurization of the enclosure compartment, environmental conditions, and flooding associated with the escape of the contained fluid. Piping restraints within the enclosure may be accounted for in limiting the effects of the postulated pipe rupture.

4.2.3 Other High-Energy Piping Systems

For the purpose of assuring compliance with the requirements of Section 4.1 for piping systems neither physically separated from essential structures and equipment nor enclosed, breaks are postulated in accordance with Section 3.2; and it is demonstrated by analysis that the effects of pipe whip, jet impingement, compartment pressurization, environmental conditions, and

flooding associated with the escape of the contained fluids cannot cause unacceptable damage to essential structures and equipment.

4.3 Protection From Through-Wall Cracks In Moderate-Energy Piping

For the purpose of assuring compliance with the requirements of Section 4.1, through-wall cracks are postulated in moderate-energy systems in accordance with Section 3.2; and it is demonstrated by analysis that environmental conditions and flooding associated with the escape of the contained fluids cannot cause unacceptable levels of damage to essential structures and equipment. In this regard, the following assumptions are utilized in the evaluation of damage to essential structures and systems.

- a) Electrical cabling and instrument wiring are assumed to be severed upon exceeding the environmental qualifications of the cabling or wiring, unless it can be demonstrated otherwise.
- b) An active component is assumed incapable of performing its active function upon experiencing environmental conditions exceeding the environmental qualifications of the active component.

4.4 Protection Criteria And Assumptions

If separation, remote location, and enclosure of fluid piping or essential components and systems do not prevent all possible damage resulting from a pipe rupture, then the following plant protection criteria are utilized in determining which systems, structures, or components must be protected to assure a safe plant shutdown.

- a) Plant Operating Conditions
 - 1) Power Level - At the time of the postulated pipe rupture,

the plant is assumed to be in the normal mode of plant operation in which the piping under investigation experiences the maximum conditions of pressure and temperature. In cases where this mode is full power operation, the power level assumed is that assumed in the evaluation of the loss-of-coolant accident, steamline break accident, or feedwater line break accident, in Chapter 15 of the FSAR.

- 2) Offsite Power - If the pipe rupture results in a loss-of-coolant accident, steam line break accident, or feedwater line break accident, a loss of offsite power is assumed to occur subsequent to the pipe rupture.
 - 3) Seismic Loadings equivalent to either the Safe Shutdown Earthquake (SSE) or the Operating Basis Earthquake (OBE), as appropriate, will be used in the analysis of piping, equipment, protective devices, etc.
- b) Consideration is given to the potential for a random single failure of an active component subsequent to the postulated pipe rupture. Where the postulated piping rupture is assumed to occur in one of two or more redundant trains of a dual-purpose moderate-energy essential system, i.e., one required to operate during normal plant conditions as well as to shut down the reactor and mitigate the consequences of the piping rupture, single failures of components in the other train or trains of that system only are not assumed, provided the system is designed to seismic Category I standards, is powered from both offsite and onsite sources, and is constructed, operated, and inspected to quality assurance, testing, and inservice

inspection standards appropriate for nuclear safety systems.

- c) In the event of a postulated rupture in the piping in one unit, safe reactor shutdown of the affected unit cannot preclude the capability for safe shutdown of the reactor of the unaffected unit.
- d) Minimum essential systems performance is provided as required for the type of break.
- e) The conditions within the control room or any other location where manual action is required to assure safe shutdown to the cold condition is such as to assure habitability and comply with the requirements of General Design Criterion 19.
- f) The effects of high-energy pipe ruptures are not allowed to result in offsite doses in excess of 10CFR100 allowable limits.
- g) A pipe rupture cannot cause a steam or feedwater line break.
- h) All non-LOCA breaks (except Steam and Feedwater Line breaks) are allowed to damage the non-LOCA portion of a single train of an ESF System, provided that unit shutdown can be achieved.
- i) All non-LOCA breaks (excluding Steam and Feedwater Line breaks) are allowed to damage any non-LOCA, non-Essential lines (except Steam and Feedwater Lines).
- j) A pipe rupture in one train of a redundant essential system or a pipe rupture which damages one train of a redundant essential system cannot result in damage to the opposite train of that system or any other essential system.
- k) A pipe rupture in a non-Seismic system (Duke System Piping Class E, G, H) cannot result in damage to an essential system.

- l) Steam and feed line ruptures are allowed to damage steam and feed lines, respectively, of the same steam generator, provided that the aggregate break size does not exceed the applicable maximum break size considered in the safety analysis.
- m) Steam and Feedwater Line breaks can damage any non-LOCA lines except required Essential System Lines.
- n) If two lines are connected to each other and are schematically the same line, then a rupture of one of these lines is allowed to damage the other.
- o) The capability for automatic and manual shutdown of the reactor shall be maintained.
- p) The capability for the removal of decay heat shall be maintained.
- q) High energy piping will only have sufficient energy to cause critical cracks in piping of larger nominal pipe size but thinner wall thickness. A critical crack is not deemed to significantly inhibit the functionality of a piping system.

4.5 Essential Systems

This section describes the systems normally employed for safe shutdown purposes. It goes on to enumerate the shutdown capability required for each event associated with the failures dealt with in this evaluation. Logic diagrams are given in Figure 4-1 through Figure 4-6. These diagrams show logic for turbine trip, main and auxiliary feedwater pump start/trip, reactor trip, reactor coolant pump start/trip, main steam isolation valves, safeguards actuation, feedwater isolation, and makeup control.

4.5.1 Normal Shutdown

The systems necessary for normal, controlled safe shutdown of the reactor are those systems associated with the major functions in both the primary and secondary sides of the Nuclear Steam Supply System. There are no individually identifiable safe shutdown systems per se. However, prescribed procedures for normally securing and maintaining the plant in a safe condition will be instituted by appropriate alignment of selected fluid and control systems. The system functions required to be aligned for maintaining normal, controlled safe shutdown of the reactor are the minimum number that will:

- 1) Prevent the reactor from achieving criticality.
- 2) Provide an adequate heat sink such that design and safety limits are not exceeded.

The designation of systems that can be used for a safe normal shutdown depends on identifying those systems which provide the following capabilities for maintaining a safe shutdown condition:

- 1) Reactor coolant boration.
- 2) Adequate supply of auxiliary feedwater.

3 |

All systems required for a safe shutdown associated with these functions have been designed in accordance with the single failure criteria, as stated in Section 4.4. The following subsections identify systems which are required for safe

shutdown of the reactor in the specific faulted conditions which this report is analyzing; these are conditions precipitated by the rupture of piping systems outside containment.

4.5.2 Emergency Shutdown

An emergency shutdown may be caused by any natural or accidental event of infrequent occurrence. This includes related consequences which affect the plant operations and require the use of other than preferred systems or Engineered Safety Feature (ESF) Systems to bring the reactor to a safe shutdown condition. In the context of this report, an emergency shutdown is also considered to be precipitated by a pipe rupture that is of sufficient magnitude that a plant shutdown is deemed necessary even though normal shutdown systems can be used to affect a reactor shutdown. All events are analyzed independently and are not assumed to occur simultaneously. In this analysis, loss of offsite power is assumed to occur in those events which cause Protection System actuation effecting a plant trip. In this case, loss of function of onsite a-c power (emergency diesels) and batteries must be prevented.

4.5.2.1 Emergency Shutdown With A Main Steam Line Rupture

The equipment necessary for a safe shutdown of the reactor is the same for any pipe break location on the main steam line. The effect of any main steam break on plant shutdown is the loss of one steam generator for reactor decay heat removal immediately after reactor trip.

For a large steam line break, the following must be available to accomplish safety functions:

- 1) Safety injection to pump borated water into the core and, thereby, limit the core power transient following the break.
- 2) Isolation of main feedwater to the steam generators to limit the Reactor Coolant System (RCS) cooldown.
- 3) Closure of the main steam isolation valves to limit RCS cooldown and reverse flow.
- 4) Auxiliary feedwater is required to dissipate reactor decay heat. In the event of a concurrent loss of offsite power, at least one of the auxiliary feed pumps would be required.

In order to cool the plant down to the Residual Heat Removal System temperature and pressure, auxiliary feedwater must be available and the steam generator power-operated relief valves must be operable.

4.5.2.2 Emergency Shutdown with a Feedwater Line Rupture

A feedwater rupture between the Containment and the feedwater check valve is considered to be the worst-case feedwater rupture because of the complete blowdown of one steam generator in addition to almost unrestricted flow from the feedwater pumps. For this rupture, the following must be available to accomplish their safety functions:

- i) Safety injection to pump borated water into the core and, thereby, limit the core power transient following the break.
- 2) Feedwater to the intact steam generators.
- 3) Closure of main steam isolation valves.

In order to cool the plant down to the Residual Heat Removal System operating temperature and pressure, auxiliary feedwater from at least one auxiliary feedwater pump must be available and the steam generator power-operated relief valves must be operable.

For a large break between the feedwater pump and the main feedwater check valve, the feed line check valve will prevent water or steam release from any of the steam generators through the break. A large break at this point is, thus, essentially a loss of normal feedwater. In this case, the equipment that must be available to accomplish the safety function consists of the Auxiliary Feedwater System and the intact portion of the Feedwater System; safety injection is not required. This case covers all lesser Feedwater and Condensate System high energy line breaks.

4.5.2.3 Emergency Shutdown With An Auxiliary Feedwater Pump Steam Supply Line Rupture

A rupture of the auxiliary feedwater pump steam supply line between the main steam line and the normally closed piston-operated valves, shown in Figure 3.7, is considered a less severe main steam line rupture, and its

necessary emergency shutdown equipment is discussed in Section 4.5.2.1. After the rupture, Auxiliary Feedwater can be provided by one of the motor-driven Auxiliary Feedwater Pumps. The check valves in each steam supply line limit the blowdown from this rupture to a single steam generator.

Based on the fact that the steam is supplied to the turbine driver only during test conditions, and upon the following considerations, the probability of a rupture in this system between the motor-operated valves and the auxiliary feedwater pump turbine driver is considered to be extremely remote.

- 1) High level of system quality control.
- 2) Periodic inspection.
- 3) Low usage factor.
- 4) Short time the system exceeds 200° and 275 psig annually.

Thus, the piping downstream of the isolation valves is considered moderate energy based on total time of operation.

4.5.2.4 Emergency Shutdown With A Steam Generator Blowdown Line Rupture

A steam generator blowdown line rupture outside the Containment would not cause a reactor or turbine trip. In this case, availability of offsite power is assumed and a "normal" plant shutdown can be effected.

For a steam generator blowdown line rupture, the affected line could be isolated and an evaluation of the severity of the leak or rupture would determine the need to shut down the reactor. If needed, the shutdown would use normal shutdown procedures.

4.5.2.5 Emergency Shutdown With A CVCS Letdown Line Rupture

For a break in the letdown line between the Containment and the letdown heat exchanger, the following must be available:

- 1) Charging plus either normal makeup or makeup from the primary makeup water storage tank, to maintain liquid inventory in the RCS until the break is identified and the letdown line isolated.
- 2) Letdown line Containment isolation valves.
- 3) Auxiliary feedwater, to cool the plant down to the conditions necessary to initiate operation of the RHRS.

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4.5.2.6 Emergency Shutdown With A CVCS Charging Line Rupture

For a full area break in the charging line between the Containment and the charging pumps, the following must be available:

- 1) Charging (from safety injection pipe) plus either normal makeup from the primary makeup water storage tank, to maintain liquid inventory in the RCS.

- 2) Auxiliary feedwater, to cool the plant down to the conditions necessary to initiate operation of the RHRS.

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4.5.2.7 Emergency Shutdown With An Auxiliary Feedwater Line Rupture

The Auxiliary Feedwater System is an ESF System. It is not only required to mitigate the consequence of accidents, eg, loss of coolant, feedwater line break, etc., but it is also required as a means of dissipating the energy from the RCS during periods when the main heat sink (main condenser dump and Main Feedwater System) is unavailable; for example, during a blackout situation.

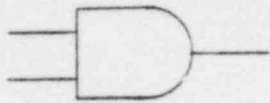
A pipe rupture in the high energy portion of the Auxiliary Feedwater System, between the main feedwater line and the first check valve is considered a less severe main feedwater line rupture. The necessary emergency shutdown equipment is discussed in Section 4.5.2.2. Following pipe rupture in one of the Auxiliary Feedwater Pump discharge lines, feedwater could be provided by one of the remaining two auxiliary feedwater pumps or through the normal feedwater system assuming offsite power availability.

LOGIC SYMBOLS

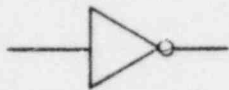
LOGIC FUNCTION



OR



AND



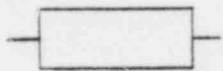
INVERTER - NOT FUNCTION



TIME DELAY



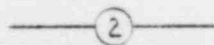
ADMINISTRATIVE PROCEDURE
CONTROL SWITCH (TYPICAL)



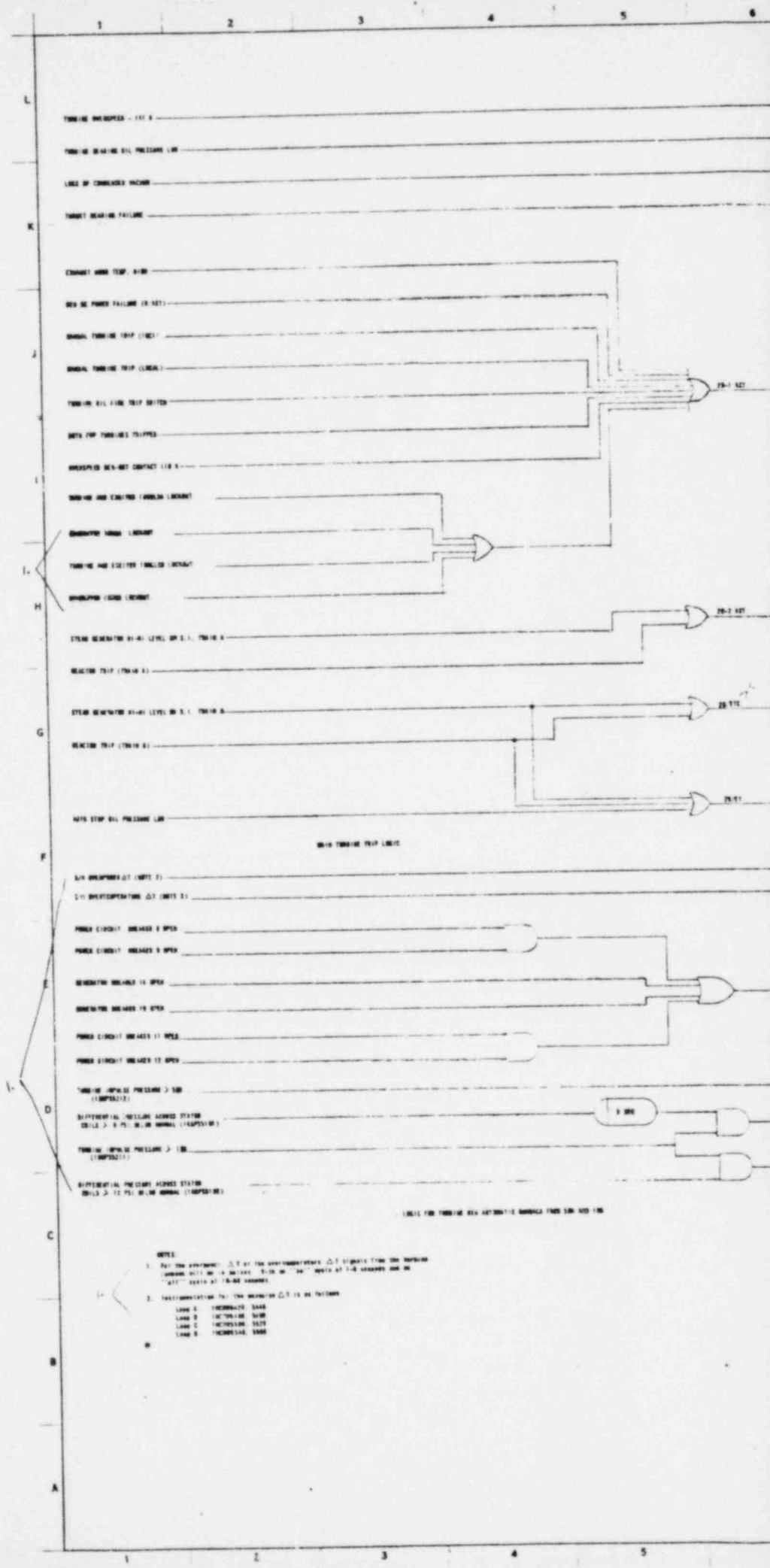
MISCELLANEOUS CONTROL
FUNCTION OR INFORMATION



INDICATES CONTROL SIGNAL
TO RESPECTIVE PUMP, FAN,
VALVE, ETC.



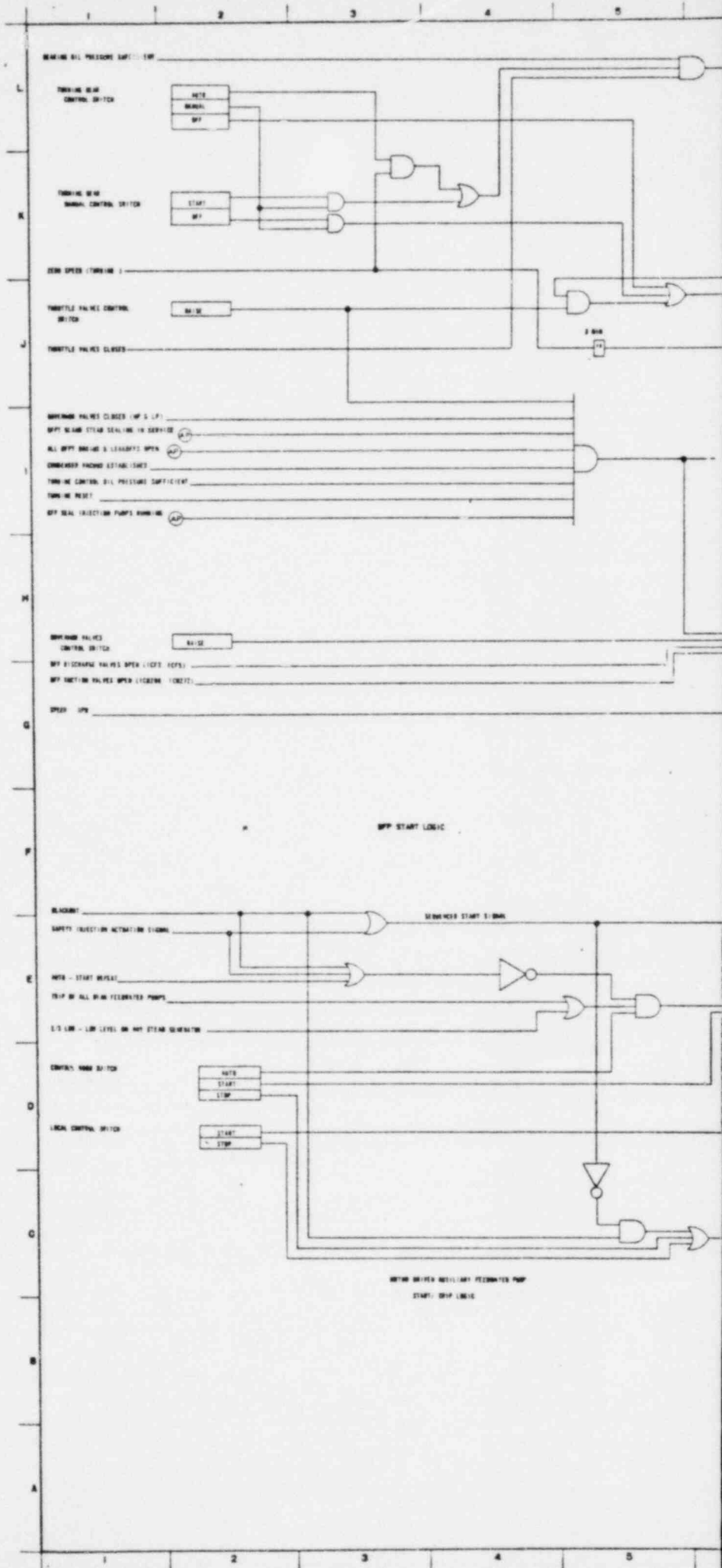
INDICATES SIGNAL TO ALL
PUMPS, FANS, OR DEVICES
OF A GIVEN TYPE

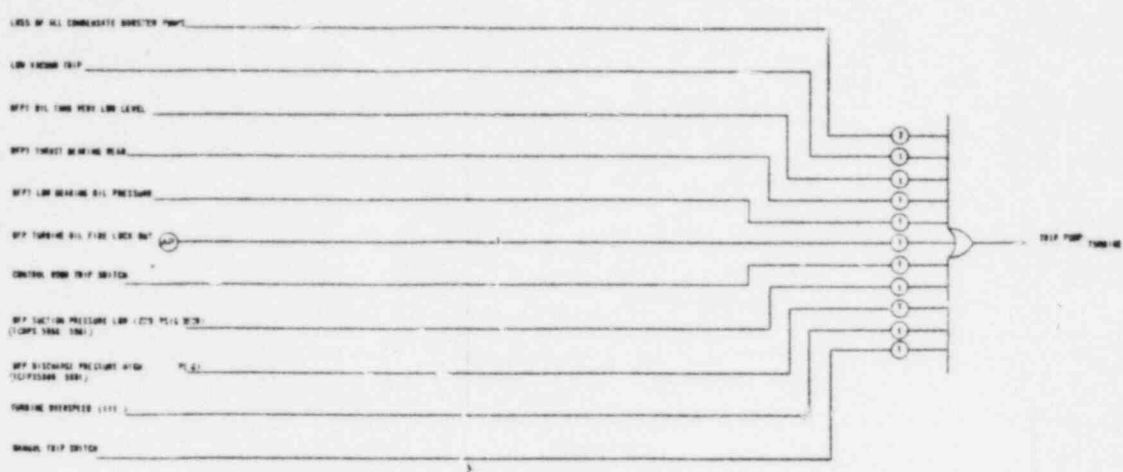
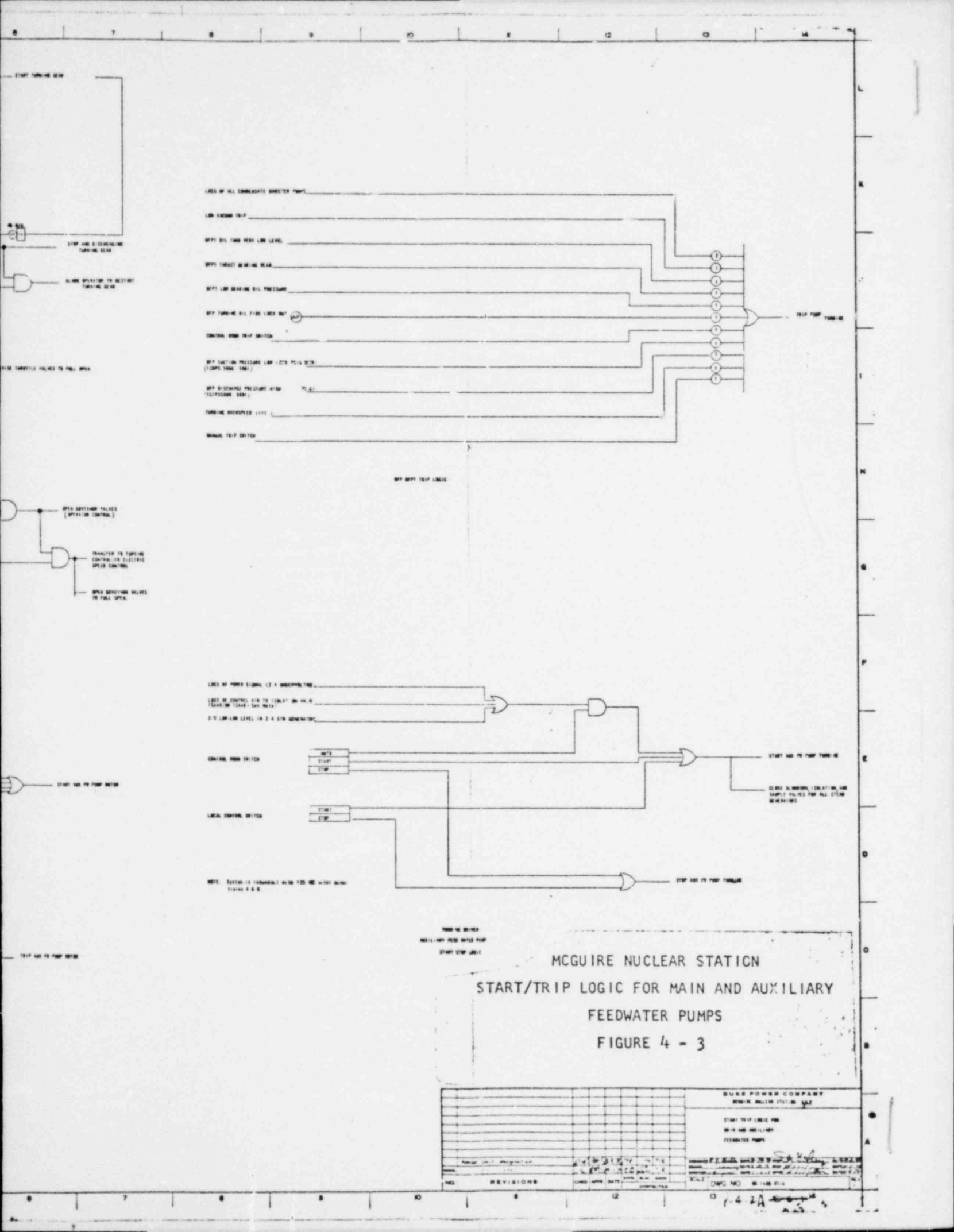


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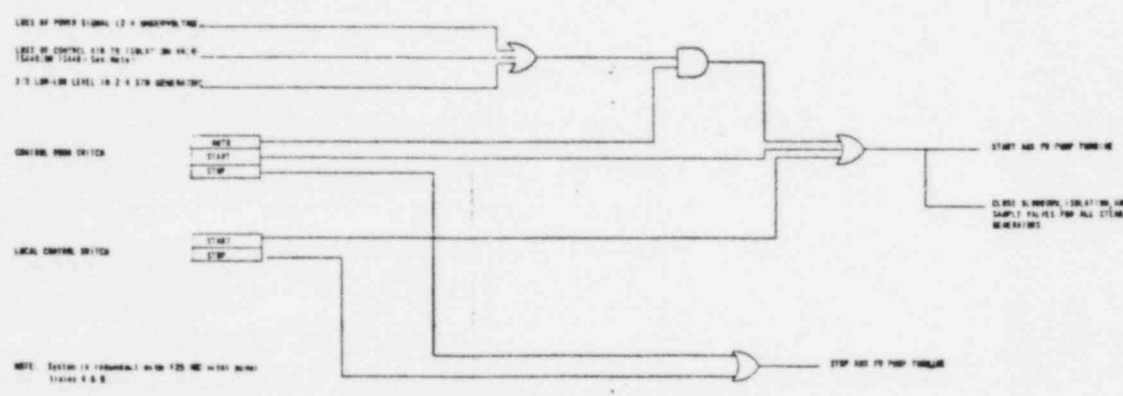
1. For the APPROXIMATE ΔT of the SUPERHEATERS ΔT OF STATE FROM THE TURBINE
 APPROXIMATE ΔT IS 1000. 500 OR 1000 DEPENDS ON THE
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2. CORRECTION TO THE APPROXIMATE ΔT IS AS FOLLOWS:
 LOAD 5 INCREASED 5000
 LOAD 6 INCREASED 5000
 LOAD 7 INCREASED 5000
 LOAD 8 INCREASED 5000

LEVEL FOR TURBINE RUN AUTOMATIC SIGNALS FROM RUN AND STOP





MFW TRIP LOGIC

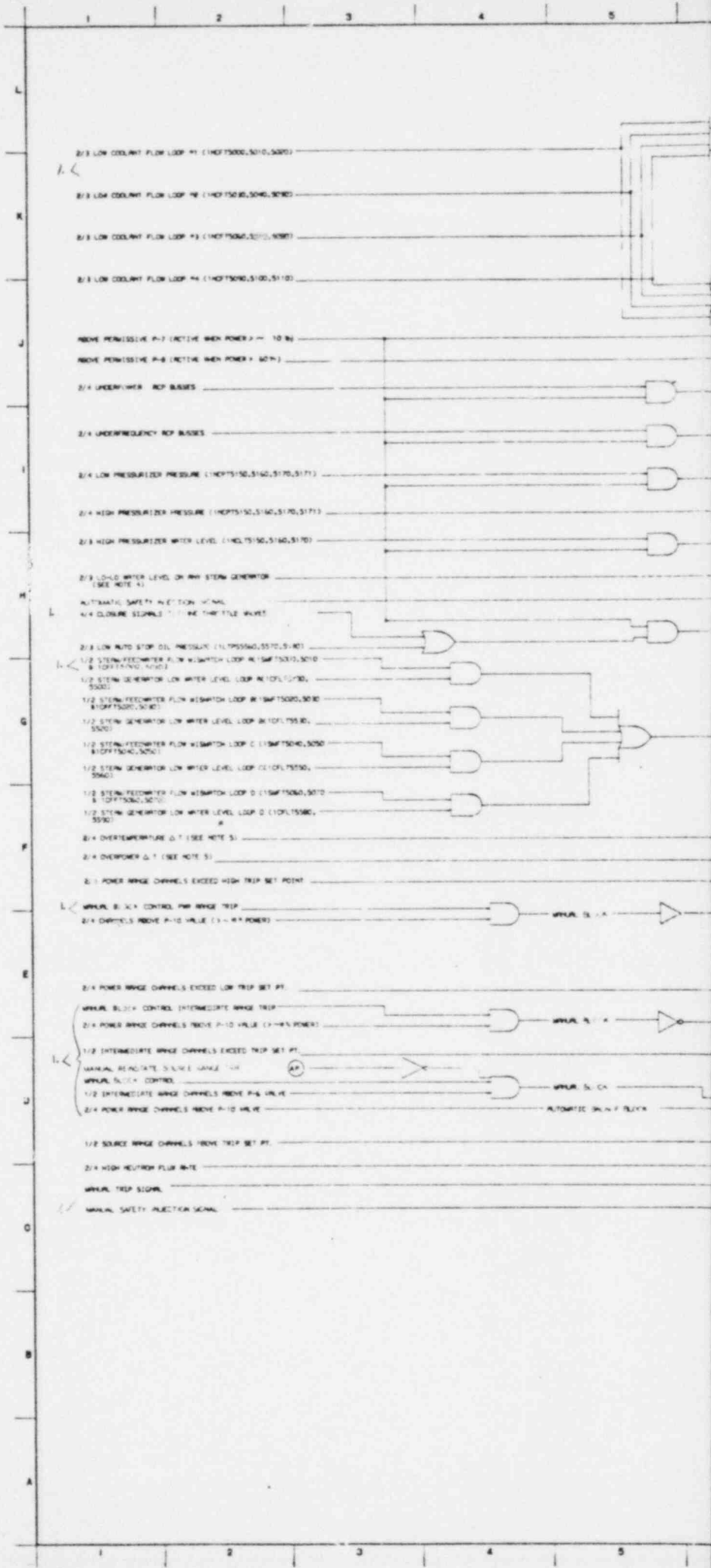


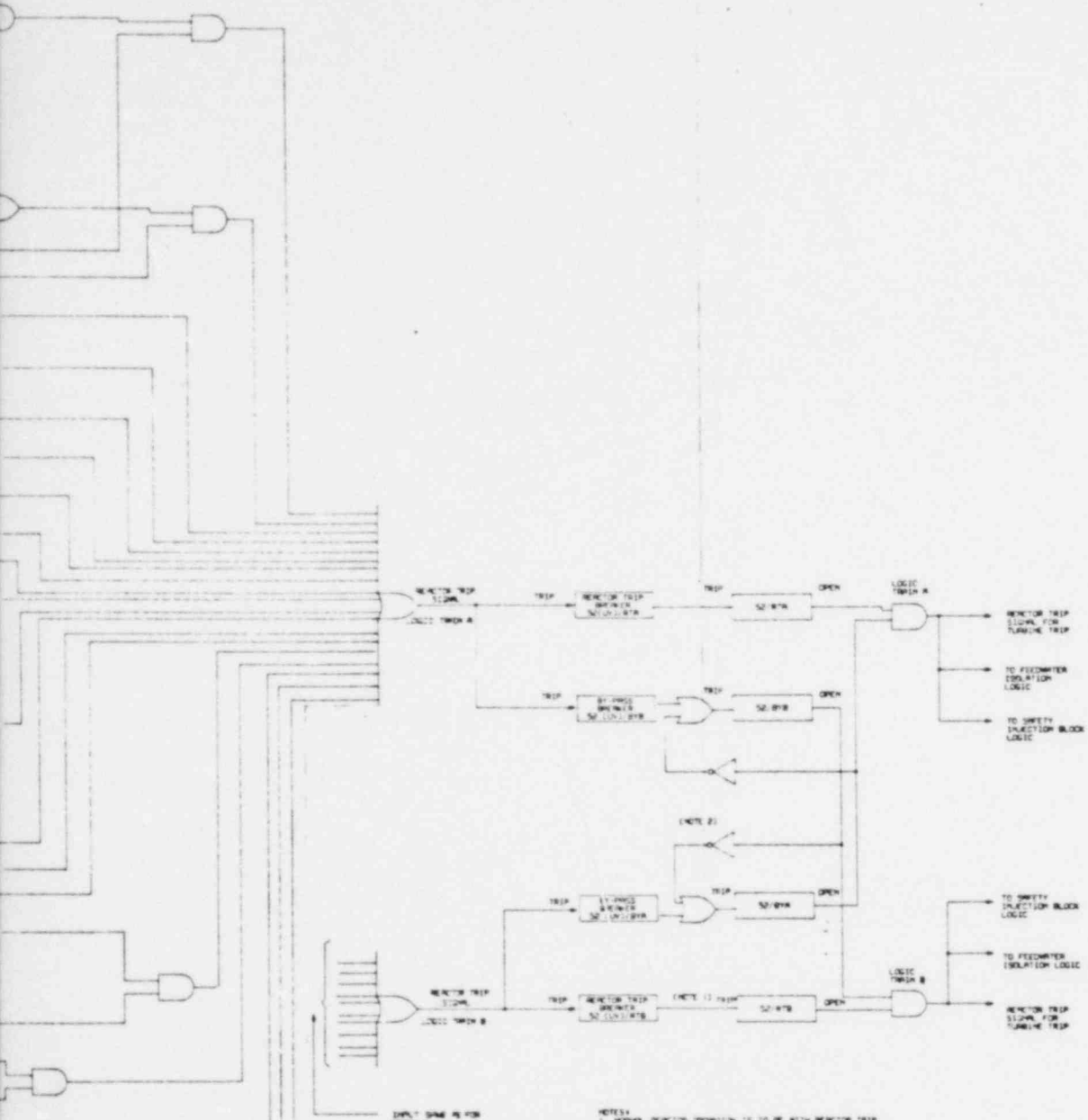
TRIP AND FEED PUMP
 MFW TRIP LOGIC FOR
 START/STOP LOGIC

MCGUIRE NUCLEAR STATION
 START/TRIP LOGIC FOR MAIN AND AUXILIARY
 FEEDWATER PUMPS
 FIGURE 4 - 3

DUKE POWER COMPANY			
MCGUIRE NUCLEAR STATION			
START/STOP LOGIC FOR MAIN AND AUXILIARY FEEDWATER PUMPS			
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- NOTES:
1. NORMAL REACTOR OPERATION IS TO BE WITH REACTOR TRIP BREAKERS IN THE OPEN POSITION AND TRIP SIGNALS IN THE TRIP POSITION. DURING TEST, THE TRIP SIGNALS ARE TO BE WITH THE TRIP SIGNALS IN THE TRIP POSITION AND THE REACTOR TRIP BREAKERS IN THE TRIP POSITION. THE TRIP SIGNALS ARE CONTROLLED FROM THE TRIP TRIP.
 2. THE SAFETY INJECTION BLOCK LOGIC IS OPERATIVE ONLY WHEN BOTH SAFETY INJECTION BLOCKS ARE IN THE OPERATE POSITION.
 3. THE SWITCHING LOGIC FOR THE REACTOR TRIP SIGNALS IS PROVIDED BY THE SAFETY INJECTION BLOCKS. THESE LOGIC ARE TO BE PROVIDED BY THE SAFETY INJECTION BLOCKS.
 4. INSTRUMENTATION FOR THE LOGIC TRIP SIGNALS IS AS FOLLOWS:
 - LOGIC TRIP A: 100-1000-100-100
 - LOGIC TRIP B: 100-1000-100-100
 - LOGIC TRIP C: 100-1000-100-100
 - LOGIC TRIP D: 100-1000-100-100
 5. INSTRUMENTATION FOR THE REACTOR TRIP IS AS FOLLOWS:
 - LOOP A: 100-1000-100-100
 - LOOP B: 100-1000-100-100
 - LOOP C: 100-1000-100-100
 - LOOP D: 100-1000-100-100

MCGUIRE NUCLEAR STATION
 REACTOR TRIP LOGIC
 FIGURE 4 - 4

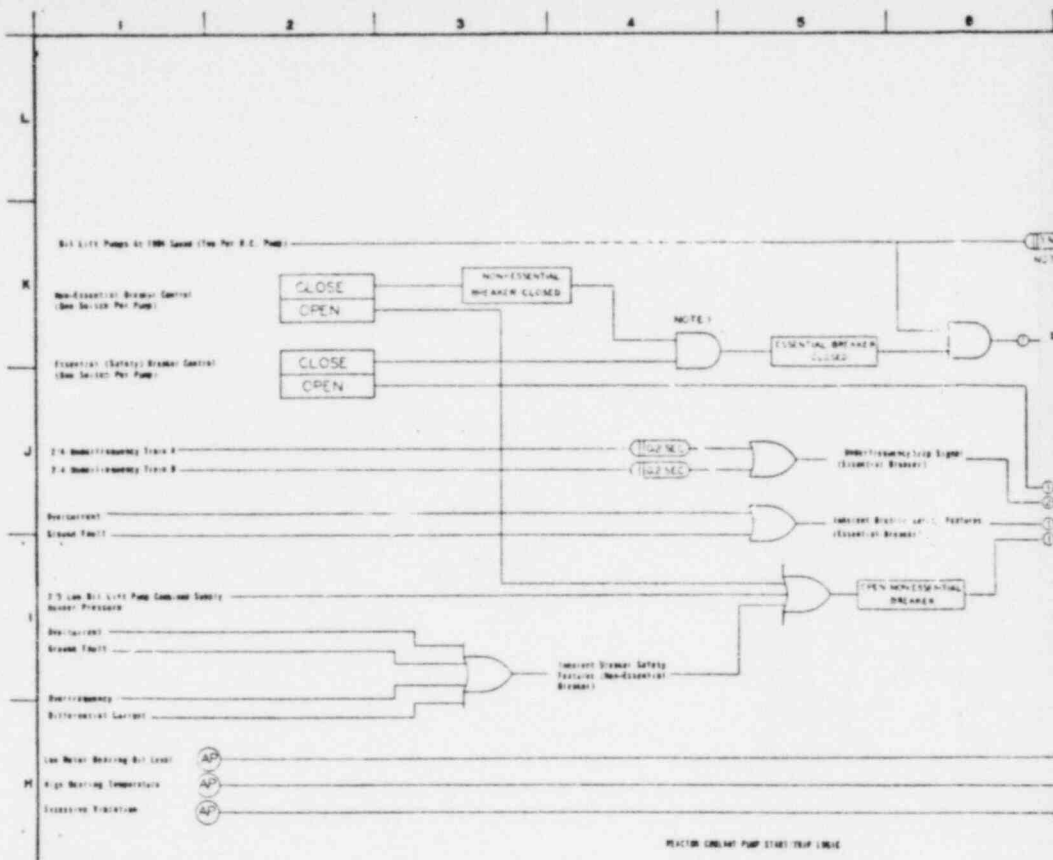
NUCLEAR SAFETY RELATED

DUKE POWER COMPANY
 MCGUIRE NUCLEAR STATION UNIT 1&2

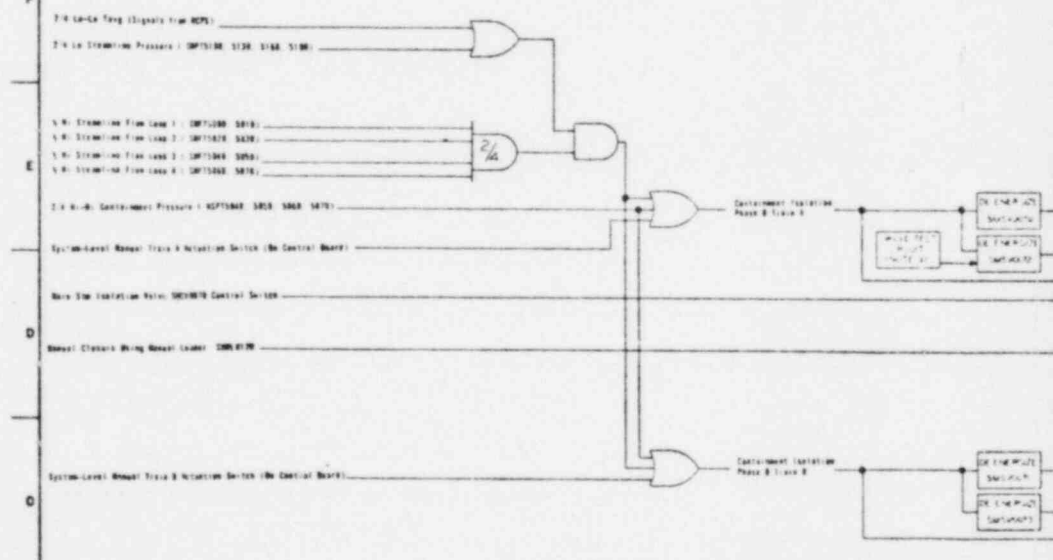
REACTOR TRIP LOGIC

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 CHECKED: *[Signature]*
 DATE: *[Date]*
 DWG. NO. MC-1000-01-5



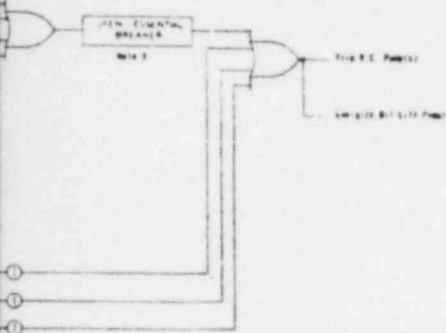
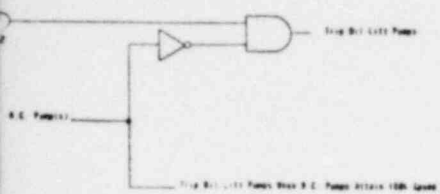
NOTES: 1. THE NON-ESSENTIAL BREAKER MUST BE CLOSED BY REACTOR COOLANT PUMP START LOGIC. THE REACTOR COOLANT PUMP START LOGIC FOR ESSENTIAL BREAKER.
 2. THE 2/4 LOW 2/4 LIFE PUMP CONTROL SAFETY HEADER PRESSURE IS MONITORED BY THE REACTOR COOLANT PUMP START LOGIC.
 3. THE REACTOR COOLANT PUMP START LOGIC IS MONITORED BY THE REACTOR COOLANT PUMP START LOGIC.



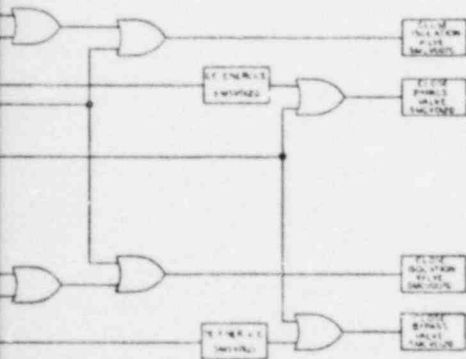
NOTES: 1. THE MAIN STEAM ISOLATION VALVE MUST BE CLOSED BY REACTOR COOLANT PUMP START LOGIC. THE REACTOR COOLANT PUMP START LOGIC FOR ESSENTIAL BREAKER.
 2. THE MAIN STEAM ISOLATION VALVE MUST BE CLOSED BY REACTOR COOLANT PUMP START LOGIC.
 3. THE MAIN STEAM ISOLATION VALVE MUST BE CLOSED BY REACTOR COOLANT PUMP START LOGIC.
 4. THE MAIN STEAM ISOLATION VALVE MUST BE CLOSED BY REACTOR COOLANT PUMP START LOGIC.

MAIN STEAM ISOLATION VALVE AND MAIN STEAM ISOLATION BYPASS VALVE LOGIC (LOOP 1 SUMMARY)

LOOP	ISOLATION VALVE	ACTUATING SIGNALS TO	BYPASS VALVE
1	SMT0100	Train A (SMT0100), Train B (SMT0100), SMT0100	SMT0100
2	SMT0100	Train A (SMT0100), Train B (SMT0100), SMT0100	SMT0100
3	SMT0100	Train A (SMT0100), Train B (SMT0100), SMT0100	SMT0100



THE STOP BUTTON ON THE
 RE-START STATION WILL
 STOP THE PUMP.



REVISIONS

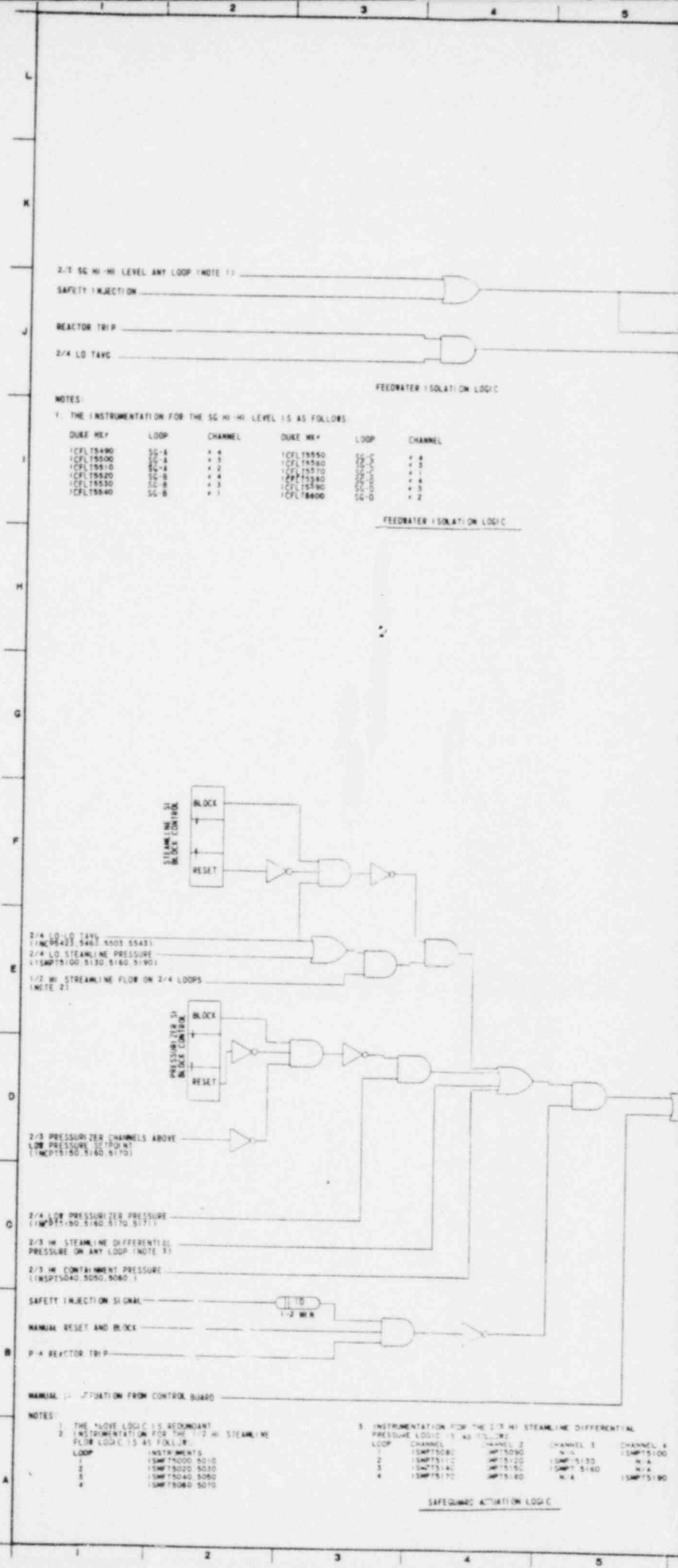
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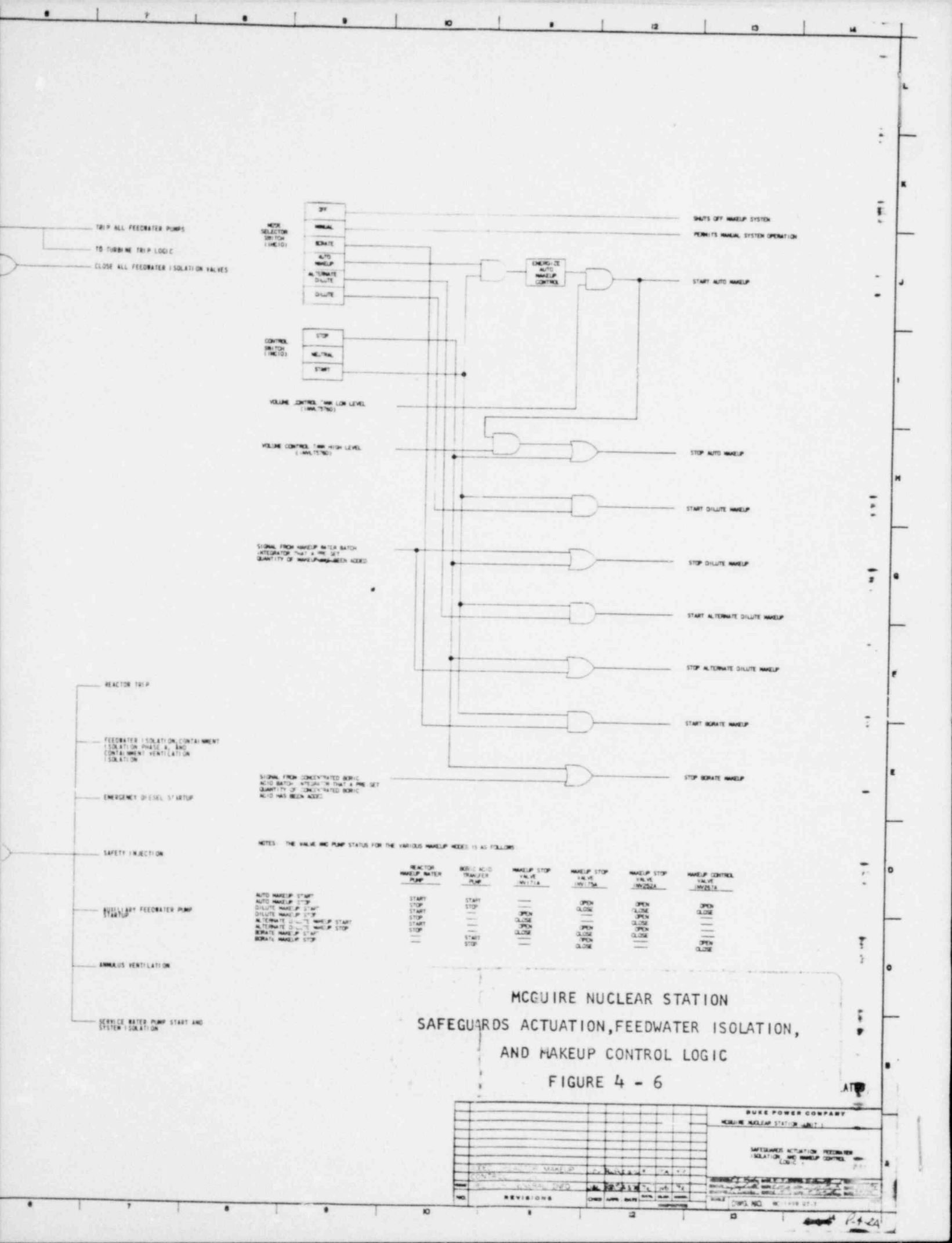
MCGUIRE NUCLEAR STATION
REACTOR COOLANT PUMP START/TRIP AND
MAIN STEAM ISOLATION VALVE LOGIC
FIGURE 4 - 5

NO.	REVISIONS	DATE	BY	APPROVED

NUCLEAR SAFETY RELATED
 DUKE POWER COMPANY
 MCGUIRE NUCLEAR STATION UNIT 1&2
 REACTOR COOLANT PUMP START/TRIP
 AND MAIN STEAM ISOLATION VALVE LOGIC

DESIGNED BY: [Signature]
 DRAWN BY: [Signature]
 CHECKED BY: [Signature]
 SCALE: AS SHOWN
 DWG. NO. MC-1499-DT-B





NOTES: THE VALVE AND PUMP STATUS FOR THE VARIOUS MAKEUP NEEDS IS AS FOLLOWS:

	REACTOR MAKEUP WATER PUMP	BORIC ACID TRANSFER PUMP	MAKEUP STOP VALVE (M/T)A	MAKEUP STOP VALVE (M/T)B	MAKEUP STOP VALVE (M/T)C	MAKEUP CONTROL VALVE (M/T)D
AUTO MAKEUP START	START	START	---	OPEN	OPEN	OPEN
AUTO MAKEUP STOP	STOP	STOP	---	CLOSE	CLOSE	CLOSE
DILUTE MAKEUP START	START	---	OPEN	---	---	---
DILUTE MAKEUP STOP	STOP	---	CLOSE	---	---	---
ALTERNATE DILUTE MAKEUP START	START	---	OPEN	---	---	---
ALTERNATE DILUTE MAKEUP STOP	STOP	---	CLOSE	---	---	---
BORATE MAKEUP START	---	START	---	OPEN	---	---
BORATE MAKEUP STOP	---	STOP	---	CLOSE	---	---

MCGUIRE NUCLEAR STATION
SAFEGUARDS ACTUATION, FEEDWATER ISOLATION,
AND MAKEUP CONTROL LOGIC

FIGURE 4 - 6

BUSS POWER COMPANY
MCGUIRE NUCLEAR STATION UNIT 1

SAFEGUARDS ACTUATION, FEEDWATER ISOLATION, AND MAKEUP CONTROL LOGIC

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5.0 DAMAGE EVALUATION

A damage evaluation has been performed for each of the breaks postulated by application of the criteria in Section 3.0.

5.1 Pipe Whip/Jet Impingement

The following describes the activities performed to determine acceptable and unacceptable damage caused by postulated high energy pipe breaks.

5.1.1 Break Criteria

Breaks are postulated according to the criteria given in Section 3.0.

5.1.2 Establish Plant Arrangement and Layout

The piping systems and other systems and equipment are arranged to provide maximum separation between piping subject to failure and those components requiring protection. Naturally other factors are considered, such as space limitations and fluid flowrate requirements. The combination of all layout considerations results in the final plant arrangement design.

5.1.3 Preliminary Interaction Matrices

Once all targets have been identified, an initial investigation is performed that evaluates the effects of potential interactions between source piping and targets. Due to the size of the plant and the complexity of the piping layouts, the plant is divided, for analysis purposes, into sections. The determinations of the size of each section is based

upon the amount of piping located and physical barriers which provide separation. For example, a section may comprise the area of a mechanical penetration room. A Preliminary Interaction Matrix is developed for each area defined. The matrix documents the results of the effects of pipe breaks within each area. Since this is a preliminary interaction analysis, discrete break locations are not postulated; instead, breaks are assumed to occur throughout an entire piping run. In addition, actual pipe whips and jets are not postulated. Instead, if targets are within close proximity to the broken piping, they are considered to be impacted by the source pipe. These occurrences are called unacceptable interactions.

5.1.4 Final Interaction Analysis

Following plant layout modification resulting from the preliminary interaction analysis, the final interaction analysis is performed. This analysis is based on investigating the actual pipe whip and jet effects due to breaks at discrete locations; i.e., terminal end and intermediate locations. The analysis of the terminal end breaks can be performed immediately following the layout modifications. The analysis of the intermediate breaks cannot be performed until the piping stress analysis is completed.

The results of the final interaction analysis are documented on Pipe Whip Interaction Matrices and Jet Impingement Interaction matrices. These matrices indicate if the targets are

beyond the range of the whipping pipe and jet, or if an unacceptable interaction occurs. This determination is accomplished through the use of the pipe whip and jet models that are developed prior to performing the final interaction analysis. These models are based on the criteria established in FSAR Section 3.6.

The unacceptable interactions must be prevented in order to allow safe plant shutdown. The method of prevention depends upon the cause of interaction. Typically, pipe whip restraints are used to prevent unacceptable pipe whip interactions, while jet deflectors are used to prevent unacceptable jet interactions.

5.1.5 Pipe Whip Model

In order to determine the extent of pipe whip, several factors must be considered. These include location of plastic hinge, direction of pipe whip, and degree of pipe whip. Each item is described in the following paragraphs as they are postulated to exist for circumferential breaks only. Pipe whip is not expected to occur for longitudinal breaks.

A plastic hinge is considered to occur due to the jet thrust force from a circumferential pipe break. Calculations were performed on typical, unsupported piping runs to determine the approximate location of the hinge. These calculations indicate that the hinge locations would occur within a few feet of the break location.

The physical geometry of the piping runs, however, allowed the analysts to determine, based on force vector and moment arms,

the probable locations of the plastic hinge. This typically occurs at the second elbow away from the break. Using this method, the hinge lengths were considerably longer than the calculated lengths. For conservatism, it was decided to use the geometrically determined hinge length because this would result in a larger sweeping of the whipping pipe and, thus, a larger area of potential interactions.

The direction of the pipe whip is also determined based on the geometric configuration of the pipe, and the resultant force vector and moment arm due to the pipe breaks. The pipe is assumed to move in the prescribed direction until it impacts another object (i.e., a target or structural interference) or it impacts the attached portion of its piping run. The pipe is assumed to whip in only one direction; and once it impacts another object or its own piping run, it is assumed to remain in that position.

5.1.6 Jet Model

The FSAR criteria of a 10° jet half angle is the basis for the jet model. This type of jet is postulated to emit from a circumferential rupture throughout the range of pipe whip. It is also postulated to emit from a longitudinal break as the pipe remains in the stationary position.

The jet is assumed to extend until it impacts either a target or structural interference. If the jet is partially obstructed,

it is assumed to extend beyond the interference at the angle at the point of interference.

5.2 Compartment Pressurization

Compartment pressurization analyses were performed for breaks of the high energy lines outside containment containing pressurized gas, steam, or subcooled water that could flash. Turbine Building piping was considered incapable of producing any unacceptable pressurization effects on safety class structures. The following lines and locations were considered:

- 1) Main Steam, Doghouse; 2) Feedwater, Doghouse; 3) Main Steam Supply to Auxiliary Equipment, Doghouse; 4) Blowdown Lines, Auxiliary Building; 5) Letdown Line, Auxiliary Building and; 6) Upper Head Injection, Auxiliary Building.

5.2.1 TMD Code - Short Term Analysis

The TMD Code was used to determine compartment pressures for main steam and feedwater breaks in the Doghouse. Flow rates and energy release rates were calculated using hand calculation methods based on Moody's two phase choked flow model^{1,2}. See Section 5.2.2 for Auxiliary Building breaks.

5.2.1.1 General Description

The mathematical modeling in TMD (Transient Mass Distribution) is similar to that of the SATAN blowdown code in that the analytical solution is

- ¹ F. J. Moody, "Maximum Flow Rate of a Single Component, Two--Phase Mixture," Journal of Heat Transfer, Trans, ASME, Series C, Vol. 86, February, 1965, p.134.
- ² F. J. Moody, "Maximum Two--Phase Vessel Blowdown from Pipes," Journal of Heat Transfer, August, 1966, p. 285.

developed by considering the conservation equations of mass, momentum and energy and the equation of state, together with the control volume technique for simulating spatial variation. The governing equations for TMD are somewhat different in that a two phase (liquid water droplets and steam - air vapor) two component (air - water) system is considered.

The control volume technique is used to spatially represent the structure. The structure is nodalized into many compartments or elements to provide a detailed representation of the structure transient pressures on the internal concrete structures for the design basis pipe break. The structural elements are connected into a flow network with volumes, flow area, flow path length, flow resistance and initial conditions specified as input for all elements and flow paths available.

For control volume analysis, time dependent equations of conservation of mass, conservation of energy and state are evaluated. The boundary conditions for each control volume are provided by the adjacent one. Nothing is lost in the integration process by shrinking the control volumes as small as we please, thus a detailed spatial solution is possible. As a second step in spatial representation, the control volume in which momentum is conserved is displaced from the control volume in which mass and energy is conserved.

The air and steam in each control volume are assumed to be uniformly mixed throughout. Thermal equilibrium is assumed whereby the temperatures for these constituents are equal within each control volume. As a result of uniform mixed assumption, each constituent occupies the entire control volume. The total pressure is the sum of the partial pressures.

5.2.2 RELAP Code

The RELAP Code was used to determine pressurization effects for all Auxiliary Building high energy pipe breaks containing steam or flashing water (see Section 5.2.1 for Doghouse Breaks).

3 | Program RELAP4/MOD5 is a computer program for investigating the transients expected in thermal-hydraulic systems. RELAP4/MOD5 was originally developed by W. H. Rittig, et al, at the Idaho Nuclear Corporation and provides a numerical method for analyzing the transient thermal-hydraulic behavior of pressurized water reactors, boiling water reactors, and experimental water reactor simulators.

3 | RELAP4/MOD5 is a comprehensive program that predicts the interrelated effects of coolant thermal-hydraulics, system heat transfer, core neutronics, and system component interactions. Because the program was

Revision 3

developed to solve a large variety of problems, the user must specify the program and system to be analyzed.

3 | The fluid dynamics portion of the RELAP4/MOD5 program solves |
the fluid mass, energy and flow equations for the system being modeled. In order to provide a reasonable degree of versatility
3 | in RELAP4/MOD5, a choice of the following basic forms of the flow |
equation is provided:

- a) Compressible single-stream flow with momentum flux.
- b) Compressible two-stream flow with one-dimensional momentum mixing.
- c) Incompressible single-stream flow without momentum flux.

The compressible two-stream flow equation has four forms to represent different flow patterns of the streams. The fluid
3 | system to be analyzed by RELAP4/MOD5 must be specified by the |
user and is modeled by fluid volumes and junctions (flow paths) between volumes. Fluid volumes (control volumes) are used to represent the fluid in the system piping, plenums, reactor core, pressurizer and heat exchangers. Any fluid volume may be chosen independently to represent a region of the system associated with a heat sink or source, such as fuel rods or a heat exchanger. The fluid volumes are connected by junctions which are used to transfer fluid into and out of fluid volumes. Options are available for selecting pump, valve and bubble rise models.

A heat conductor model is used to transfer heat to or from the fluid in a fluid volume. The geometry and conditions of the heat conductor are specified by the user.

3

Several options

are also available for describing heat exchangers.

3

RELAP4/MOD5 is the most recent of a series of computer programs developed at the Idaho National Engineering Laboratory to describe the thermal-hydraulic conditions attendant to various

3

postulated transients in light water reactor systems. RELAP4/MOD5 incorporates various improvements and modifications to previous versions which include:

3

1. A metal-heat conduction solution is available and the capability of describing both the primary and secondary flow systems with a "heat exchanger" heat conduction path connecting the two systems.
2. The RELAP4/MOD5 fluid conservation equations include approximations of all terms contained in the theoretical derivations for one-dimensional, homogeneous, thermal equilibrium flow. The solution to the conservation equations describing the flow of energy and momentum of the coolant has been extended to include the kinetic and potential energy terms and an optional one-dimensional momentum flux description of area and density changes. This modification to the solution of the conservation equations fundamentally allows description

Revision 3

3 |

- of the incompressible Bernoulli effect as well as the compressible fluid flow phenomena. Additionally, flow choking can be calculated from these modified equations. The inclusion of the momentum flux and kinetic energy terms makes the RELAP4/MOD5 conservation equations more general than the equations contained in previous versions and provides the program with the capability of predicting the hydrodynamic behavior of jet pumps.
3. An improved pump model is available which accounts for inertial and friction effects and allows the simulation according to homologous characteristics.
 4. Steam tables that contain the derivatives of fluid properties.
 5. An implicit integration technique is included. The implicit weighting factor can be modified to obtain a fully explicit calculation as well as the normal fully implicit calculation for the flow solution.
 6. Flow models have been incorporated in RELAP4/MOD5 which allow the flow of any combination of air, steam and water. This homogeneous equilibrium (HEM) flow model uses a perfect gas assumption for air.

3 |

5.2.3 Hand Calculations

Conservative hand calculations were performed for the Upper Head Injection System in the Auxiliary Building. This method assumes a constant isentropic choked flow into the compartment from the nitrogen source. After entering the compartment, the nitrogen flow is assumed to decelerate to stagnation conditions and mix with the air in the compartment. The mixing is assumed to occur at constant temperatures. The air-nitrogen mixture leaves the compartment through vent openings at subsonic velocities.

As long as the flow rate into the compartment exceeds the flow rate out of the compartment, the pressure will rise. When the flow into and out of the compartment is equal, the peak compartment pressure can be obtained.

5.3 Flooding Effects

All high energy and moderate energy systems have been reviewed to determine conservative maximum flooding levels in the Auxiliary Building. The Auxiliary Building doorways, stairwells, and floor drains are such that discharged fluids will drain into either the Auxiliary Feedwater Pump Room at Elevation 716 (see Figure 2-3, Rooms 10 and 53) or the Containment Spray and Residual Heat Removal Pump Rooms at Elevation 695 (see Figure 2-2). Pipe breaks in the Auxiliary Feedpump Room would be contained in the room. Breaks in all other areas containing fluid piping would eventually result in flow to the pump rooms at Elevation 695.

5.3.1 Discharge Rates

Discharge rates are based on Bernoulli's equation for non-flashing water and either Moody's two-phase choked flow model or Duke program SONVEL (reference FSAR Section 3.6.4.2) for steam and flashing water. All steam and steam/water mixtures were assumed to condense for the flooding analysis.

5.3.2 Moderate Energy Piping

All moderate energy systems are reviewed by system to determine maximum discharge rates from cracks. System fluid capacity is considered in determining the severity of each system's flooding potential.

5.3.3 High Energy Piping

Each high energy pipe break is reviewed to determine maximum discharge rates from the break. In addition, all piping or equipment (non-essential) that could be damaged by an unrestrained whipping pipe is considered failed and the resulting discharge of fluid considered.

5.4 Environmental Consequences

Environmental consequences as discussed in this report include the effects of water spray from all moderate energy piping and the temperature effects from all high and moderate energy piping containing steam or subcooled water that could flash to steam.

5.4.1 Water Spray

All moderate energy piping has been reviewed to determine if critical cracks could cause unacceptable damage to essential components. Several methods were used to eliminate unacceptable consequences: (1) relocate equipment, (2) reroute piping, (3) eliminate cracks in Class B, C or F piping by showing stresses are less than $0.4 (1.2S_h + S_A)$, (4) upgrade equipment, (5) add spray shields.

5.4.2 Temperature Effects

All Auxiliary Building piping containing steam or subcooled water capable of flashing has been reviewed to determine peak temperatures due to postulated failures. The RELAP Code was used to calculate compartment temperatures in the Auxiliary

Building. This code is discussed in Section 5.2.2.

The following lines were considered: (1) Steam Generator Blowdown, (2) Chemical and Volume Control,

(3) Residual Heat Removal System, (4) Auxiliary Steam, (5) Boron Recycle System and (6) Liquid Waste Recycle System.

Temperature effects in the Doghouses were evaluated using the RELAP4/MOD5 code and considering a circumferential break within the guard pipe of the Main Steam (SM) system (See Figure 3-16). The SM break enveloped all other piping breaks in the Doghouse.

Revision 3
Entire Page Revised

5.0 SUMMARY AND CONCLUSIONS

In the following sections, the analysis of postulated high energy and moderate energy piping failures is summarized. These sections include the effects of the breaks and the plant modifications required to mitigate the effects.

6.1 Pipe Whip and Jet Impingement Effects

In accordance with the criteria and methods in Sections 4.0 and 5.0, Table 6-1 and Table 6-2 have been prepared to identify each high energy break considered in this analysis and to identify those breaks that result in unacceptable consequences. For those cases where unacceptable consequences occur, pipe whip restraints, jet impingement barriers, guard pipes or other mitigation devices are provided. The criteria used for design of these devices is presented in Appendix A.

6.1.1 Main Steam and Feedwater Systems

The Main Steam and Feedwater Systems have the greatest potential for damage from postulated pipe breaks. The pipe sizes, operating temperatures and pressures, and the large amount of stored energy require that these systems receive the most thorough review with respect to pipe rupture effects. Portions of the main steam and feedwater piping have been provided with guard pipe to limit pipe movement and restrict blowdown flow rates in the event of a process pipe failure inside the guard pipe. Main steam and feedwater piping drawings are shown in Figures 3.16 and 3.12 respectively.

Main steam breaks are chosen based on consequence as described in FSAR Table 3.6.5-1. Pipe whip restraints are provided for

each postulated break to assure that essential structures and components are not damaged. Only circumferential breaks are postulated since the stress at all locations is less than $0.8 (1.2S_h + S_A)$. It should be noted that the main steam and feedwater penetrations are free floating and do not constitute an anchor. Figure 6-1 shows the main steam system isometric drawings. Typical guard pipe to process pipe internal restraints are shown in Figure 6-2. Energy absorber and rigid pipe rupture restraints are shown in Figures 6-3 and Figure 6-4. Feedwater break locations are chosen at those locations shown on Figure 3.12 based on high stress points; only circumferential breaks are postulated since the stresses are less than $0.8 (1.2S_h + S_A)$ at all locations. Pipe rupture restraints are provided for all postulated feedwater breaks to assure that essential components and structures are not damaged.

6.1.2 Steam Generator Blowdown System

The high energy portion of the blowdown system consists of one blowdown line per steam generator entering the Auxiliary Building in the Elevation 716/733 penetration room and exiting to the Turbine Building (see figures 3.1 and 3.10). Only circumferential breaks are postulated since the piping is of seamless construction and the stresses are less than $0.8 (1.2S_h + S_A)$ at all locations*.

The pipe whip and jet impingement evaluation indicates that damage to several essential valves is possible for breaks

*In Section AB 716-07 only one intermediate break location is shown for 2 of the four lines because of the low stress levels.

3 | BB1-01, -03. Pipe whip restraints will be provided to protect these valves. |

6.1.3 Auxiliary Feedwater System

3 | The high energy portion of this system consists of the Motor Driven Auxiliary Feedwater Pump 1A and 1B discharges from the pumps at Elevation 716 (room 716-10 on Figure 203) to the main feedwater lines in the doghouse (rooms 750-23 and -26, Figure 205). The flow diagrammatic is shown in Figure 3.2 and the piping layout is shown on Figure 3.11 and 3.12. The Turbine Driven Pump is considered moderate energy since it will only be used for emergency or testing conditions. The stress levels in the system are less than 0.8. $(1.2S_h + S_A)$ and seamless piping is used. As a result, only circumferential breaks are postulated.

3 | A review of the potential consequences resulting from pipe whip and jet impingement indicates pipe whip restraint and jet impingement barriers are required for postulated breaks CA1-51 and CA1-70A to protect essential electric equipment. |

6.1.4 Safety Injection System (Upper Head)

3 | The high energy portion of the Safety Injection System consists of the Upper Head Injection Accumulator Tanks and connecting piping. The flow diagrammatic is given in Figure 3.4, sheet 2 and the piping layout is shown on Figure 3.13. The stresses at all locations are less than 0.4 $(1.2S_h + S_A)$ and seamless piping is used; therefore, only circumferential terminal end breaks are postulated. |

3 The pipe whip and jet impingement evaluation of the UHI postulated pipe breaks shows that no unacceptable damage to essential components results for any of the postulated breaks. A rupture restraint is provided for the 12" diameter line break at N11-02 to prevent unacceptable interaction with the building column at HH and 52.

6.1.5 Boron Thermal Regeneration System

3 The high energy portion of this system consist of the piping from the letdown line to the Letdown Reheat Heat Exchanger and back to the letdown line. This piping can be seen on the flow diagram on Figure 3.5 and on the piping drawing shown on Figure 3.14. Only circumferential terminal end breaks were postulated (at inlet and outlet of Letdown Reheat Heat Exchanger) since terminal ends are flanged; these would cause no unacceptable damage to essential components. No intermediate breaks were postulated since the stress levels are less than 0.4 ($1.2 S_h + S_A$).

6.1.6 Chemical and Volume Control System (CVCS)

The high energy portion of the CVCS System consists of the normal charging and sealwater injection lines and the letdown line. The flow diagrammatics for this system are shown in Figure 3.6 while the pipe routings are shown in Figure 3.15. Circumferential breaks are postulated in the normal charging and sealwater lines only at terminal ends since the stresses are less than 0.4 ($1.2 S_h + S_A$) at all locations. The pipe whip and jet impingement evaluation indicates that no damage to essential components would occur for these breaks.

Revision 3

3 | In the letdown line, breaks are postulated to occur at the terminal ends and at the only two locations where stress levels in excess of $0.4 (1.2S_h + S_A)$ occur.

Only circumferential breaks are postulated since the stress levels are less than $0.8 (1.2S_h + S_A)$ at all locations. The pipe whip and jet impingement evaluation shows that damage to electrical cables, containment spray piping and/or nuclear service water piping is possible due to breaks NV1-07 and NV1-13. Pipe whip restraints will be provided to preclude this damage.

6.1.7 Main Steam Supply to Auxiliary Equipment - SA

The high energy portion of the SA System consists of the piping from main steam headers B and C to the normally closed piston operated valves. Figure 3.7 shows the flow diagrams for the system while Figure 3.16 shows the piping configuration.

3 | Circumferential pipe breaks are postulated at the terminal ends and at the highest stressed intermediate locations in the seamless piping. All other stresses in the high energy portion are less than $0.4 (1.2S_h + S_A)$. The pipe whip and jet evaluation indicate that no unacceptable damage would occur for these breaks.

6.2 Compartment Pressurization Effects

The only breaks outside containment that result in significant differential pressures are the main steam and feedwater systems which are located in the doghouse areas. Inside the Auxiliary Building, the venting through HVAC ducts, hallways and doors prevent excessive pressure differentials from occurring.

6.2.1 Main Steam and Feedwater

Main steam and feedwater breaks in the doghouse were reviewed for pressurization effects using the TMD Code (see Section

5.2.1 for description). Pressure time histories were calculated for Main Steam and Feedwater breaks. Figure 6-5 shows the doghouse model (see areas 23, 24, 25, and 26 on Figure 2-5 for location) used for all calculations. The doghouses vent directly to atmosphere. Table 6-4 provides the compartment volumes, flow paths, areas, loss coefficients, and blowdown data used in the analysis. The most severe peak pressure calculated was 5.83 psig for main steam breaks and 8.85 psig for feedwater breaks.

6.2.2 Upper Head Injection System

The Upper Head Injection Accumulator and piping are located in Rooms 750-01, -04, and -05 (see Figure 2-5 and Figure 3.13). Ruptures in the twelve inch piping would not cause unacceptable consequences.

3

Breaks N11-07 through N11-18 were reviewed using the methods described below. The most severe differential pressure was found to be 0.03 psig. This would not cause unacceptable consequences.

The differential pressures for Upper Head Injection System breaks were determined as follows.

The nitrogen in the accumulator and the air in the compartment are considered to be in thermal equilibrium initially, and the discharge process is conservatively assumed to produce an isothermal process in the nitrogen remaining in the tank.

The nitrogen flow from the accumulator into the compartment is

assumed to be a constant isentropic choked flow:

$$W_1 = \frac{AP_0}{\sqrt{T_0}} \sqrt{\frac{kg}{R} \left(\frac{2}{k+1} \right)^{\frac{k+1}{k-1}}}$$

where W_1 = Flow rate, lbm/sec.

A = Pipe break flow area, in².

P_0 = Reservoir pressure, psia.

T_0 = Reservoir temperature, °R

k = Ratio of specific heats

R = Gas constant, ftlb/lbm °R.

g = Acceleration of gravity, ft/sec².

After entering the compartment, the nitrogen flow is assumed to decelerate to stagnation conditions and mix with the air in the compartment. The mixing is assumed to occur at constant temperature. The air-nitrogen mixture leaves the compartment through the vent openings normally at subsonic velocities and enters the atmosphere. The equation of flow thus is:

$$W_2 = C P_c A_c \sqrt{\left(\frac{2g}{RT_c} \right) \left(\frac{k}{k-1} \right) \left[\left(\frac{P_A}{P_c} \right)^{2/k} - \left(\frac{P_A}{P_c} \right)^{\frac{k+1}{k}} \right]}$$

where W_2 = Flow rate, lbm/sec.

P_c = Pressure in the compartment, psia.

A_c = Vent area of the compartment, in².

P_A = Atmospheric pressure, psia.

T_c = Temperature in the compartment, °R.

k, R, g = Same as previously defined.

C = Discharge coefficient (0.6).

As long as the flow rate into the compartment exceeds the flow

rate out of the compartment, the pressure will rise. When the flow rate into and out of the compartment are equal, the peak compartment pressure will be attained. Therefore, setting $w_1 = w_2$:

$$\frac{AP_o}{T_o} \sqrt{\frac{kg}{R} \left(\frac{2}{k+1}\right)^{\frac{k+1}{k-1}}} = CP_c A_c \sqrt{\left(\frac{2q}{RT_c}\right) \left(\frac{k}{k+1}\right) \left[\left(\frac{P_A}{P_c}\right)^{2/k} \left(\frac{P_A}{P_c}\right)^{\frac{k+1}{k}}\right]}$$

from which P_c can be obtained by iteration.

6.2.3 Blowdown System

The Blowdown System piping is shown in Figure 3.10. All postulated break locations are located in Volume 716-07 (see Figure 2-3). The differential pressures were calculated using the RELAP Code as described in Section 5.2.2. The most severe differential pressure was found to be 0.27 psig, at a time of 2.2 sec. This condition would not result in unacceptable consequences.

6.2.4 Letdown Line, CVCS System

The letdown line is shown on Figure 3.15, sheets 3, 5, and 6. The letdown flow is restricted by a 0.272" diameter flow orifice inside containment (see Figure 3.6, sheet 3). Differential pressures calculated from the RELAP Code are negligible.

6.3 Flooding Effects

The review of flooding effects from postulated piping failures includes moderate energy and high energy systems. As discussed in Section 5.3, fluid discharged in the Auxiliary Building would eventually drain to either elevation 695 or to the Auxiliary Feedwater Pump rooms, depending on the location of the break. The doghouse has been sealed to prevent fluid from a main steam or feedwater break from draining to the Auxiliary Feedwater Pump rooms. Curbs are provided around passageways to the Auxiliary.

Building from the Turbine Building. These curbs are of adequate height to contain flood water caused by the break of the main condenser circulating water expansion joint, or the most severe Condensate System failure for a minimum of fifteen minutes. There are no pipe or cable chase entrances below the elevation of the top of the curbs. This flooding condition does not render any essential system or component inoperable.

6.3.1 Auxiliary Feedwater Pump Room

The most severe pipe rupture discharging into the Auxiliary Feedwater Pump Room (Room 716-10, Figure 2-3) is a break at the discharge nozzle of one of the Auxiliary Feedwater Pumps. This flow would be 700 gpm with a flooding capacity of 30,000 gal. in the room. Detection of such a break or similar break would be from the redundant, high sump level alarms in the control room. No damage to essential components would occur, assuming operator action after thirty minutes.

6.3.2 Containment Spray/Residual Heat Removal Pump Rooms

A review of all postulated failures in the remainder of the Auxiliary Building indicates that the most severe discharge rate is due to a critical crack in the 36" nuclear service water piping. This flow of 1335. gpm would accumulate in the pump rooms and hall at Elevation 695 (See figure 2-2, rooms 1, 2, 3, 4, 8, 9, 10, 11, 12, 13, and 17). A capacity of 80,000 gallons in these rooms is adequate assuming operator action after 30 minutes. Detection of such a break or similar break would be

from the redundant high sump level alarms in the control room.

6.4 Environmental Consequences

6.4.1 Water Spray

Most essential components are sealed against the adverse effects of water spray. Those components which cannot be sealed were identified and all high and moderate energy systems in the Auxiliary Building were reviewed to determine water spray effects. Those essential components that could be sprayed and are not sealed are identified in Table 6-3.

6.4.2 Temperature Effects

High energy lines identified in Table 6-5 are evaluated for direct results of jet impingement and for all adverse environmental effects (pressure and temperature). Moderate energy lines identified in Table 6-5 are evaluated for water spray and temperature effects. A relative humidity of 100% is assumed to result from any high or flashing moderate energy pipe break.

A matrix of lines, line breaks and safety-related equipment within Auxiliary Building compartments was developed and the pipe breaks determined in accordance with Section 3.0. Flow rates and enthalpy release rates for each break were calculated using either the RELAP4/MOD5 code or by hand calculation using the Moody two-phase choked flow model. Non-restricted, frictionless blowdown was assumed in all break cases except for the BB System breaks where

two-phase choked flow model. Non-restricted, frictionless
blowdown was assumed in all break cases except for the BB
system breaks where

3 | restrictions for friction and equipment were accounted for. Table
6-5 provides a list of the system line breaks, break types,
initial line conditions and mass and energy release rates of
the breaks under consideration.

1 | A preliminary enveloping was performed for all break cases
3 | identified in Table 6-5 by estimating the maximum temperatures
1 | expected, either local to the rupture or throughout the build-
3 | ing. Detailed RELAP4/MOD5 computer models were built for
1 | each floor of the Auxiliary Building using as many as 50 volumes
3 | and 100 connecting flow paths. The regions in the vicinity where
1 | the pipe break was postulated were extensively nodalized while the
3 | more distant regions, including the upper and lower floors, were
1 | modeled with less detail. All portals, stairways, pipe chases
and HVAC equipment were modeled as flow paths in addition to
doorways and corridors.

3 | The break flow was introduced as a fill into the appropriate
1 | volume housing the broken piping, and the temperature of each
3 | volume in the model, as calculated by the RELAP4/MOD5 computer
code, was analyzed for environmental consequences. Through the
use of enveloping runs, the entire series of over 55 individual
break cases was reduced to a set of breaks causing the worst
case environmental responses. The peak node temperature which
occurs anywhere within the first 30 minutes following pipe rupture
is considered as the criteria for defining the worst case or
enveloping environmental temperature. The equipment design tem-
1 | perature was then compared to the worst environmental temperature

1 due to the break cases analyzed. Enveloping was performed
for the environmental response only and all individual break
cases were considered for jet impingement or water spray effects.

6.4.2.1 Auxiliary Building Environmental Analysis Assumptions

The following assumptions were made for all analyses:

1. Initial Auxiliary Building compartment pressures and temperatures were assumed to be 14.7 psia and 70°F.
2. All lines were assumed to be at their maximum normal operating pressures and temperatures.
3. All blowdown mass and energy release rates were based on unrestricted and frictionless blowdown except for the Steam Generator Blowdown System (BB). The mass and energy release rates for the BB System include restrictions for friction and mechanical equipment in the piping system.
4. All doors were assumed to be closed. Communications between floors occurs only via open HVAC ducts, spiral stairways, and through gratings along the containment penetration areas.
5. Through the use of temperature activated HVAC dampers and sealed portals, all connections between the regions where pipe ruptures occur and the Battery Rooms (areas along columns AA to EE in E1. 733'), the Control Room (areas between columns AA to EE in E1. 767'), the HVAC equipment room (areas between columns DD to GG in E1. 767') and also, areas along columns AA to EE in E1. 75C and E1. 716 are isolated.

from pipe rupture effects.

6. The YH line has a fluid boiling point temperature of 240°F and the fluid will not flash to steam in the case of a pipe rupture. This line is, therefore, reviewed only for water spray effects.

7. Initial doghouse pressure and temperature were assumed to be 14.7 psia and 135°F.

6.4.2.2 Auxiliary Building Environmental Analysis Results

6.4.2.2.1 Auxiliary Steam Line

A crack was postulated in the 2" Auxiliary Steam line in the southwest corner (region along columns 51-52, PP-NN on E1. 750', Figure 6-6) of the Auxiliary Building. Line fluid conditions of 65 psia and 298°F were used for this analysis.

No significant environmental effects will occur due to the low mass and energy release rates (90.6 BTU/sec. and 0.077 lbm/sec.) calculated for this break.

Because the crack was postulated in an open area, the temperature rise will be so slow that the environmental effects produced by this crack are enveloped by the NV and ND line breaks on this floor. Thus, a detailed analysis was not performed on this break.

6.4.2.2.2 Steam Generator Blowdown (BB)

Design basis circumferential breaks were postulated in the 2" and 4" BB

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lines along the Unit 1 and 2 penetrations between El. 716' and El. 733' (see Figure 3.10). The 4" line was analyzed due to the fact that its higher mass and energy release rates envelop the 2" lines. This line has an initial temperature and pressure of 557°F and 1107 psia. Due to the friction and high upstream restrictions, the blowdown flow rate was calculated to be 287 lbm/sec during the first 1.95 sec. of the break transient. After 1.95 seconds, the blowdown would stay at a constant rate of 75 lbm/sec. The energy release followed the same pattern with a value of 5.9×10^5 BTU/sec for the first 1.95 seconds and 42.3×10^3 BTU/sec thereafter. Typical environmental temperature response curves for this break are provided in Figure 6-7 which illustrates the environmental response to the changing blowdown energy release rates. Following a postulated design basis break in the penetration area, the whole penetration area and the pipe shafts heat up rapidly, reaching a peak temperature of 212°F. The steam soon propagates to the lower level (El. 716') through the gratings

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around the penetration area, causing regions on that floor to heat up to a peak of 212°F in 30 minutes.

6.4.2.2.3 Boron Recycle System (NB)

A crack was postulated in the 6" line (Auxiliary Steam Supply) in the boron recycle unit room (between columns 56-57, NN-PP) at El. 710'. This line has an initial operating temperature and pressure of 298°F and 65 psia, and the blowdown rate was calculated to be 0.75 lbm/sec.

No significant environmental effects will occur. The temperature of the room in which the crack is postulated rises very slowly (7° in 2 minutes) due to the small blowdown rate. The hot mixture vents through a small opening to the rest of the building' thus, no rise in temperature will be observed in other regions. The room contains no essential equipment. It is concluded that this crack is enveloped by other breaks on this floor and no adverse results occur.

6.4.2.2.4 Residual Heat Removal (ND)

Cracks were postulated to occur in the RHR lines at their maximum normal operating

condition of 350°F and 365 psia. The cracks were assumed to occur in various locations in the 3", 8", 14" and the 18" piping. A total of 16 cracks were considered. Successive enveloping of the cracks reduced them to three cases with severe environmental effects capability.

6.4.2.2.4a Residual Heat Removal Pump Room

Cracks were postulated to occur in the 8" RHR lines in the residual heat removal pump room at El. 695 (see Fig. 2-2-areas 1,2, 10,11). The floor reaches a maximum temperature of 212° in 30 minutes. The heated air/steam mixture soon fills the floor and propagates to higher levels through the spiral staircase.

6.4.2.2.4b Pipe Chase at El. 695'

A crack postulated in the 18" ND line in the pipe chase at El. 695' (see Fig. 2-2-areas 5,6, 14,15) causes the break node to heat up to 212°F.

The
air/steam mixture propagates into the other

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3
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3
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regions in El. 695', mainly through the HVAC ducts. The mixture also vents to El. 716 through pipe shafts

The heated air/steam mixture also vents to the central regions of El. 716' through HVAC ducts and El. 733' through the gratings around the penetration area,

6.4.2.2.4c Residual Heat Removal Heat Exchanger Room

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3
1

A postulated crack in the 8" RHR pipe in the heat exchanger room (see Fig. 2-5-areas 2,3,28, 29) causes the room temperature to rise to 212°F in 30 minutes. The only communication between this room and the rest of the building is through a wire gate at El. 750'. Due to the fact that regions on this floor are quite open and have comparatively few restrictions, the air/steam mixture soon propagates freely throughout the floor and establishes regional temperatures around the vicinity of the heat

exchanger room between 132°F and 212°F in 30 minutes. Temperatures on Elevation 767 reach 130°F within 30 minutes.

6.4.2.2.5 Chemical and Volume Control (NV)

Based on break enveloping, two 3" NV line break cases in the Letdown Line were analyzed to determine environmental effects. The NV line breaks were postulated to occur at their maximum normal operating condition of 350°F and 365 psia. The postulated breaks considered are located in the Mechanical penetration areas at El. 750' (see fig. 2-5-areas 1 and 2/).

These breaks produced localized temperature effects due to upstream restrictions. The temperatures in the areas reached 132°F within 30 minutes with the rest of El. 750' being enveloped by ND breaks in the RHR Heat Exchanger Rooms.

6.4.2.2.6 Liquid Waste Recycle (WL)

Two cracks were postulated to occur in the 4" WL line (Auxiliary Steam supply) at El. 710' (room between columns LL-KK, 60-61,

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Figure 6-6). This line has a blowdown condition of 298°F and 65 psia and is enveloped by the ND break for environmental effects. No computer simulation was performed for this line.

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6.4.2.3. Doghouse Environmental Analysis Results

A circumferential break was postulated within the guard pipe of the SM line inside the Doghouse. Saturated steam conditions at 1107 psia were used for the analysis. A peak temperature of 330°F was obtained seconds after the break initiation due to the large mass and energy release (4655 lb/sec and 5.5×10^6 BTU/sec, respectively) and relatively small Doghouse volume. The steam environment propagates directly to atmosphere through large vents located in the upper portion of the structure.

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6.4.2.4 Temperature Effects Due to Jet Impingement & Water Spray

Jet impingement and/or water spray effects were evaluated for each of over 55 individual pipe rupture locations identified in Table 6-5. Break case enveloping, which was performed for the Auxiliary Building Environmental Analysis, was not permitted for the jet impingement/water spray analysis.

Targets were chosen by the construction of a jet cone

with a 10° half angle from each circumferential or crack break. Minor jet obstructions such as HVAC ducting or mechanical equipment were not considered to be barriers in the target selection and only permanent partitions and structures were taken to be jet barriers. In some regions (notably, elevation 695') where the compartments are very small, the structural barriers themselves were assumed to act as jet deflectors, directing the water spray into the target, thus creating additional targets. Target impingement temperatures were conservatively set at the stagnation temperature of the source piping.

6.4.2.5 Evaluation of Consequences

All essential equipment affected by these temperature effects (both environmental and water spray) are reviewed to assure that equipment qualification temperatures are not exceeded. The worst case environmental temperatures within the first 30 minutes for the Auxiliary Building compartment due to breaks on both the Unit 1 and Unit 2 sides are summarized in Figure 6-6.

TABLE 6-1

HIGH ENERGY PIPE BREAK LOCATION

Sheet 1 of 15

Break Number	System	Description	Fig. No.	Location	Stress Ratio to Allowable (1)
BB1-01	Steam Gen. Blowdown Recycle Sys.	4" BB Blowdown "1A" T.E. @ M-300	3.10 Sheet 3 of 4	AB-716-07	Terminal End Break
BB1-02	Steam Gen. Blowdown Recycle Sys.	4" BB Blowdown "1B" T.E. @ M-301	3.10 Sheet 3 of 4	AB-733-06	Terminal End Break
BB1-03	Steam Gen. Blowdown Recycle Sys.	4" BB Blowdown "1C" T.E. @ M-303	3.10 Sheet 3 of 4	AB-733-06	Terminal End Break
BB1-04	Steam Gen. Blowdown Recycle Sys.	4" BB Blowdown "1D" T.E. @ M-304	3.10 Sheet 3 of 4	AB-716-07	Terminal End Break
BB1-36B	Steam Gen. Blowdown Recycle Sys.	2" Intermediate break @ inlet of 2nd tee downstream of valve 1BB3B	3.10 Sheet 3 of 4	AB-733-06	0.56
BB1-27B	Steam Gen. Blowdown Recycle Sys.	2" Intermediate break @ 1st standard elbow downstream of M-301	3.10 Sheet 3 of 4	AB-733-06	0.57
BB1-28	Steam Gen. Blowdown Recycle Sys.	2" Intermediate break @ outlet of 1st tee downstream of valve 1BB32	3.10 Sheet 4 of 4	AB-733-13	0.64
BB1-35	Steam Gen. Blowdown Recycle Sys.	2" Intermediate break @ outlet of 1st tee downstream of valve 1BB29	3.10 Sheet 4 of 4	AB-733-13	0.80

(1) Ratio = Stress, $S/(0.8 (1.2S_h + S_A))$; intermediate breaks only

Table 6-1 2 of 15

(Deleted)

TABLE 6-1

HIGH ENERGY PIPE BREAK LOCATION

BREAK NO.	SYSTEM	DESCRIPTION	FIG. NO.	LOCATION	STRESS RATIO TO ALLOWABLE (1)
CAI-43A	Auxiliary Feedwater	Inlet of elbow downstream of valve 1CF135A	3.11A	AB-750-23	0.836
CAI-43B	Auxiliary Feedwater	Outlet of elbow downstream of valve 1CF135A	3.11A	AB-750-23	0.882
CAI-76A	Auxiliary Feedwater	At inlet of elbow upstream of valve 1CF156	3.11 Sheet 7 of 8	AB-750-23	0.733
CAI-51	Auxiliary Feedwater	At 4" x 2" half coupling downstream of valve 1CF136A	3.11 Sheet 5 of 8	AB-750-23	0.784
CAI-50	Auxiliary Feedwater	At 6" takeoff from 18" CF line between valves 1CF119 and 1CF28	3.11 Sheet 5 of 8	AB-750-23	Terminal End Break
CAI-87	Auxiliary Feedwater	Terminal end at 6" takeoff from 16" x 16" x 6" tee between the valves 1CF30 and 1CF120	3.11 Sheet 5 of 8	AB-750-23	Terminal End Break
CAI-96	Auxiliary Feedwater	Terminal end at 6" takeoff from 18" x 18" x 6" tee downstream of valve 1CF135A	3.11 Sheet 5 of 8	AB-750-23	Terminal End Break
CAI-62	Auxiliary Feedwater	Terminal end at 6" takeoff from 18" x 18" x 6" tee downstream of valve 1CF136A	3.11 Sheet 5 of 8	AB-750-23	Terminal End Break

(1) Ratio = Stress, $S/[0.8(1.2 Sh + SA)]$; intermediate breaks only.

(2) Stress is less than $0.4(1.2 Sh + SA)$.

HIGH ENERGY PIPE BREAK LOCATION

BREAK NO.	SYSTEM	DESCRIPTION	FIG. NO.	LOCATION	STRESS RATIO TO ALLOWABLE (1)
CAI-113	Auxiliary Feedwater	Terminal end at motor driven FDW pump IA discharge	3.11 Sheet 6 of 8	AB-716-10	Terminal End Break
CAI-126	Auxiliary Feedwater	Terminal end at motor driven FDW pump IB discharge	3.11 Sheet 6 of 8	AB-716-10	Terminal End Break
CAI-70A	Auxiliary Feedwater	Intermediate break at inlet of elbow downstream of valve ICF158	3.11 Sheet 8 of 8	AB-750-26	0.966
CAI-70B	Auxiliary Feedwater	Intermediate break at outlet of elbow downstream of valve ICF158	3.11 Sheet 8 of 8	AB-750-26	0.990
CAI-110	Auxiliary Feedwater	Terminal end at 6" takeoff from 18" CF line upstream of valve ICF160	3.11 Sheet 4 of 8	AB-750-26	Terminal End Break
CAI-120	Auxiliary Feedwater	Terminal end at 6" takeoff from 16" x 16" x 6" tee downstream of valve ICF134A	3.11 Sheet 4 of 8	AB-750-26	Terminal End Break
CAI-109	Auxiliary Feedwater	Terminal end at 6" takeoff from 18" CF line upstream of valve ICF163	3.11 Sheet 4 of 8	AB-750-26	Terminal End Break
CAI-150	Auxiliary Feedwater	Terminal end at 6" takeoff from 16" x 16" x 6" tee downstream of valve ICF137A	3.11 Sheet 4 of 8	AB-750-26	Terminal End Break

(1) Ratio = Stress, $S/\{0.8(1.2 Sh + SA)\}$; intermediate breaks only.

(2) Stress is less than $0.4(1.2 Sh + SA)$.

TABLE 6-1
HIGH ENERGY PIPE BREAK LOCATIONS

BREAK NUMBER	SYSTEM	DESCRIPTION	FIGURE NO.	LOCATION	STRESS RATIO TO ALLOWABLE (1)
CF1-01	Feedwater	@ Inlet of first elbow before penet. M-262	Fig. 3.12 Sht. 1 of 6	AB-750-23	0.519
CF1-02	Feedwater	@ Outlet of first elbow before penet. M-262	Fig. 3.12 Sht. 1 of 6	AB-750-23	0.465
CF1-03	Feedwater	@ Inlet of first elbow before pene. M-308	Fig. 3.12 Sht. 1 of 6	AB-750-23	0.630
CF1-04	Feedwater	@ Outlet of first elbow before penet. M-308	Fig. 3.12 Sht. 1 of 6	AB-750-23	0.474
CF1-05	Feedwater	@ Inlet of first elbow before penet. M-153	Fig. 3.12 Sht. 1 of 6	AB-750-26	0.431
CF1-06	Feedwater	@ Outlet of first elbow before penet. M-153	Fig. 3.12 Sht. 1 of 6	AB-750-26	0.411
CF1-07	Feedwater	@ Inlet of first elbow before penet. M-440	Fig. 3.12 Sht. 1 of 6	AB-750-26	0.511
CF1-08	Feedwater	@ Outlet of first elbow before penet. M-440	Fig. 3.12 Sht. 1 of 6	AB-750-26	0.456

(1) Ratio = Stress, $S / (0.8 (1.2S_h + S_A))$; intermediate breaks only

(2) Stresses at all locations in the piping run are less than $0.4 (1.2S_h + S_A)$.

TABLE 6-1
HIGH ENERGY PIPE BREAK LOCATIONS

BREAK NUMBER	SYSTEM	DESCRIPTION	FIGURE NO.	LOCATION	STRESS RATIO TO ALLOWABLE (1)
CF1-09	Feedwater	18" CF to steam generator 1A @ feedwater header	See Turbine Building	TB-739	Terminal End Break
CF1-10	Feedwater	18" CF to steam generator 1B @ feedwater header	See Turbine Building	TB-739	Terminal End Break
CF1-11	Feedwater	18" CF to steam generator 1C @ feedwater header	See Turbine Building	TB-739	Terminal End
CF1-12	Feedwater	18" CF to steam generator 1D @ feedwater header	See Turbine Building	TB-739	Terminal End

(1) Ratio = Stress, $S / (0.8 (1.2S_h + S_A))$; intermediate breaks only

(2) Stresses at all locations in the piping run are less than $0.4 (1.2S_h + S_A)$

TABLE 6-1
HIGH ENERGY PIPE BREAK LOCATIONS

BREAK NUMBER	SYSTEM	DESCRIPTION	FIGURE NO.	LOCATION	STRESS RATIO TO ALLOWABLE (1)
N11-01	Safety Injection	12" NI-T.E. Brk. @ upper head injection nitrogen accumulator	3.13	AB-750-01	Terminal End Break (2)
N11-02	Safety Injection	12" NI-T.E. Brk. @ Penet. M-334	3.13	AB-750-01	Terminal End Break (2)
N11-03	Safety Injection	12" NI-T.E. Brk. @ Penet. M-349	3.13	AB-750-01	Terminal End Break (2)
N11-04	Safety Injection	12" NI- From UHI nitrogen accumulator to water accumulator, T.E. Brk. @ upper head inj. water accum.	3.13	AB-750-05	Terminal End Break (2)
N11-05	Safety Injection	12" NI-To Valve INI244B. T.E. Brk. @ upper Head Inj. water accumulator.	3.13	AB-750-05	Terminal End Break (2)
N11-06	Safety Injection	12" NI-To Valve INI242B. T.E. Brk. @ upper head inj. water accumulator.	3.13	AB-750-05	Terminal End Break (2)
N11-07	Safety Injection	1 1/2" NI-To Vlv. INI326. T.E. brk @ water accumulator.	3.13	AB-750-05	Terminal End Break (2)

(1) Ratio = Stress, $S / (0.8 (1.2S_h + S_A))$; intermediate breaks only.

(2) Stresses at all locations in the piping run are less than $0.4 (1.2S_h + S_A)$.

TABLE 6-1
HIGH ENERGY PIPE BREAK LOCATIONS

BREAK NUMBER	SYSTEM	DESCRIPTION	FIGURE NO.	LOCATION	STRESS RATIO TO ALLOWABLE (1)
N11-12	Safety Injection	2" NI To valve IN1320, T.E. Brk. @ Upper Head Inj. Water Accumulator	3.13	AB-750-05	Terminal End Break (2)
N11-13	Safety Injection	2" NI To valve IN1314 T.E. Brk. @ Water Accumu.	3.13	AB-750-05	Terminal End Break (2)

1. Ratio = Stress, S/ (0.8 (1.25_h + S_A)); intermediate breaks only.
2. Stresses at all locations in the piping run are less than 0.4 (1.25_h + S_A).

TABLE 6-1
HIGH ENERGY PIPE BREAK LOCATIONS

BREAK NUMBER	SYSTEM	DESCRIPTION	FIGURE NO.	LOCATION	STRESS RATIO TO ALLOWABLE (1)
N11-16	Safety Injection	1½" NI To valve INI280. T.E. Brk. @ Nitrogen Accumulator.	3.13	AB-750-01	Terminal End Break (2)
N11-17	Safety Injection	2" NI To valve INI2558. T.E. Brk. @ 12" NI Line	3.13	AB-750-05	Terminal End Break (2)
N11-18	Safety Injection	1½" To valve INI279. T.E. brk. @ Water Accumulator.	3.13	AB-750-05	Terminal End Break (2)

(1) Ratio = Stress, $S / (0.8 (1.25 S_h + S_A))$; intermediate breaks only.

(2) Stresses at all locations in the piping run are less than $0.4 (1.25 S_h + S_A)$.

TABLE 6-1
HIGH ENERGY PIPE BREAK LOCATIONS

BREAK NUMBER	SYSTEM	DESCRIPTION	FIGURE NO.	LOCATION	STRESS RATIO TO ALLOWABLE (1)
NR1-01	Boron Thermal Regeneration Sys.	4" NR to letdown reheat H.X.-T.E. Brk. @ Inlet of Letdown Reheat H.X.	3.14	AB-750-16	Terminal End Break (2)
NR1-02	Boron Thermal Regeneration Sys.	4" NR To letdown reheat H.X.-T.E. Brk. @ Outlet of letdown reheat H.X.	3.14	AB-750-16	Terminal End Break (2)

(1) Ratio = Stress, $S / (0.8 (1.25 S_h + S_A))$; intermediate breaks only.

(2) Stresses at all locations in the piping run are less than $0.4 (1.25 S_h + S_A)$.

Revision 3

TABLE 6-1
HIGH ENERGY PIPE BREAK LOCATIONS

BREAK NUMBER	SYSTEM	DESCRIPTION	FIGURE NO.	LOCATION	STRESS RATIO TO ALLOWABLE (1)
NV1-01	Chemical and Volume Control	3" NV -Letdown H.X. T.E. @ Inlet of Letdown H.X.	Fig. 3.15 Sht. 9 of 9	AB-750-17	Terminal End Break
NV1-02	Chemical and Volume Control	3" NV-From Letdown H.X., Outlet of Letdown H.X.	Fig. 3.15 Sht. 9 of 9	AB-750-17	Terminal End Break
NV1-03	Chemical and Volume Control	3" NV-To Regenerative H.X. T.E. Brk. @ Penet. M-329	Fig. 3.15 Sht. 2 of 9	AB-716-06	Terminal End Break
NV1-04	Chemical and Volume Control	T.E. Brk. @ Discharge of Reciprocating Charging Pmp	Fig. 3.15 Sht. 8 of 9	AB-716-01	Terminal End Break (2)
NV1-05	Chemical and Volume Control	4" NV To Cent. Charging Pump "IA". T.E. @ Cent. Chg. Pump IA Disch. Nozzle	Fig. 3.15 Sht. 2 of 9	AB-716-02	Terminal End Break (2)
NV1-06	Chemical and Volume Control	T.E. Brk. @ the Cent. Chg. Pump "IB". Discharge Nozzle	Fig. 3.15 Sht. 2 of 9	AB-716-03	Terminal End Break (2)
NV1-07	Chemical and Volume Control	3" NV to Letdown H.X. T.E Brk. @ M-347	Fig. 3.15 Sht. 6 of 9	AB-750-01	Terminal End Break
NV1-08	Chemical and Volume Control	T.E. Brk. @ Outlet of seal-water Inj. Filter IB	Fig. 3.15 Sheet 4 of 9	AB-733-11	Terminal End Break (2)

(1) Ratio = Stress, S/ (0.8 (1.2S_h + S_A)); intermediate breaks only.

(2) Stresses at all locations in the piping run are less than 0.4 (1.2S_h + S_A).

TABLE 6-1
HIGH ENERGY PIPE BREAK LOCATIONS

BREAK NUMBER	SYSTEM	DESCRIPTION	FIGURE NO.	LOCATION	STRESS RATIO TO ALLOWABLE (1)
NVI-09	Chemical and Volume Control	T.E. Brk. @ Inlet of seal water Inj. Filter "IA."	Fig. 3.15 Sht. 4 of 9	AB-733-10	Terminal End Break (2)
NVI-10	Chemical and Volume Control	T.E. Brk. @ Outlet of seal water Inj. Filter "IA."	Fig. 3.15 Sht. 4 of 9	AB-733-10	Terminal End Break (2)
NVI-11	Chemical and Volume Control	T.E. Brk. @ Inlet of Seal water Inj. Filter "IB."	Fig. 3.15 Sht. 4 of 9	AB-733-11	Terminal End Break (2)
NVI-12	Chemical and Volume Control	Int. Brk. in 3" NV to Letdown Reheat H.X.	Fig. 3.15 Sht. 3 of 9	AB-750-17	0.5032
NVI-13	Chemical and Volume Control	T.E. Brk. @ Penetration M-339 4" dia.	Fig. 3.15 Sht. 5 of 9	AB-733-7	Terminal End Break (2)
NVI-14	Chemical and Volume Control	T.E. Brk. @ Penetration M-344 4" dia.	Fig. 3.15 Sht. 5 of 9	AB-733-7	Terminal End Break (2)
NVI-15	Chemical and Volume Control	T.E. Brk. @ Penetration M-343 4" dia.	Fig. 3.15 Sht. 2 of 9	AB-733-7	Terminal End Break (2)
NVI-16	Chemical and Volume Control	T.E. Brk. @ Penetration M-350 4" dia.	Fig. 3.15 Sht. 2 of 9	AB-733-7	Terminal End Break (2)

(1) Ratio = Stress, $S / (0.8 (1.2S_h + S_A))$; intermediate breaks only.

(2) Stresses at all locations in the piping run are less than $0.4 (1.2S_h + S_A)$.

TABLE 6-1
HIGH ENERGY PIPE BREAK LOCATIONS

BREAK NUMBER	SYSTEM	DESCRIPTION	FIGURE NO.	LOCATION	STRESS RATIO TO ALLOWABLE (1)
SA1-01	Main Steam Supply to Aux. Equip.	6" SA To CA Turbine T.E.brk @SM Line 1B	Fig. 3.16 Sht. 3 of 3	AB-750-23	Terminal End Break
SA1-02	Main Steam Supply to Aux. Equip.	6" SA To CA Turbine T.E. Brk. @ SM Line 1C	Fig. 3.16 Sht. 3 of 3	AB-750-23	Terminal End Break
SA1-03	Main Steam Supply to Aux. Equip.	6" SA Int. Brk. @ Inlet of valve ISA48	Fig. 3.16 Sht. 3 of 3	AB-750-23	0.5543

(1) Ratio = Stress, $S / (0.8 (1.25 S_h + S_A))$; intermediate breaks only.

(2) Stresses at all locations in the piping run are less than $0.4 (1.25 S_h + S_A)$

TABLE 6-1
HIGH ENERGY PIPE BREAK LOCATIONS

BREAK NUMBER	SYSTEM	DESCRIPTION	FIGURE NO.	LOCATION	STRESS RATIO TO ALLOWABLE (1)
SMI-01	Main Steam	Inter. Brk. @ Inlet 1st (SM) elbow downstream of M-261	Fig. 3.16 Sht. 1 of 3	AB-750-23	0.3258
SMI-02	Main Steam	Inter. Brk. @ outlet 1st (SM) elbow downstream of M-261	Fig. 3.16 Sht. 1 of 3	AB-750-23	0.365
SMI-03	Main Steam	Inter. Brk. @ Inlet 1st (SM) elbow downstream of M-393	Fig. 3.16 Sht. 1 of 3	AB-750-23	0.493
SMI-04	Main Steam	Inter. Brk. @ Outlet 1st (SM) elbow downstream of M-393	Fig. 3.16 Sht. 1 of 3	AB-750-23	0.520
SMI-05	Main Steam	Inter. Brk. @ Inlet of 1st (SM) elbow downstream of M-154	Fig. 3.16 Sht. 1 of 3	AB-750-26	0.323
SMI-06	Main Steam	Inter. Brk. @ Outlet 1st (SM) elbow downstream of M-154	Fig. 3.16 Sheet 1 of 3	AB-750-26	0.362
SMI-07	Main Steam	Inter. Brk. @ Inlet of 1st (SM) elbow downstream of M-441.	Fig. 3.16 Sht. 1 of 3	AB-750-26	0.486

(1) Ratio = Stress, $S / (0.8 (1.2S_h + S_A))$; intermediate breaks only.

(2) Stresses at all locations in the piping run are less than $0.4 (1.2S_h + S_A)$

TABLE 6-1
HIGH ENERGY PIPE BREAK LOCATIONS

BREAK NUMBER	SYSTEM	DESCRIPTION	FIGURE NO.	LOCATION	STRESS RATIO TO ALLOWABLE (1)
SMI-08	Main Steam	Inter. Brk. @ Outlet of 1st (SM) elbow downstream of M-441.	Fig. 3.16 Sht. 1 of 3	AB-750-26	0.510
SMI-09	Main Steam	34" (SM) 1A @ High Pressure Turbine T.E. @ Inlet of H.P. Turbine	Turbine Bldg.	TB-786	Terminal End Break
SMI-10	Main Steam	34" (SM) 1B @ High Pressure Turbine T.E. Brk. @ Inlet of H.P. Turbine	Turbine Bldg.	TB-786	Terminal End Break
SMI-11	Main Steam	34" (SM) 1C @ High Pressure Turbine T.E. Brk. @ Inlet of H. P. Turbine	Turbine Bldg.	TB-786	Terminal End Break
SMI-12	Main Steam	34" (SM) 1D @ H.P. Turbine T.E. Brk. @ Inlet of H.P. Turbine.	Turbine Bldg.	TB-786	Terminal End Break

(1) Ratio = Stress, $S / (0.8 (1.2S_h + S_A))$; intermediate breaks only

(2) Stresses at all locations in the piping run are less than $0.4 (1.2S_h + S_A)$.

TABLE 6-2
 RESULTS OF DAMAGE EVALUATION
 UNACCEPTABLE PIPE WHIP/JET IMPINGEMENT EFFECTS

BREAK NUMBER	TYPE BREAK	EFFECT ON REQUIRED COMPONENTS	RESOLUTION
BB1-01	Circumferential (forward flow)	Pipe whip damage to: Valve 1NI173A, 1NI121A. Jet impingement damage to: Vlv. 1NI173A.	Added pipe whip restraint
BB1-03	Circumferential (forward flow)	Pipe whip damage to: Valve 1NI173A, 1NI121A. Jet damage to valve 1NI173A.	Add pipe whip restraint

3

TABLE 6-2
 RESULTS OF DAMAGE EVALUATION
 UNACCEPTABLE PIPE WHIP/JET IMPINGEMENT EFFECTS

BREAK NUMBER	TYPE BREAK	EFFECT ON REQUIRED COMPONENTS	RESOLUTION
CA1-51	Circumferential (forward flow)	Pipe whip and jet impingement on essential electrical cable to valve ICA50B.	Add whip restraint and jet barrier.
CA1- 70A	Circumferential (forward flow)	Jet impingement on essential electrical cables to valves ICA38B and ICA42B.	Add jet barrier

3

TABLE 6-2
RESULTS OF DAMAGE EVALUATION
UNACCEPTABLE PIPE WHIP/JET IMPINGEMENT EFFECTS

BREAK NUMBER	TYPE BREAK	EFFECT ON REQUIRED COMPONENTS	RESOLUTION
CF1-01	Circumferential (forward flow) (reverse flow)	Unacceptable damage to Doghouse structure and essential components located inside the doghouse.	Add pipe whip restraint.
CF1-02	Circumferential (forward flow) (reverse flow)	Unacceptable damage to Doghouse structure and essential component located inside the doghouse.	Add pipe whip restraints.
CF1-03	Circumferential (forward flow) (reverse flow)	Unacceptable damage to Doghouse structure and essential components located inside the doghouse.	Add pipe whip restraint.
CF1-04	Circumferential (forward flow) (reverse flow)	Unacceptable damage to Doghouse structure and essential components located inside the doghouse.	Add pipe whip restraints.
CF1-05	Circumferential (forward flow) (reverse flow)	Unacceptable damage to Doghouse structure and essential components located inside the doghouse.	Add pipe whip restraint.
CF1-06	Circumferential (forward flow) (reverse flow)	Unacceptable damage to Doghouse structure and essential components located inside the doghouse.	Add pipe whip restraints.
CF1-07	Circumferential (forward flow) (reverse flow)	Unacceptable damage to Doghouse structure and essential components located inside the doghouse.	Add pipe whip restraint.
CF1-08	Circumferential (forward flow) (reverse flow)	Unacceptable damage to Doghouse structure and essential components located inside the doghouse.	Add pipe whip restraints.

TABLE 6-2
RESULTS OF DAMAGE EVALUATION
UNACCEPTABLE PIPE WHIP/JET IMPINGEMENT EFFECTS

BREAK NUMBER	TYPE BREAK	EFFECT ON REQUIRED COMPONENTS	RESOLUTION
N11-02	Circumferential (forward flow)	Pipe whip damage to Column at HH and 52	Add pipe whip restraint.
NV1-03	Circumferential (forward and reverse flow)	Essential electrical components	Add pipe whip restraint.
NV1-07	Circumferential (forward flow)	Pipe whip damage to: a) 8" NS - To M-362 b) NSR Electrical "B" c) Valves: INS32A, INS43A, INV7B Jet damage to: a) NSR Electrical "B" b) NS Valves c) 8" RN - To Pond d) 8" ND/NS - To M-369 e) 8" NS - To M-362	Add pipe whip restraint.
NV1-07	Circumferential (reverse flow)	Jet damage to: a) NSR Electrical "B" b) 8" NS - To M-362 c) Valve INV7B	

TABLE 6-2
RESULTS OF DAMAGE EVALUATION
UNACCEPTABLE PIPE WHIP/JET IMPINGEMENT EFFECTS

BREAK NUMBER	TYPE BREAK	EFFECT ON REQUIRED COMPONENTS	RESOLUTION
SMI-01	Circumferential (forward flow) (reverse flow)	Unacceptable damage to Doghouse structure and essential components located in the doghouse.	Add pipe whip restraint.
SMI-02	Circumferential (forward flow) (reverse flow)	Unacceptable damage to Doghouse structure and essential components located in the doghouse.	Add pipe whip restraint.
SMI-03	Circumferential (forward flow) (reverse flow)	Unacceptable damage to doghouse structure and essential components located inside the doghouse.	Add pipe whip restraint.

TABLE 6-2
RESULTS OF DAMAGE EVALUATION
UNACCEPTABLE PIPE WHIP/JET IMPINGEMENT EFFECTS

BREAK NUMBER	TYPE BREAK	EFFECT ON REQUIRED COMPONENTS	RESOLUTION
SMI-04	Circumferential (forward flow) (reverse flow)	Unacceptable damage to doghouse structure and essential components located inside the doghouse.	Add pipe whip restraint.
SMI-05	Circumferential (forward flow) (reverse flow)	Unacceptable damage to doghouse structure and essential components located inside the doghouse.	Add pipe whip restraint.
SMI-06	Circumferential (forward flow) (reverse flow)	Unacceptable damage to doghouse structure and essential components located inside the doghouse.	Add pipe whip restraint.
SMI-07	Circumferential (forward flow) (reverse flow)	Unacceptable damage to doghouse structure and essential components located inside the doghouse.	Add pipe whip restraint.
SMI-08	Circumferential (forward flow) (reverse flow)	Unacceptable damage to doghouse structure and essential components located inside the doghouse.	Add pipe whip restraint.

TABLE 6-3
RESULTS OF DAMAGE EVALUATION
WATER SPRAY - MODERATE ENERGY PIPING

Attachment # 2 Sheet 1 of 7

LOCATION (SEE FIGURES 2-2 thru 2-6)	EFFECT ON ESSENTIAL COMPONENTS	RESOLUTION
AB-695-1	Water spray affecting the following: TB212	Add seal to cabinet doors and add sealant to cable entrance.
AB-695-3	Water spray affecting the following: TB211	Add seal to cabinet doors and add sealant to cable entrance.
AB-716-06 AB-733-07	Water spray affecting the following: a) TB182 b) TB183 c) TB189 d) TB184 e) TB185	Add seal to cabinet doors add sealant to cable entrance.
AB-716-12 AB-716-13	Water spray affecting the following: a) TB108 h) ATB149 b) TB188 i) TB150 c) TB201 d) TB113 e) IATC21 f) IATC3 g) TB192	Add seal to cabinet doors add sealant to cable entrance.

TABLE 6-3
RESULTS OF DAMAGE EVALUATION
WATER SPRAY - MODERATE ENERGY PIPING

Sheet 2 of 7

LOCATION (SEE FIGURES 2-2 thru 2-6)	EFFECT ON ESSENTIAL COMPONENTS	RESOLUTION
AB-716-10	Water spray affecting the following: a) TB406 b) TB232 c) TB112 d) TB203 e) Aux. Shutdown Panel	Add seal to cabinet doors, add sealant to cable entrance.
AB-716-53	Water spray affecting the following: a) TB407 b) TB195 c) TB204 d) TB202 e) TB193 f) TB233	Add seal to cabinet doors, add sealant to cable entrance.
AB-733-42	Water spray affecting the following: a) TB111	Add seal to cabinet doors, and add sealant to cable entrance.

Revision 3

TABLE 6-3
RESULTS OF DAMAGE EVALUATION
WATER SPRAY - MODERATE ENERGY PIPING

Sheet 3 of 7

LOCATION (SEE FIGURES 2-2 thru 2-6)	EFFECT ON ESSENTIAL COMPONENTS	RESOLUTION
AB-733-16 AB-733-17	Water spray affecting the following: a) TB107 b) IATC22 c) IATC4A d) IATC4	Add seal to cabinet doors, add sealant to cable entrance.
AB-750-01	Water spray affecting the following: a) TB415 b) TB416 c) TB146 d) TB224 e) TB239 f) TB225 g) TB238	Add seal to cabinet doors, and add sealant to cable entrance.
AB-750-09	Water spray affecting the following: a) Refueling water storage tank panel.	Add seal to panel doors, add sealant to cable entrance.
AB-750-18 AB-750-19 AB-750-20	Water spray affecting the following: a) IATC2A d) TB148 b) IATC2 e) TB147 c) TB109 f) Nuclear Ser. Water Relay Cabinet # 1	Add seal to cabinet doors, add sealant to cable entrance.

3

Revision 3

TABLE 6-3
RESULTS OF DAMAGE EVALUATION
WATER SPRAY - MODERATE ENERGY PIPING

Sheet 4 of 7

LOCATION (SEE FIGURES 2-2 thru 2-6)	EFFECT ON ESSENTIAL COMPONENTS	RESOLUTION
AB-750-26	Water spray affecting the following: a) TB262	Add seal to cabinet doors, add sealant to cable entrance.
AB-767-05	Water spray affecting the following: a) IRBEC P5	Add seal to cabinet doors, add sealant to cable entrance.
AB-750-23	Water spray affecting the following: a) TB260 b) ISMTC	Add seal to cabinet doors, add sealant to cable entrance.
AB-767-11	Water spray affecting the following: a) IATC10 b) IRBEC P4	Add seal to cabinet doors, add sealant to cable entrance.

Revision 3

TABLE 6-3
RESULTS OF DAMAGE EVALUATION
WATER SPRAY - MODERATE ENERGY PIPING

Sheet 5 of 7

LOCATION (SEE FIGURES 2-2 thru 2-6)	EFFECT ON ESSENTIAL COMPONENTS	RESOLUTION
AB-767-08	Water spray affecting the following: a) HVAC-A b) HVAC-B c) R-CP-3 d) TB368 e) TB367 f) IATC11 g) CRAC-1 h) CRAC-2 i) CRA-OAPFT-2 j) CRACP1 k) CRACP2 l) R-CP-2 m) R-CP-1 n) CRA-OAPFT-1	Add seal to cabinet doors, add sealant to cable entrance.
AB-767-10	Water Spray affecting the following: a) IEMXG	Add spray shield

Revision 3

TABLE 6-3

RESULTS OF WATER SPRAY EVALUATION

SH. 6 of 7

LOCATION (SEE FIG. 2.2 - 2-6)	EFFECT ON ESSENTIAL COMPONENTS	RESOLUTION
AB-733 Unit 1 Switchgear Room	1ELXD 1EMXB4 Busswork D/G Relay Cab. 1B 1ETB D/G Load Seq. Pnl. 1B	Add spray shields.
AB-750 Unit 1 Cable Splitting Room	Aux. Relay Cabinet	Add spray shields.
AB-750 Unit 1 Switchgear Room	1ELXC 1ELXE 1SSFARC D/G Load Seq. Pnl. 1A 1ETA 1ATC7 1ATC23 D/G Relay Cab. 1A	Add spray shields.
AB-767 Unit 1 Elec. Penetration Room	West RX Trip Switchgear	Add spray shields.
AB-767-8	2EMXG	Add spray shields.
AB-767-10	1EMXG	Add spray shields.
AB-750-27	2EMXA 2EMXA-1	Add spray shields.

TABLE 6-3

RESULTS OF WATER SPRAY EVALUATION

S^h. 7 of 7

LOCATION (SEE FIG. 2.2 - 2-6)	EFFECT ON ESSENTIAL COMPONENTS	RESOLUTION
AB-733 Unit 4 Switchgear Room	D/G Load Seq. Pnl. 2B 2ATC9 D/G Relay Cab. 2B 2ETB 2ELXF 2ELXD	Add spray shields.
AB-750 Unit 2 Cable Splitting Room	Aux. Relay Cab. 2IR2 2IR3 2IC8 2IC5	Add spray shields.
AB-750 Unit 2 Switchgear Room	D/G Load Seq. Pnl. 2A 2ATC7 2ATC23 2ETA 2ELXC 2EMXA-3	Add spray shields.
AB-767 Unit 2 Elec. Penetration Room	West RX Trip Switchgear	Add spray shields.
AB-750 21	1EMXA	Add spray shields.
AB-733-17	2EMXA	Add spray shields.

TABLE 6-4
MCGUIRE NUCLEAR STATION
DOGHOUSE PRESSURE RISE ANALYSIS

Data for TMD Model

1. Doghouse ambient conditions

Temperature = 135°F

Relative Humidity = 20%

Barometric Pressure = 28.06-30.14 In Hg

2. Compartment Net Volumes

<u>Compartment Number</u>	<u>Compartment Net Volume (FT³)</u>
1	2683
2	2824
3	4834
4	1743
5	2674
6	2603
7	2998
8	3388
9	4981
10	2403
11	3080
12	2505
13	4221
14	4360
15	6598
16	3397
17	3619
18	3315
19	2840
20	2530
21	4154
22	2953
23	2668
24	2953

TABLE 6-4 (Continued)

3. Flow Path Minimum Areas, Inertial Lengths, and Loss Coefficients (K-Factor)

<u>Flow Path</u>	<u>Minimum Area (FT²)</u>	<u>Inertial Length (FT)</u>	<u>K-Factor</u>
1-2	38	7.75	0.86
1-4	12	3.38	1.37
1-7	119	12.73	0.24
2-3	158	13.54	0.10
2-5	107	6.09	0.76
2-8	135	12.50	0.24
3-6	78	5.98	0.90
3-9	182	11.23	0.32
4-5	82	15.57	0.08
4-10	66	10.05	0.34
5-6	109	16.93	0.08
5-11	120	13.98	0.08
6-12	107	13.32	0.10
7-8	101	13.64	0.26
7-10	96	6.24	0.77
7-13	118	13.92	0.28
8-9	133	13.23	0.28
8-11	144	7.23	0.58
8-14	135	15.05	0.28
9-12	102	6.91	0.75
9-15	199	14.33	0.26
10-11	112	17.00	0.06
10-16	84	13.76	0.28
11-12	112	17.00	0.08
11-17	80	11.77	0.48
12-18	74	11.63	0.48
13-14	191	18.54	0.06
13-16	110	5.71	0.77
13-19	135	11.48	0.39
14-15	174	14.04	0.34
14-17	132	4.53	0.80
14-20	125	11.04	0.37
15-18	93	6.02	0.90
15-21	232	12.78	0.19
16-17	154	18.99	0.04
16-ATM(Front)	40	4.40	1.41
16-22	134	13.32	0.58
17-18	144	17.80	0.17
17-ATM(Front)	54	4.50	1.41
17-23	126	12.99	0.27
18-24	136	13.64	0.24
18-ATM(Front)	38	4.38	1.42
19-20	94	18.34	1.74
19-22	187	10.90	0.98
19-ATM(Side)	99	11.10	2.71
20-21	104	16.38	1.96
20-23	165	10.76	0.96
21-24	187	10.90	0.98
21-ATM(Side)	208	11.10	2.71
22-23	88	13.47	2.04
22-ATM(Front)	187	7.16	1.94
22-ATM(Side)	88	8.33	2.87
23-24	88	13.47	2.04

TABLE 6-4 (Continued)

3. (Continued)

<u>Flow Path</u>	<u>Minimum Area (FT²)</u>	<u>Inertial Length (FT)</u>	<u>K-Factor</u>
23-ATM(Front)	165	6.78	1.92
24-ATM(Front)	187	7.16	1.94
24-ATM(Side)	88	8.33	1.87

4. Blowdown Flowrates

a. Main Steam

Blowdown in compartment # 1

Constant flow = 2300 lbm/sec

Enthalpy = 1188.0 btu/lbm

b. Main Steam

Blowdown in compartment # 2

Constant flow = 4830 lbm/sec

Enthalpy = 1188.0 btu/lbm

c. Main Steam

Blowdown in compartment # 14 or #17

Constant flow = 4830 lbm/sec

Enthalpy = 1188.0 btu/lbm

d. Feedwater Blowdown in compartments 3, 4, 6, & 9

<u>Flow Direction</u>	<u>Time (sec)</u>	<u>Enthalpy (Btu/lbm)</u>	<u>Total Flow</u>
Forward	$0 < t < 1.648$	424.6	7090.0
Forward	$1.648 < t$	424.6	5252.2
Reverse	$0 < t < 1.665$	424.6	7090.0
Reverse	$1.665 < t$	0	0

5. Computer Runs

<u>Run #</u>	<u>System</u>	<u>Location of blowdown</u>	<u>Time Duration</u>
1	Main Steam	Compartments #1 and # 14 simultaneously	2.00 sec
2	Main Steam	Compartments # 2 and # 17 simultaneously	2.00 sec
3	Feedwater	Compartment # 3	2.00 sec
4	Feedwater	Compartment # 4	2.00 sec
5	Feedwater	Compartment # 6	2.00 sec
6	Feedwater	Compartment # 9	2.00 sec

TABLE 6-5
LIST OF SYSTEM LINE CONDITIONS, BREAK AREAS, MASS AND ENERGY RELEASE RATES

System	Pipe Size (in)		Type	Area ⁽¹⁾ (in ²)	Line Fluid Temp. (°F)	Line Fluid Pressure (psia)	Mass Flux (#m/sec-ft ²)	Energy Flux (Btu/sec-ft ²)
AS	2"		Crack	0.08	298	65	138	163069
BB	4"		Circum	(2)	557	1107	(2)	(2)
BB	2"		Circum	(2)	557	1107	(2)	(2)
ND	18"		Crack	1.617	350	365	2888	930514
ND	14"		Crack	0.844	350	365	2888	930514
ND	8"		Crack	0.508	350	365	2888	930514
ND	3"		Crack	0.166	350	365	2888	930514
NV	3"		Circum	7.393	350	365	2888	930514
NB	6"		Crack	0.42	298	65	138	163069
WL	4"		Crack	0.239	298	65	138	163069
YH	6"		Crack	28.890	240	140	(3)	(3)
YH	5"		Crack	20.005	240	140	(3)	(3)
YH	4"		Crack	12.73	240	140	(3)	(3)
YH	3"		Crack	7.393	240	140	(3)	(3)
YH	2½"		Crack	4.778	240	140	(3)	(3)
YH	1½"		Crack	2.036	240	140	(3)	(3)
SM	31.5		Circum	291.5(4)	557	1107	2300	2732400

(1) Circumferential areas shown here are for single ended break only.

(2) See text. (Section 6.4.2.2.2.)

(3) Non-flashing liquid, included for water spray only.

(4) Area considers guard pipe restrictions and is total flow area.

TABLE 6-6

ESSENTIAL COMPONENTS WITH EXCESSIVE
ENVIRONMENTAL CONSEQUENCES

SH. 1 of 5

LOCATION	EQUIPMENT	DESIGN TEMPERATURE	PIPE BREAK TEMPERATURE	RESOLUTION
El. 716'	TB1366	150	212	Requalified for environment.
	1ATC 21	150	212	Requalified for environment.
	Nuclear Service Water Pump Motors	104	212	Requalified for environment.
	Centrifugal Charging Pump Motors	110	212	Requalified for environment.
	Safety Injection Pump Motors	110	212	Requalified for environment.
	2ATC3	150	212	Requalified for environment.
	2ATC21	150	212	Requalified for environment.
	2ATC6	165	212	Requalified for environment.
	TB366	150	212	Requalified for environment.
	1ATC3	150	212	Requalified for environment.
TB408	150	212	Requalified for environment.	

TABLE 6-6

ESSENTIAL COMPONENTS WITH EXCESSIVE
ENVIRONMENTAL CONSEQUENCES

SH. 2 of 5

LOCATION	EQUIPMENT	DESIGN TEMPERATURE	PIPE BREAK TEMPERATURE	RESOLUTION
EI. 733'	2ATC4	150	212	Requalified for environment.
	2ATC4A	150	212	Requalified for environment.
	2ATC22	150	212	Requalified for environment.
	TB1184	311	350 (JET)	Requalified for environment.
	1ATC22	125	212	Requalified for environment.
	1ATC4	150	212	Requalified for environment.
	1ATC4A	125	212	Requalified for environment.
	Nuclear Service Water Cabinet #2	150	212	Requalified for environment.
Component Cooling Pump Motors	104	212	Requalified for environment.	

TABLE 6-6

ESSENTIAL COMPONENTS WITH EXCESSIVE
ENVIRONMENTAL CONSEQUENCES

SH. 3 of 5

LOCATION	EQUIPMENT	DESIGN TEMPERATURE	PIPE BREAK TEMPERATURE	RESOLUTION
El. 750'	TB415	150	293 (JET)	Rupture restraint added for break.
	TB416	150	293 (JET)	Rupture restraint added for break.
	IATC2	150	240 (JET)	Detailed stress analysis showed all stresses less than 0.4 (1.2 Sh + SA).
	IATC2A	150	240 (JET)	Detailed stress analysis showed all stresses less than 0.4 (1.2 Sh + SA).
	Nuclear Service Water Cabinet #1	150	240 (JET)	Detailed stress analysis showed all stresses less than 0.4 (1.2 Sh + SA).
	Component Cooling Pump Motors	104	240 (JET)	Detailed stress analysis showed all stresses less than 0.4 (1.2 Sh + SA).
	2EMXA	125	240 (JET)	Spray shield added.
	2EMXA-1	125	240 (JET)	Spray shield added.
	Feedwater Level Control Panel 2	110	208	Requalified for environment.
	TB1238	185	191	Requalified for environment.
	TB1239	185	191	Requalified for environment.
TB1489	150	208	Requalified for environment.	

TABLE 6-6

ESSENTIAL COMPONENTS WITH EXCESSIVE
ENVIRONMENTAL CONSEQUENCES

SH. 4 of 5

LOCATION	EQUIPMENT	DESIGN TEMPERATURE	PIPE BREAK TEMPERATURE	RESOLUTION
E1. 750' (cont.)	TB1490	150	208	Requalified for environment.
	TB1491	150	208	Requalified for environment.
	2ATC2	150	240 (JET)	Requalified for environment.
	2ATC2A	150	240 (JET)	Requalified for environment.
	TB1415	150	191	Requalified for environment.
	TB1416	150	191 350 (JET)	Requalify for environment. Rupture restraint added for break.
	1EMXH	125	240 (JET)	Spray shield added.
	TB416	150	350 (JET)	Rupture restraint added for break.
	Feedwater Level Control Panel 1	114	132	Requalify for environment.

TABLE 6-6

ESSENTIAL COMPONENTS WITH EXCESSIVE
ENVIRONMENTAL CONSEQUENCES

SH. 5 of 5

LOCATION	EQUIPMENT	DESIGN TEMPERATURE	PIPE BREAK TEMPERATURE	RESOLUTION
El. 767'	2ATC10	90	130 240 (JET)	Requalify for environment.
	2RB-ECP-4A	150	240 (JET)	Requalify for environment.
	TB1442	150	240 (JET)	Requalify for environment.
	TB1443	150	240 (JET)	Requalify for environment.
	ISMTC1	150	204	Requalify for environment.
	IATC10	90	130 240 (JET)	Requalify for environment.
	TB442	150	240 (JET)	Requalify for environment.
	TB443	150	240 (JET)	Requalify for environment.
	Doghouse	TB260	150	330
TB261		311	330	Relocate to auxiliary building.
TB262		311	330	Relocate to auxiliary building.

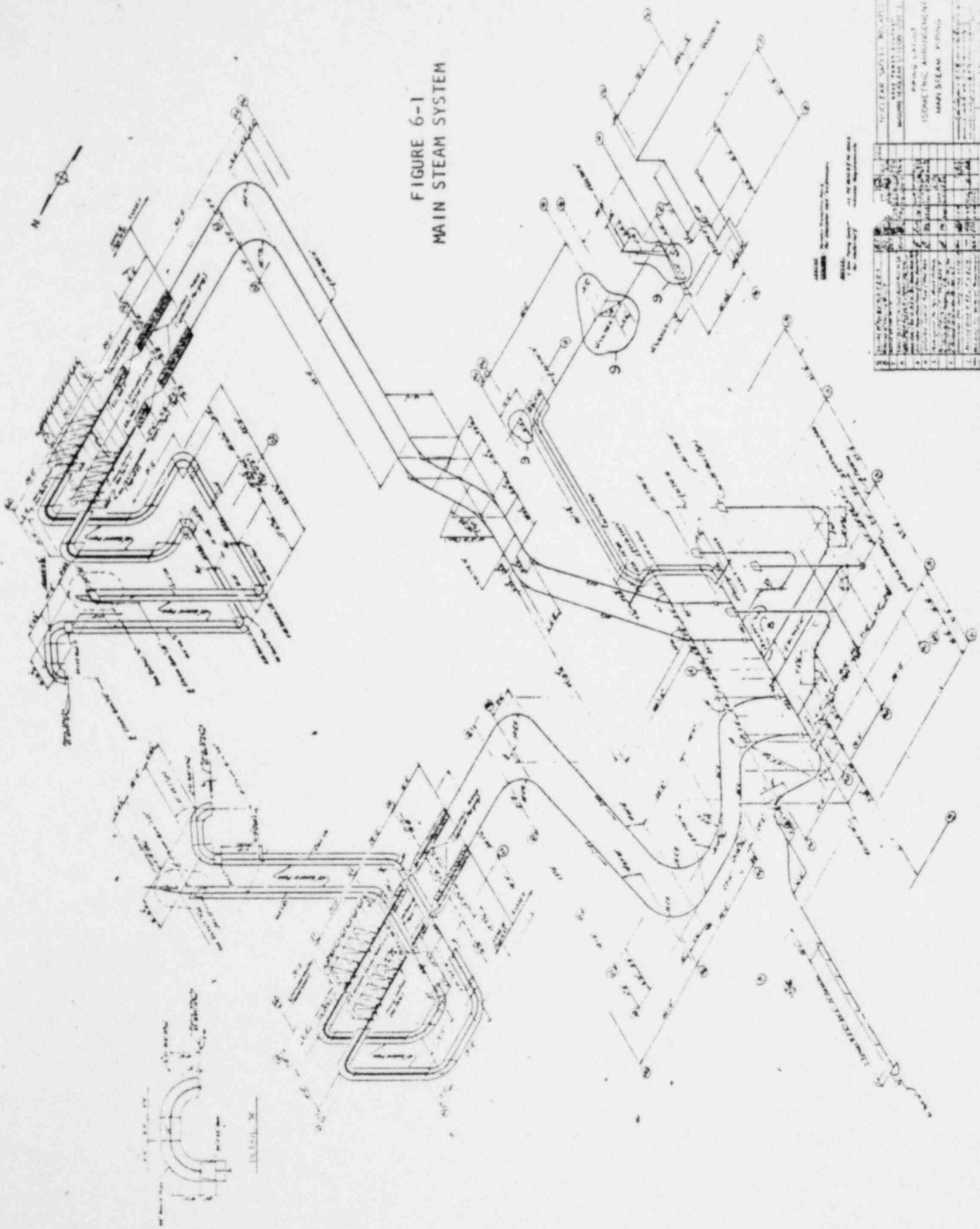


FIGURE 6-1
MAIN STEAM SYSTEM

DRAWN BY: [Name]
 CHECKED BY: [Name]
 DATE: [Date]
 SCALE: [Scale]

REVISIONS		APPROVALS	
NO.	DESCRIPTION	DATE	BY
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10	REVISED TO SHOW [Change]		

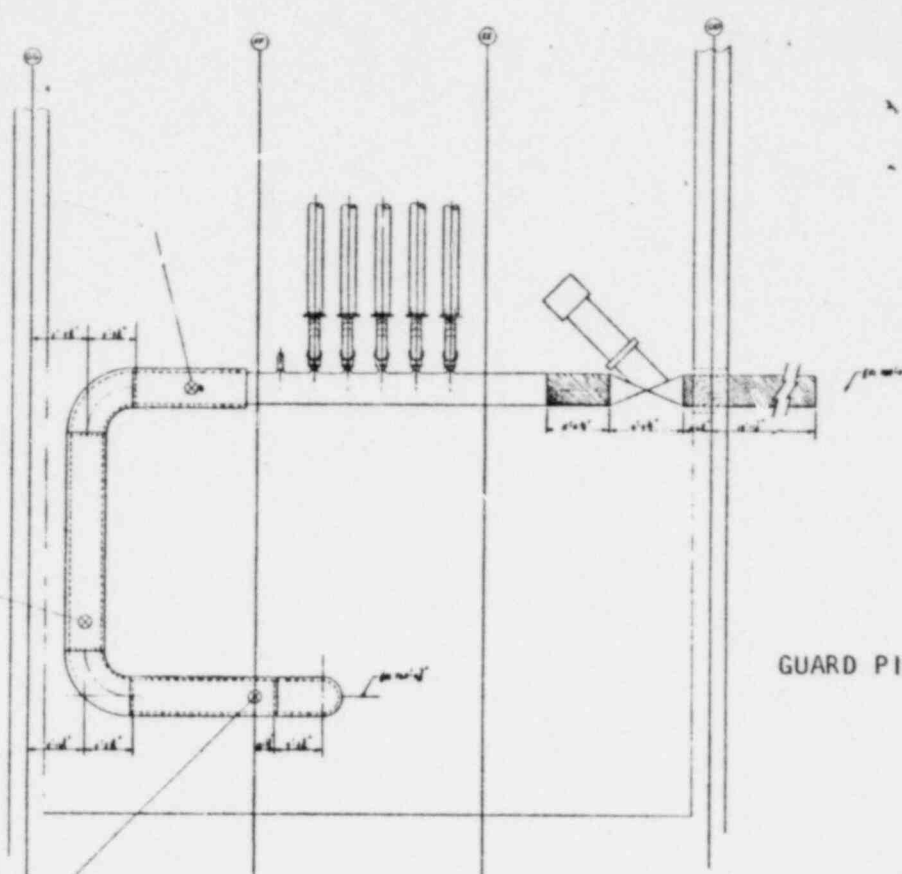
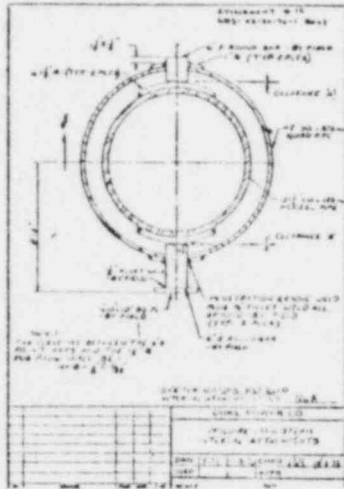
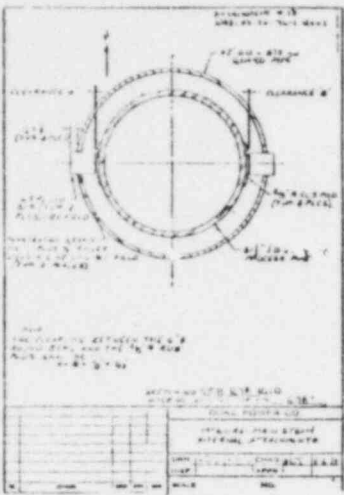
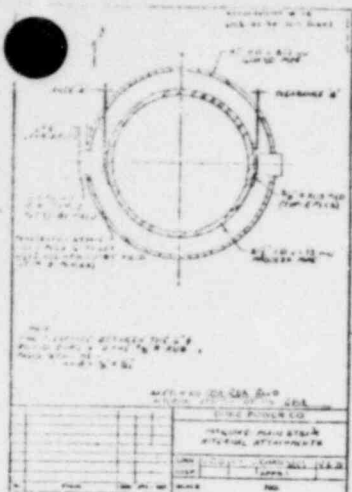


FIGURE 6-2
GUARD PIPE/PROCESS PIPE DETAILS

SMA-WD-18
 Material: Mild Steel
 1/2" NOM. EAST-WEST VIEW
 SWAGING INTERNAL GASKETS

REVISIONS			BILL OF MATERIALS		
NO.	DATE	DESCRIPTION	QTY	UNIT	PRICE

PIPE RUPTURE RESTRAINT (SEE
DRAWING) ENT-INST
(SEE QUOTE PAGE)

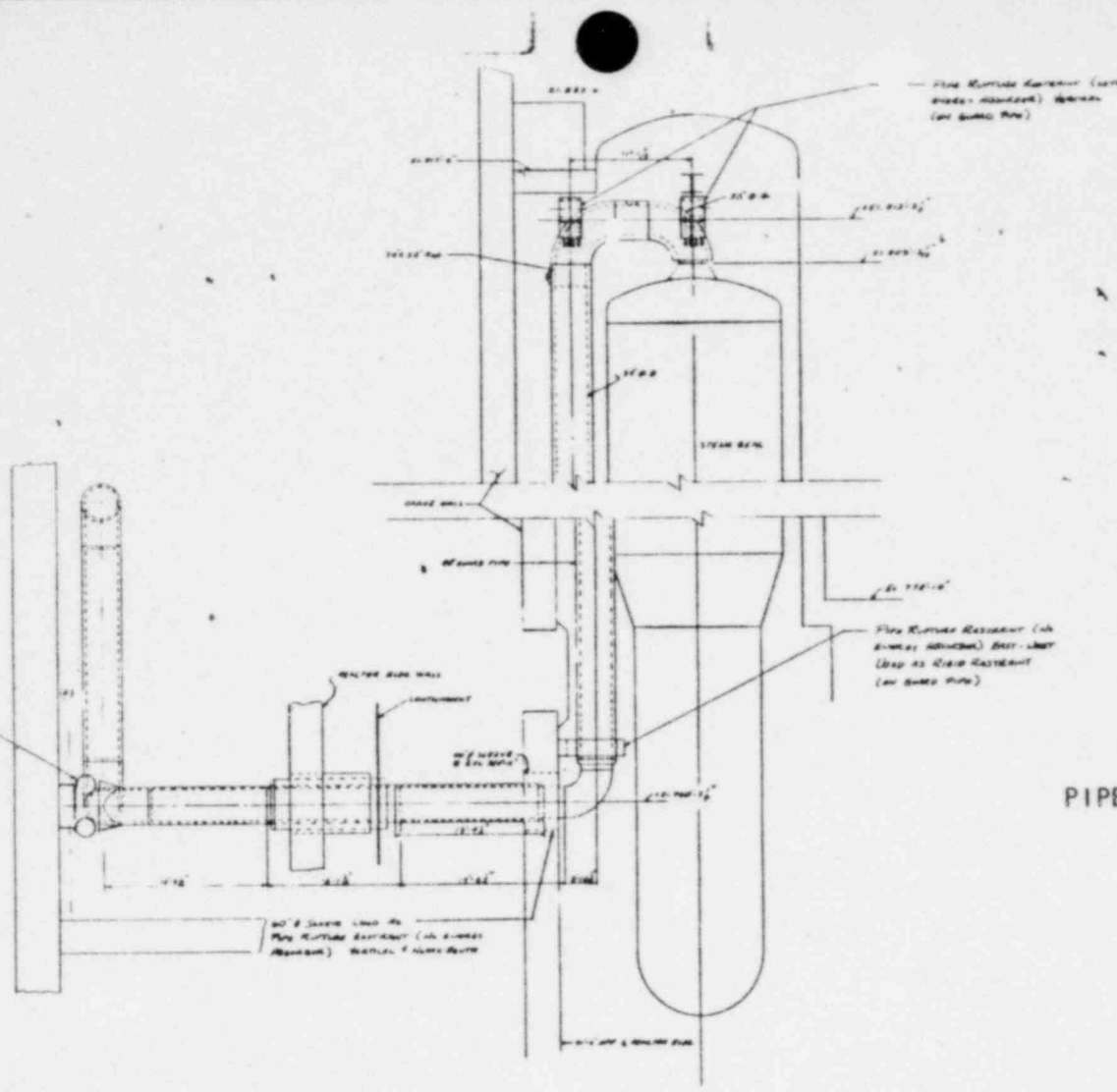


FIGURE 6-3
PIPE RUPTURE RESTRAINTS

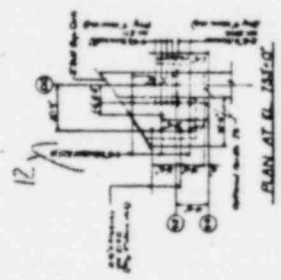
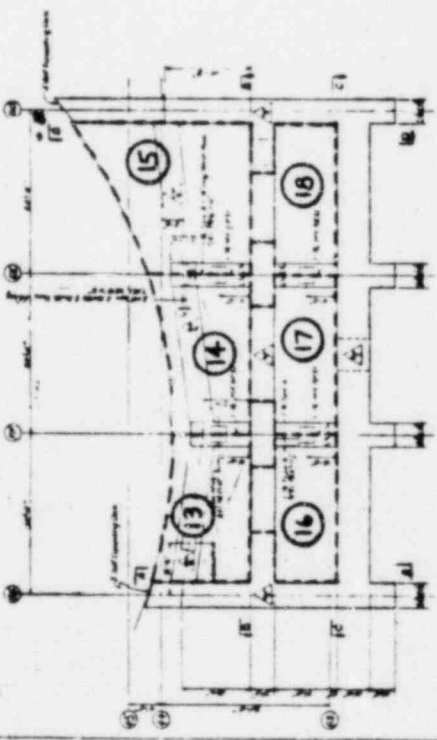
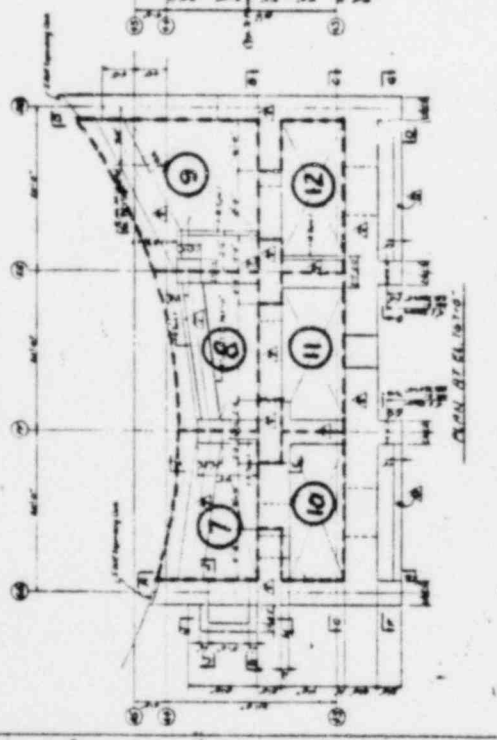
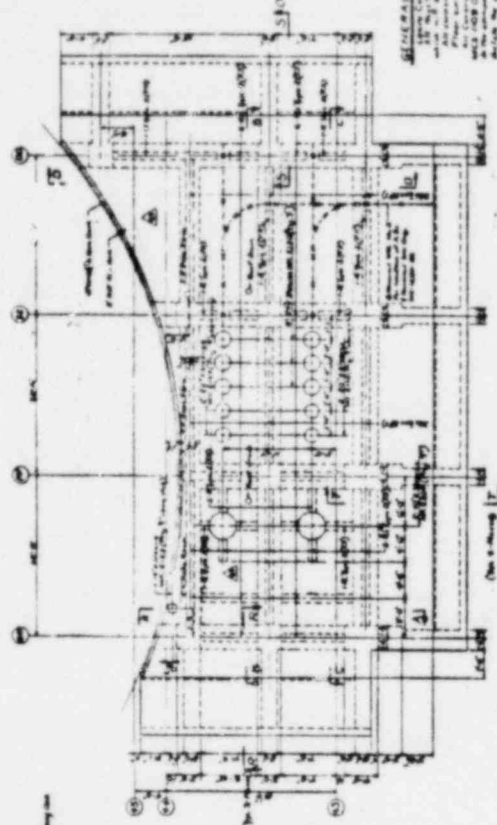
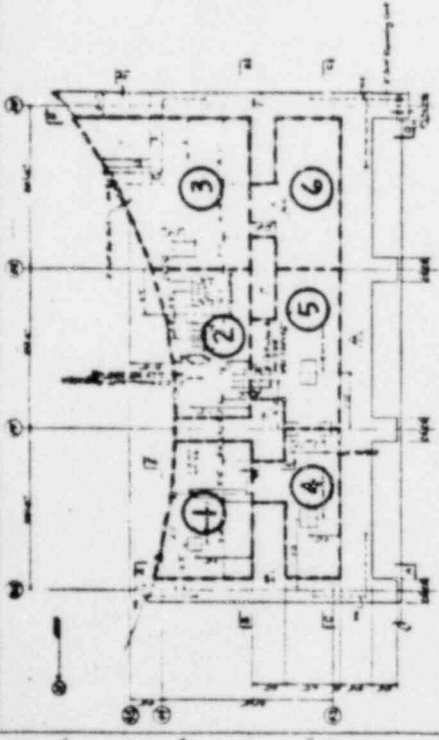
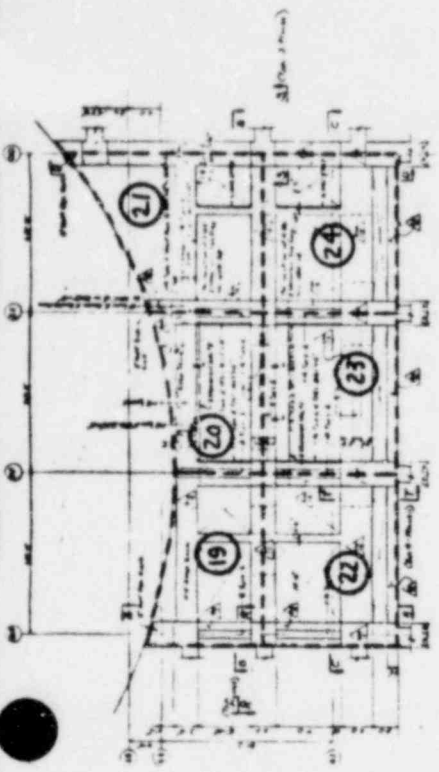
NO. 2 STEEL 1/2\"/>

SMA-WD-24
MLQUE - MAIN STEAM
RESTRAINT DIA PIPE SIZE (SEE QUOTE PAGE)
SHOULD PIPE RUPTURE RESTRAINTS

NOTE: SEE DRAWING TO THE
LEFT FOR DIMENSIONS

RESTRAINTS				PIPE POWER CAPACITY		
NO.	DIAM.	LENGTH	TYPE	STEEL	STEEL	STEEL
1				150	150	150
2				150	150	150
3				150	150	150
4				150	150	150
5				150	150	150
6				150	150	150
7				150	150	150
8				150	150	150
9				150	150	150
10				150	150	150
11				150	150	150
12				150	150	150
13				150	150	150
14				150	150	150
15				150	150	150

FIGURE 6-5
Sheet 1 of 2



GENERAL NOTES
 1. ALL WORK SHALL BE IN ACCORDANCE WITH THE LATEST EDITIONS OF THE BUILDING CODES AND SPECIFICATIONS.
 2. THE CONTRACTOR SHALL BE RESPONSIBLE FOR OBTAINING ALL NECESSARY PERMITS AND APPROVALS.
 3. ALL MATERIALS AND WORKMANSHIP SHALL BE SUBJECT TO INSPECTION AND APPROVAL BY THE ARCHITECT.
 4. THE CONTRACTOR SHALL MAINTAIN ACCESS TO ALL ADJACENT PROPERTIES AT ALL TIMES.
 5. ALL UTILITIES SHALL BE PROTECTED AND MARKED PRIOR TO CONSTRUCTION.
 6. THE CONTRACTOR SHALL BE RESPONSIBLE FOR THE PROTECTION OF ALL EXISTING UTILITIES.
 7. ALL WORK SHALL BE COMPLETED WITHIN THE SPECIFIED TIME FRAME.
 8. THE CONTRACTOR SHALL MAINTAIN A NEAT AND ORDERLY WORK SITE AT ALL TIMES.
 9. ALL MATERIALS SHALL BE STORED PROPERLY AND PROTECTED FROM THE ELEMENTS.
 10. THE CONTRACTOR SHALL BE RESPONSIBLE FOR THE DISPOSAL OF ALL WASTE MATERIALS.
 11. ALL WORK SHALL BE COMPLETED IN ACCORDANCE WITH THE DRAWINGS AND SPECIFICATIONS.
 12. THE CONTRACTOR SHALL BE RESPONSIBLE FOR THE PROTECTION OF ALL ADJACENT PROPERTIES AT ALL TIMES.

NUCLEAR SAFETY RELATED

NO.	DESCRIPTION	DATE	BY
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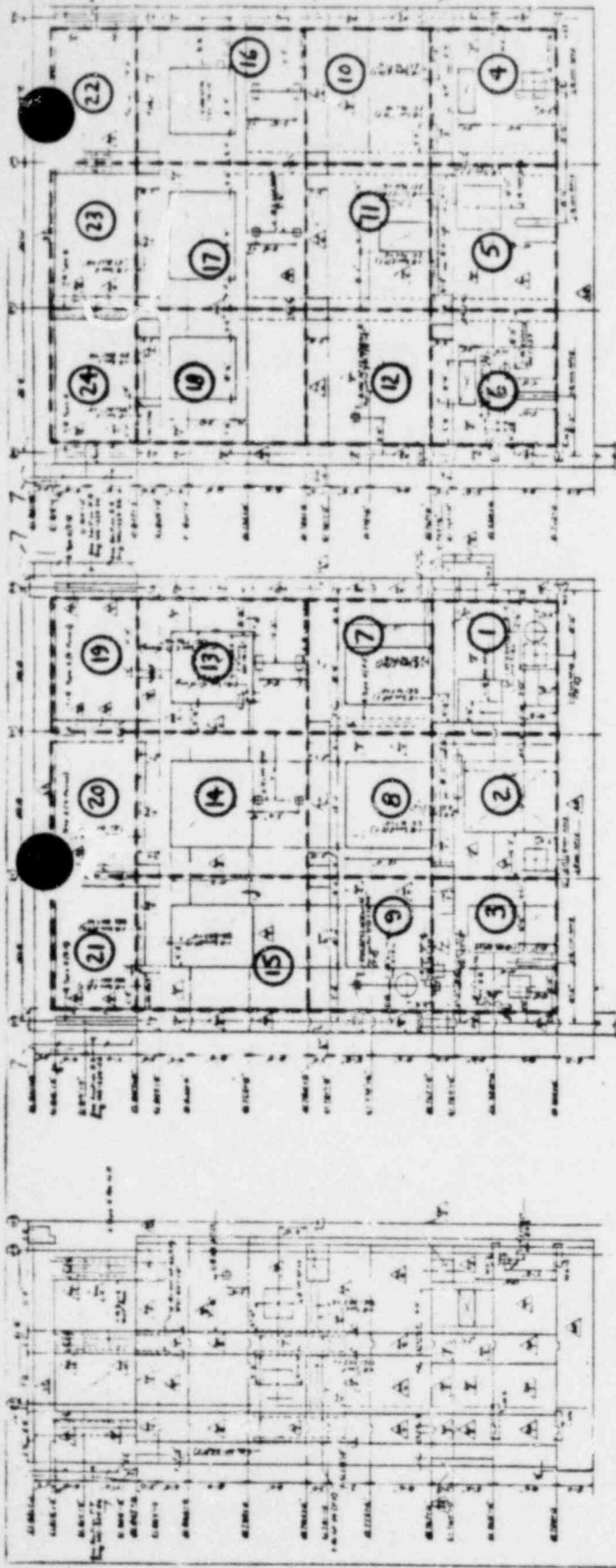
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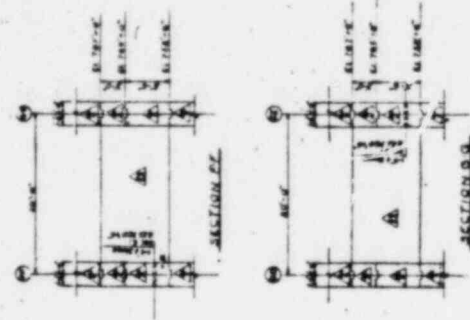
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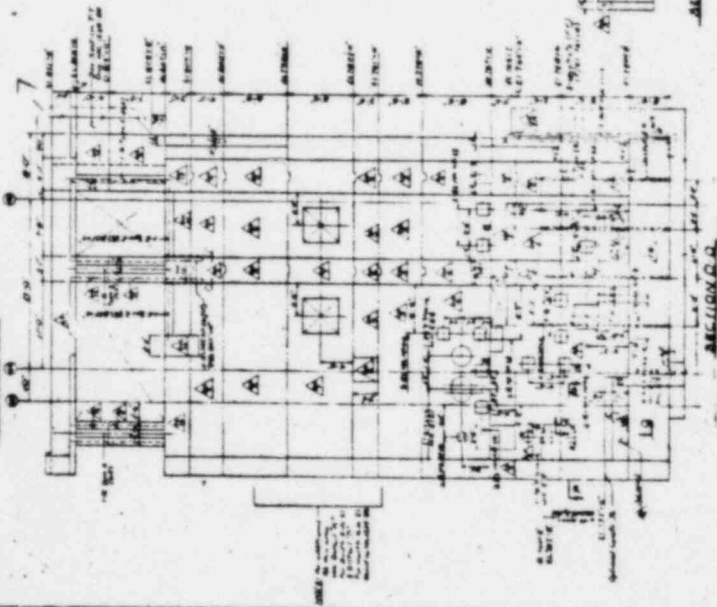
SECTION C.C.

FIGURE 6-5
Sheet 2 of 2

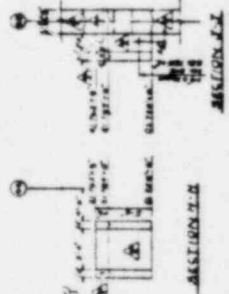
SECTION B.B.



SECTION A.A.



SECTION E.E.



SECTION D.D.

NO.	DESCRIPTION	QTY	UNIT	AMOUNT
1	CONCRETE	100	CU YD	100
2	REINFORCEMENT	50	TONS	50
3	BRICK	200	1000'S	200
4	CEMENT	50	TONS	50
5	IRON	100	TONS	100
6	STEEL	50	TONS	50
7	WOOD	100	CU YD	100
8	GLASS	100	SQ YD	100
9	PAINT	100	TONS	100
10	PLASTER	100	CU YD	100
11	ROOFING	100	SQ YD	100
12	MECHANICAL	100	TONS	100
13	ELECTRICAL	100	TONS	100
14	HEATING	100	TONS	100
15	Cooling	100	TONS	100
16	Other	100	TONS	100
17	Other	100	TONS	100
18	Other	100	TONS	100
19	Other	100	TONS	100
20	Other	100	TONS	100
21	Other	100	TONS	100
22	Other	100	TONS	100
23	Other	100	TONS	100
24	Other	100	TONS	100

NUCLEAR SAFETY RELATED
 SECTION 1. 7. 10. 11. 12. 13. 14. 15. 16. 17. 18. 19. 20. 21. 22. 23. 24.
 OTHER SHEET NO. 8

ALL DIMENSIONS ARE IN FEET AND INCHES.
 ALL WALLS ARE 8" THICK UNLESS OTHERWISE NOTED.
 ALL FLOORS ARE 4" THICK UNLESS OTHERWISE NOTED.
 ALL CEILING ARE 8" THICK UNLESS OTHERWISE NOTED.
 ALL ROOFS ARE 8" THICK UNLESS OTHERWISE NOTED.
 ALL DOORS ARE 4' 0" HIGH UNLESS OTHERWISE NOTED.
 ALL WINDOWS ARE 4' 0" HIGH UNLESS OTHERWISE NOTED.
 ALL STAIRS ARE 8" THICK UNLESS OTHERWISE NOTED.
 ALL ELEVATORS ARE 8" THICK UNLESS OTHERWISE NOTED.
 ALL MECHANICAL ARE 8" THICK UNLESS OTHERWISE NOTED.
 ALL ELECTRICAL ARE 8" THICK UNLESS OTHERWISE NOTED.
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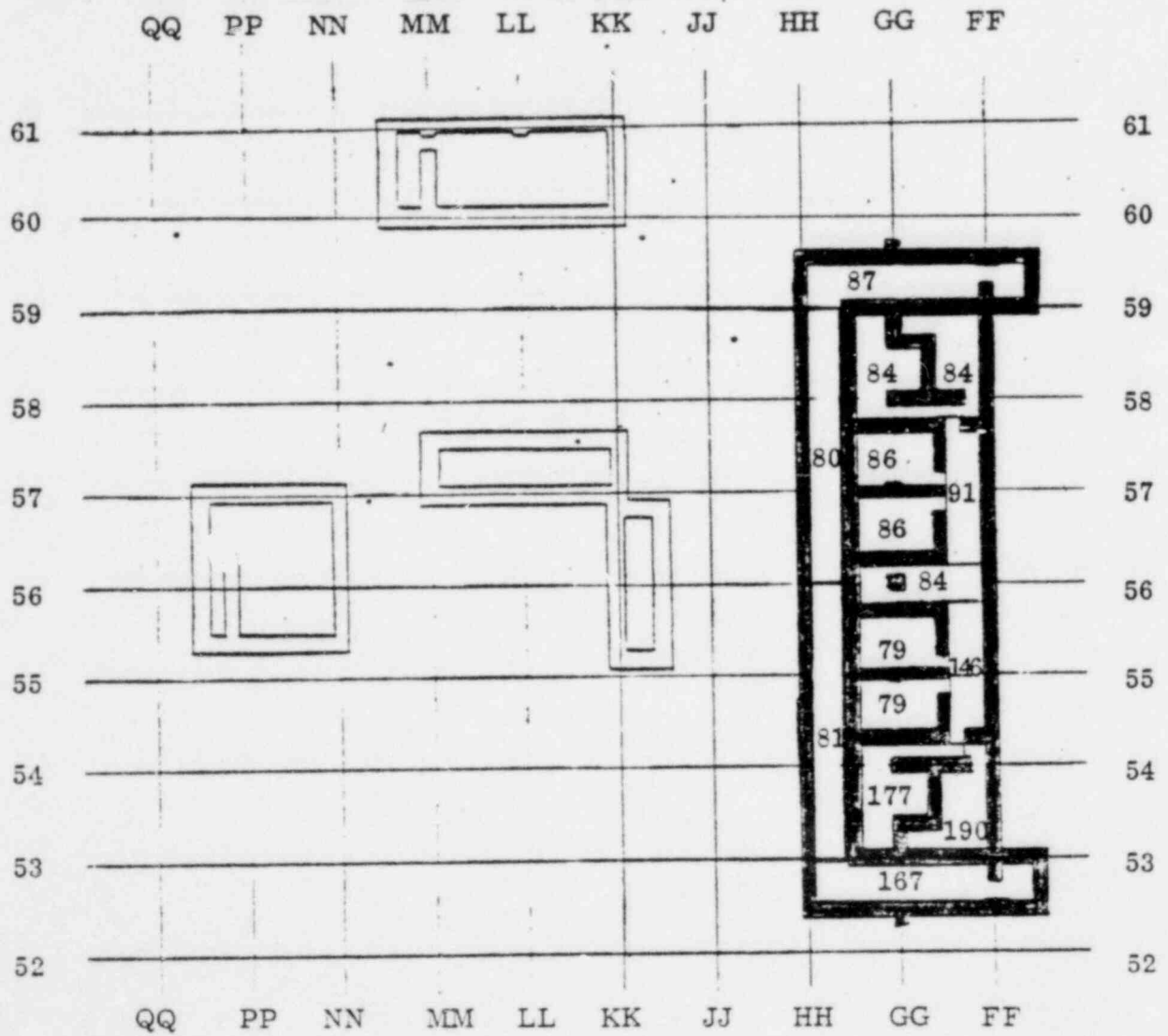
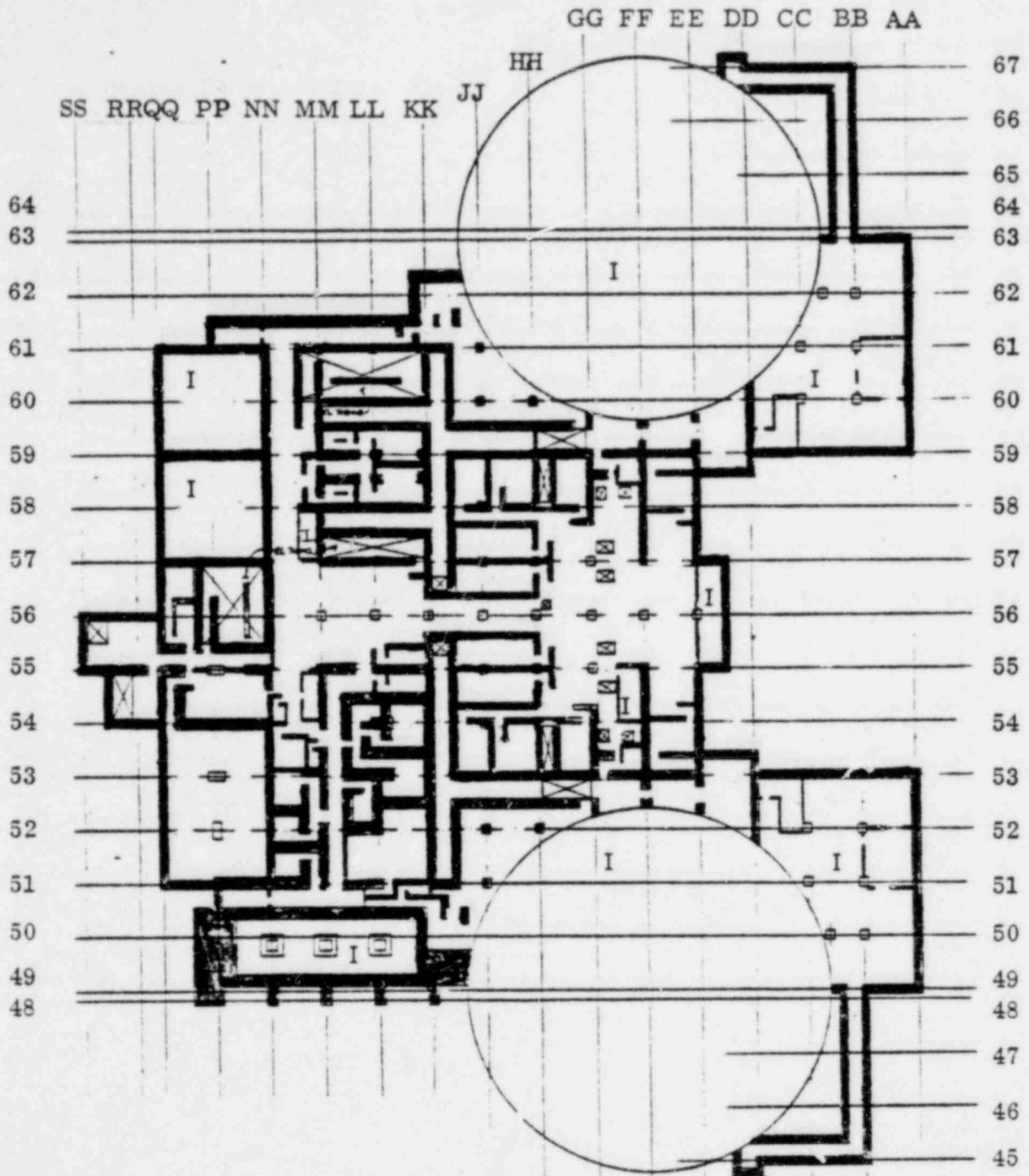


FIGURE 6-6 ELEVATION 695'-0" PEAK TEMPERATURES



Regions Denoted I are Isolated from Environmental Temperature Effects.

This Entire Floor is at 212°F Except Where Denoted I.

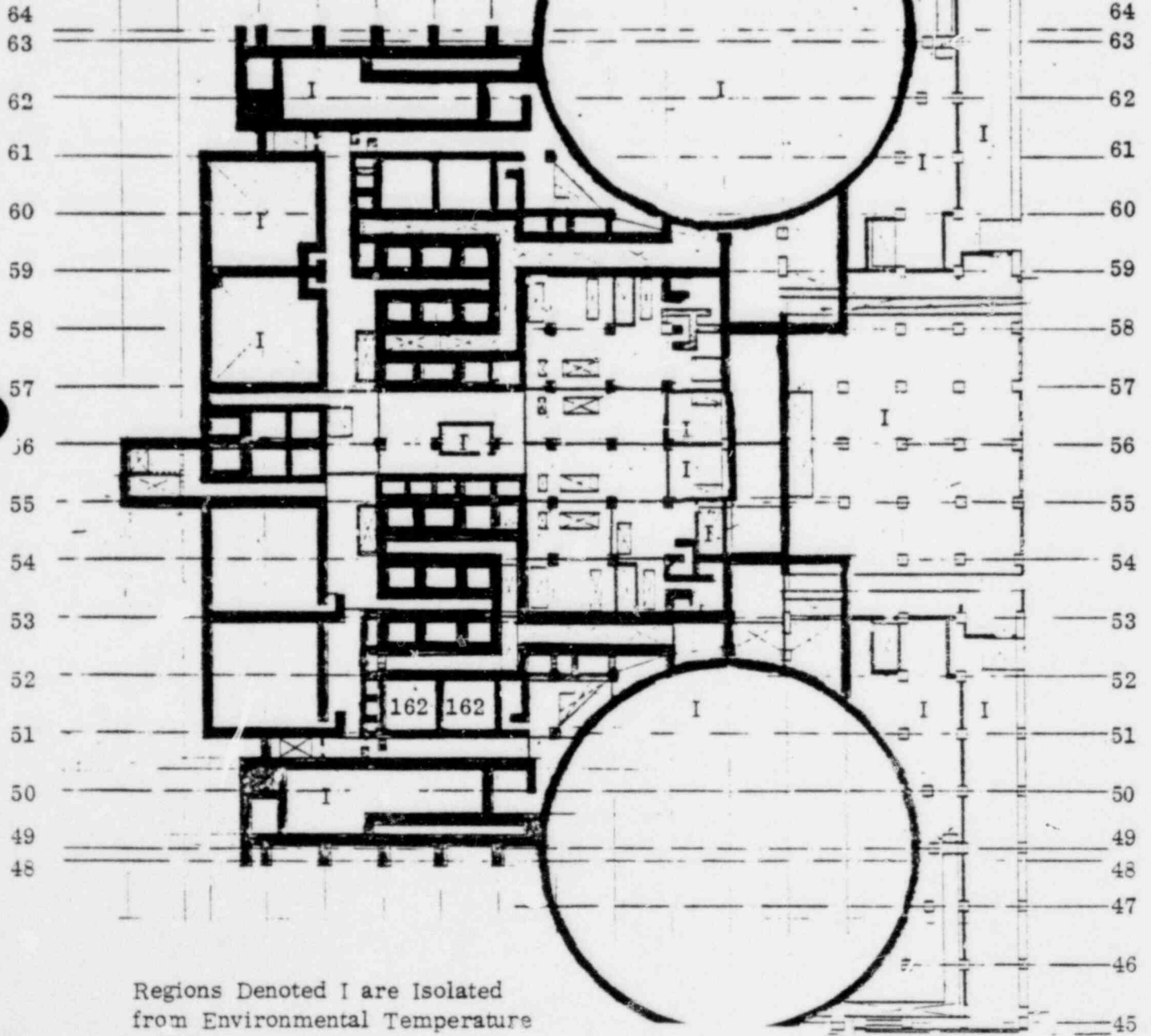
FIGURE 6-6 ELEVATION 716'-0" PEAK TEMPERATURES

Sheet 2 of 5

DD CC BB AA

JJ HH GG FF EE

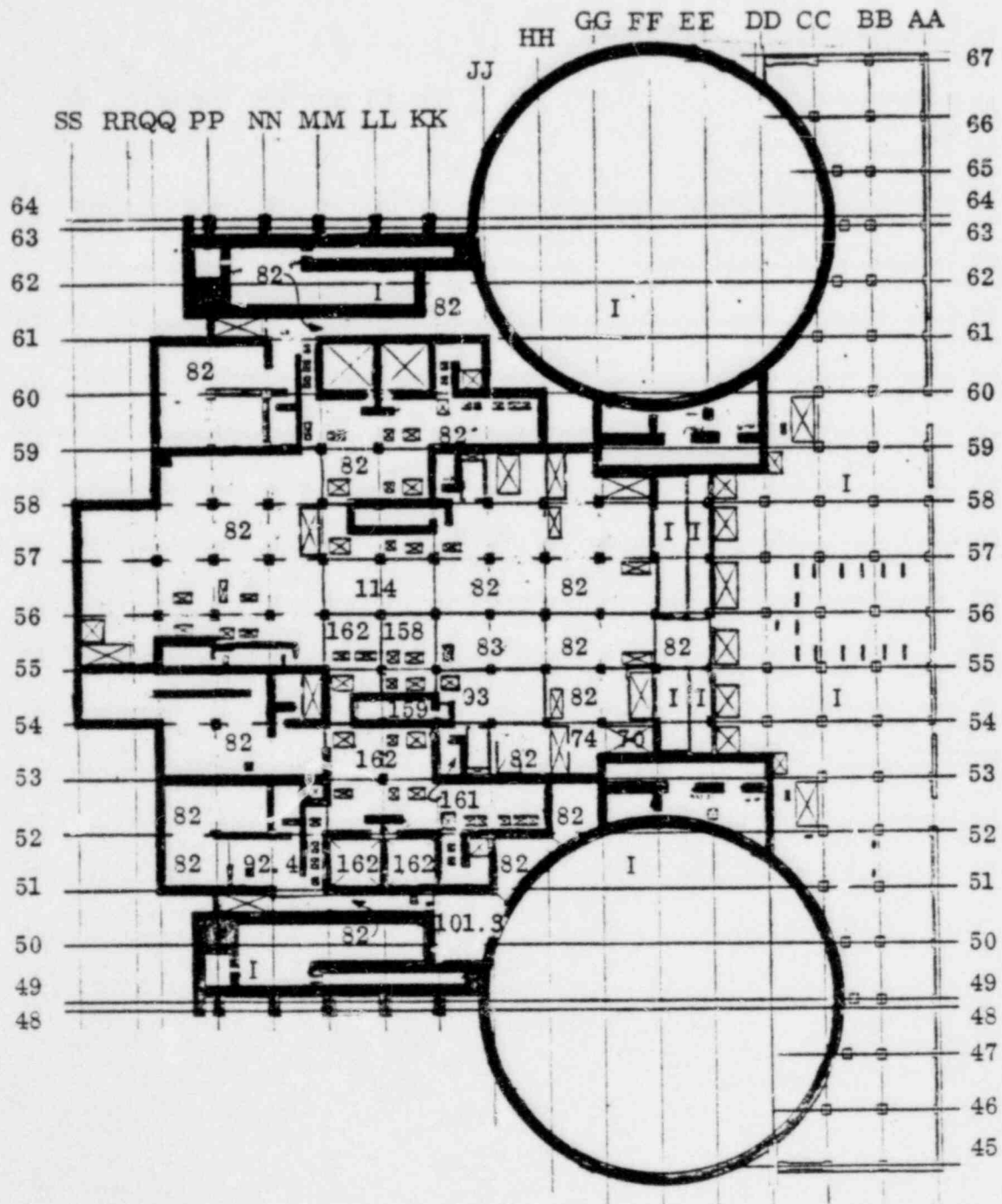
SS RRQQ PP NN MM LL KK



Regions Denoted I are Isolated from Environmental Temperature Effects

This Entire Floor is at 212°F Unless Otherwise Noted

FIGURE 6-6 ELEVATION 733'-0" PEAK TEMPERATURES

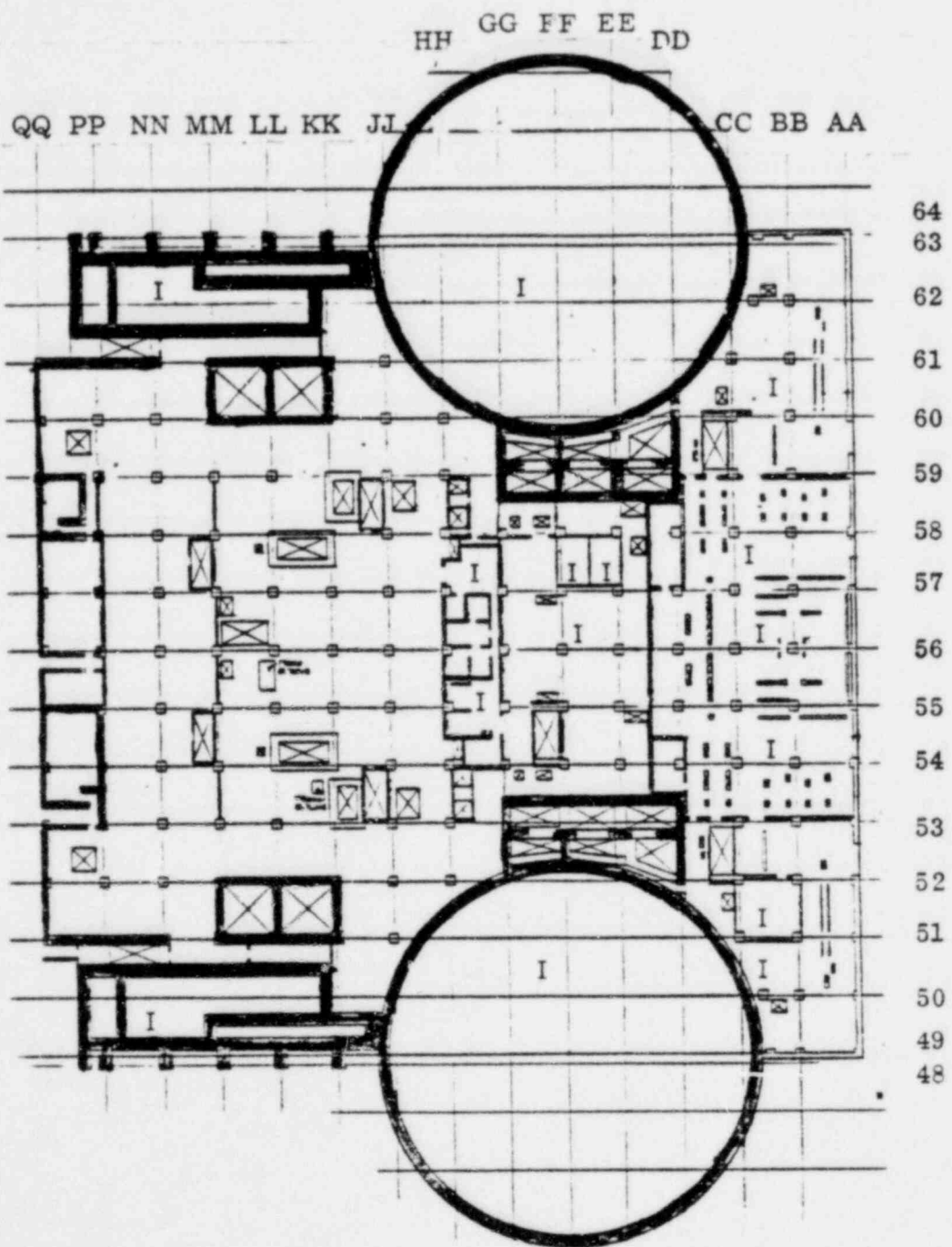


Regions Denoted I are Isolated From
Environmental Temperature Effects

FIGURE 6-6 ELEVATION 750'-0" PEAK TEMPERATURES

Sheet 4 of 5

Revision 1
New Page



Regions Denoted I are Isolated from Environmental Temperature Effects
 This Entire Floor Has Temperature 76°F Unless Where Marked I

FIGURE 6-6 ELEVATION 767'-0" PEAK TEMPERATURES
 Sheet 5 of 5

Revision 1
 New Page

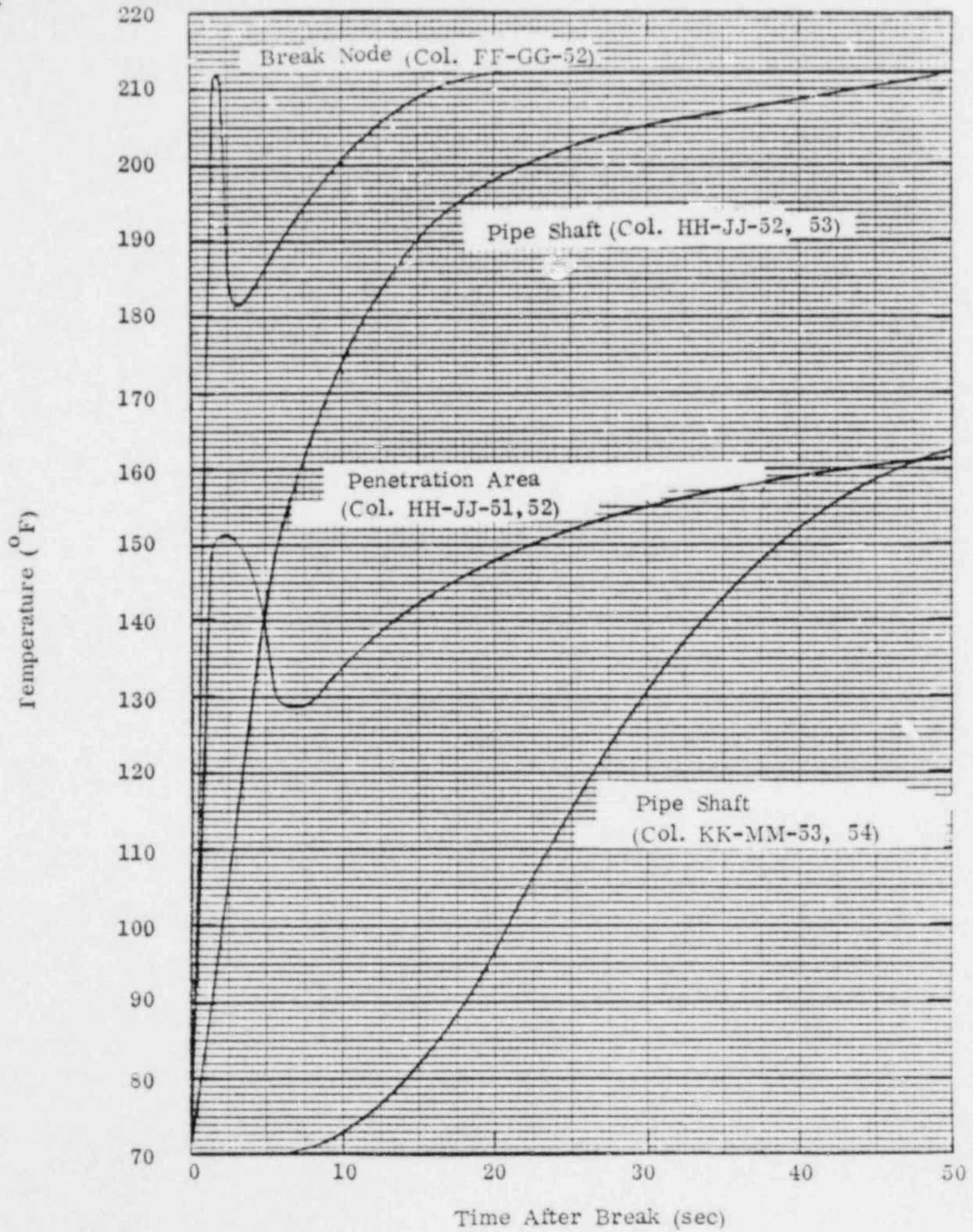


FIGURE 6-7 TYPICAL TEMPERATURE RESPONSE CURVES
FOR BB LINE BREAK

APPENDIX A

A.1 General Design Considerations

Pipes which have been identified as creating unacceptable adverse effects due to pipe movement are provided with means of controlling their motion through the use of separation, restraints, barriers, or some other acceptable method of protection.

In order to properly determine the adequacy of a system including pipe and restraints following a postulated pipe rupture, two design considerations are evaluated, the dynamic event and steady-state equilibrium.

The following references are used in pipe whip restraint design:

1. USNRC Regulatory Standard Review Plan, Section 3.6.2, March, 1975, Washington, D.C. 20545
2. ANSI-N176 Draft 4, May 1975, Section 6, Washington, D.C. 20555
3. AISC Manual of Steel Construction, Seventh Edition, 1970, New York, New York

a) Dynamic Event.

Upon the occurrence of the postulated pipe rupture, the piping system, consisting of pipe, restraint structures, etc., will respond dynamically to the applied fluid dynamic forces. The pipe will acquire kinetic energy as the forces move it to impact against the restraint system. The additional energy imparted to the system after initial contact is also considered. The total kinetic energy is dissipated and absorbed in the piping and restraint systems. Elastic rebound is not considered because all restraint designs utilize plastic material deformation in the energy absorber.

b) Steady-State Equilibrium

Following the occurrence of the dynamic event (when motion ceases and the fluid dynamic forces reach steady-state values), the system is capable of supporting the steady-state blowdown thrust force. All systems satisfy the requirements of a static equilibrium analysis.

A.2 Restraints

The primary function of a pipe whip restraint is to control pipe motion upon the occurrence of a pipe rupture. As used in this context, a restraint is considered to be different from a support. In certain instances, a pipe rupture restraint may also function as a pipe support and should be designed according to Tables 3.9.2-3 and 3.9.2-4 of the McGuire FSAR with hanger materials per the McGuire hanger specification.

Tables 3.9.2-3 and 3.9.2-4 limit stress under faulted conditions to the yield stress.

The pipe whip restraint system is separated into three primary components: a) the Energy-Absorbing Device, b) the Restraint Structure, and c) the Plant Structure.

- a. The Energy-Absorbing Device is a structural, mechanical hydraulic cushion or other material or device which is designed to minimize the forces imposed on the Restraint Structure by absorbing kinetic energy.
- b. The Restraint Structure is that structural assemblage which connects the Energy-Absorbing Device to an anchorage, generally an embedded plate in the Plant Structure.
- c. A Plant Structure is any major portion of a building which transmits loads to the building foundation.

A.2.1 Design Limits for Energy Absorbing Devices

Pipe whip restraints are designed for one-time usage, and as such are allowed to have greater distortion, plastic deformation, etc., than is normally permitted for support design. When an elastic dynamic system analysis is performed to determine the restraint loadings, the amount of permanent deformation is limited in order to preserve the validity of that analysis. For a plastic dynamic system analysis, wherein the effects of strain-rate, strain-hardening, etc., are included, the permitted strain in metallic ductile materials is limited to the following ductility factors (μ):

bending: $\mu < 12.5$ (open sections; I, WF, T, etc.)
 $\mu < 25.0$ (closed sections; pipe, box, etc.)
 $\mu < 6.0$ (members where shear governs design)

axial tension: $\mu < 50 \frac{\epsilon_u}{\epsilon_y}$, where

ϵ_u = uniform ultimate strain
 ϵ_y = yield strain

columns: ductility factors are not used for columns in compression.

The ductility factor, μ , is defined as the ratio of allowable deflection to the elastic deflection at specified yield stress:

$$\mu = \frac{\text{allowable deflection}}{\text{elastic deflection at yield}}$$

When crushable materials are used for Energy-Absorbing Devices, the energy-absorbing capacity is determined from recommended design values provided by the material manufacturer. Conservative Estimates of the energy absorbing capacity of materials supplied by Duke Power Company are substantiated with test data.

A.2.2 Design limits for Restraint Structures

Stresses in restraint structural members are limited to the yield stress at operating temperature according to Tables 3.9.2-3 and 3.9.2-4 of the McGuire FSAR. However, deflection may exceed the values predicted by elastic theory providing the ductility factors established in A.2.1 are not exceeded.

A.3 Method of Analysis

A.3.1 Simplified Method

Simplified methods have been developed to determine the dynamic and steady-state blowdown loads imposed on the Energy-Absorbing Device and on the Restraint Structure. The steps involved in this method are outlined below.

A.3.1.1 Analysis of Energy-Absorbing Device

- a) The blowdown thrust at the time of rupture is calculated by equation 6.1, using the fluid properties at normal operating conditions. The calculated blowdown thrust is assumed to develop instantaneously at rupture, and remains constant until the fluid reservoir has dissipated.
- b) The energy to be absorbed by the Energy-Absorbing Device from the whipping pipe is calculated to be the steady-state blowdown force times the distance the pipe travels, less the rotational strain energy developed in the pipe at the hinge point. The rotational strain energy may be set equal to zero for conservatism. The distance traveled by the pipe is considered to be the distance between the face-of-pipe and the face of the Energy-Absorbing Device plus the deflection of the Energy-Absorbing Device and the deflection of the restraint structure.
- c) The energy to be absorbed by the Energy-Absorbing Device in "b" above is equated to the energy absorbing capability of the device as represented by its force-deflection curve. For an Energy-Absorbing Device whose response is represented by a bi-linear force deflection curve, the resulting equation is

$$F(g + d + \Delta) = \frac{1}{2} P_E d_E + P_E (d - d_E) + \frac{1}{2} (P - P_E) (d - d_E) + \frac{P \Delta}{2}$$

where

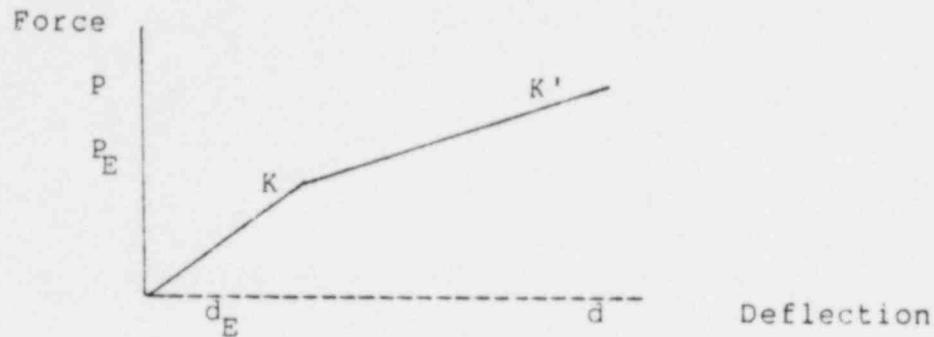
- F = blowdown thrust force
- g = gap distance
- d = deflection of the Energy-Absorbing Device required to bring the whipping pipe to rest

P_E = yield force of bi-linear force-deflection curve

d_E = deflection at yield on force-deflection curve

P = Energy-Absorbing Device maximum force

Δ = Elastic deflection of the restraint structure at load



Bi-linear force-deflection curve

In addition to the maximum deflection and force calculated above, optionally a time history of force and deflection may be obtained by computer techniques using a single-degree-of freedom model of the whipping pipe impacting an Energy-Absorbing Device described by the bi-linear force-deflection curve.

A.3.1.2 Analysis of Restraint Structural System

The maximum load realized by the structural system will be the maximum load transferred by the Energy-Absorbing Device (designated as P in Section A.3.1.1), times the Dynamic Amplification Factor.

$$F_{\text{restr}} = P \times \text{DAF}$$

where: F_{restr} = Force imposed on the Structural System

DAF = the Dynamic Amplification Factor

The dynamic amplification of the structural system is a function of the dynamic loading, the frequency of the Structure and the structural damping. If the structural damping is conservatively assumed equal to zero, and the dynamic loading assumed to be an instantaneously applied constant load, the Dynamic Amplification Factor for a single degree-of-freedom system can be written as follows:

$$\text{DAF} = 1 - \cos(\omega t)$$

where: w = The frequency of the simple degree-of-freedom system representing the structural system
 t = Length of time after initial loading

The maximum Dynamic Amplification Factor for the assumed conditions is equal to 2.0.

If the maximum Dynamic Amplification factor is applied to the restraint structure, the resulting loads transmitted to the plant structure can be treated as equivalent static forces.

A.3.2 Detailed Method

The larger and more complex piping and restraint systems sometimes require more detailed analysis than can be provided by the energy balance method. When these circumstances are encountered, one of the analytical methods outlined below is used:

a) Complete System Dynamic Time History Analysis

The piping and pipe whip restraint system is modeled in sufficient detail to reflect its dynamic characteristics. The lumped-mass approach is employed. Inertia and stiffness effects of the system, and gaps between piping and the restraints, are included. The pipe rupture time history forcing function(s) is applied and time history responses of restraint loads, pipe deformations, etc., are computed by numerical integration. This approach requires the use of digital computer programs.

b) Simplified Dynamic Time History Analysis

A dynamic time history analysis of a portion of a piping system is performed instead of an analysis of the complete system. Consideration is given to dynamic coupling and forcing functions acting on the complete system in the representation of the flexibility and inertial effects of the remainder of the system.

c) Quasi-Dynamic Time History Analysis

Dynamic time history analysis is performed on representative systems to develop design data in the form of curves, graphs, etc., to be used for restraint location and design. The study should be parametric in nature, considering dynamic coupling, system forcing functions, the effects of gaps, support flexibilities, the pipe size, and schedule. This approach allows interpolation between the calculated data, and therefore avoids continued reanalysis of standard designs due to small changes in a particular parameter.

A.4 Piping Design Limits

Whipping pipes are considered to be controlled when the motion of the pipe is halted, or when the pipe is prevented from impacting essential components. In order to verify that pipe motion will be controlled, the energy absorbed in the pipe and restraint system during the dynamic event is shown to be less than their energy-absorbing capacity.

When the energy balance method is used, the energy-absorbing capacity of the pipe is considered to be the rotational strain energy of the pipe at the hinge point; when more detailed methods are used, the energy-absorbing capacity of the pipe is determined from the total force-displacement characteristics of the pipe as determined by analysis.

In order to assure stability of the pipe after the dynamic event, the load-carrying capacity of the pipe is shown to be greater than the loads imposed by the steady-state blowdown thrust.

A.4.1 Material Properties

Because higher yield strengths are developed during dynamic loading, a 10% increase in the minimum specified yield strength of the process pipe, S_y , is used in the dynamic analysis. Material properties of the pipe are based on the data provided in Section III of the ASME Boiler & Pressure Vessel Code.

APPENDIX B BLOWDOWN FORCES AND METHODS OF CALCULATION

All high-energy pipe breaks are assumed to occur within one milli-second and result in either a circumferential or longitudinal rupture as defined in Section 3. Circumferential and longitudinal breaks are not assumed to occur simultaneously.

B.1 Frictional Losses

In calculating forces acting on the piping system, credit is taken for any restrictions or line losses between the break and the pressure reservoir(s). Typical restrictions to flow which are considered are: orifices, nozzles, valves, reduced cross sections, spargers, friction, and elbows.

B.2 Closed-End Lines

Depending upon the proximity of the closed-end of the line (dead end or normally closed valves) to the postulated break, if it is shown that the energy contained within the line is insufficient to cause pipe whip, the pipe whip response is not considered. However, the effects of jet impingement on non-whipping closed-end or limited flow lines are evaluated.

B.3 Calculation of Fluid Reactions

The simplified method used to calculate blowdown forces utilizes the following formula:

$$T_{SS} = C_T A P_o$$

where

$$C_T = \text{Steady-state thrust coefficient.}$$

$$P_o = \text{Initial total pressure in the pipe.}$$

$$A = \text{Break area of the pipe.}$$

$$T_{SS} = \text{Steady-state blowdown force.}$$

Steady-state forces are assumed to be reached instantaneously at the time of rupture and to remain constant with time thereafter. The steady-state forces

are based upon the fluid properties at normal operating conditions. Figure B.1 shows typical pipe reaction rupture thrust force transients for friction and frictionless flow from a constant-pressure.

The steady-state thrust coefficient is dependent on the fluid properties and frictional characteristics of the pipe. In the following subsections, the steady-state thrust coefficient (C_T) is established as a function of the fluid state for incorporation into Eq. B-1.

B.3.1 Saturated Gas-Liquid Mixtures

Predictions of steady-state saturated gas-liquid, or two-phase mixture, flows have been made by several investigators, notably Henry and Fauske, and Moody. Moody's work generally overpredicts flow (and therefore thrust coefficient) for all but low-quality cases and non-equilibrium flow. The steady-state frictionless thrust coefficient as predicted by Moody is shown in Figure B-2. Henry and Fauske's work has been used for calculation of low-quality flowrates and subcooled blowdown; this is discussed in the following subsection.

B.3.2 Subcooled Gas-Liquid Mixture

The steady-state frictionless thrust coefficient for subcooled liquid will be between the values for gas, $C_T = 1.25$, and for liquid, $C_T = 2.0$, as shown in Figure B-2. Flashing occurs very close to the exit plane, if not just beyond, depending on frictional losses in the pipe. Effects of friction on subcooled blowdown are shown in Figure B-3.

B.3.3 Subcooled Water

Figure B-4 gives the subcooled water thrust coefficient as a function

of total (stagnation) enthalpy, h_o , for total (stagnation) pressures, P_o , ranging from 800 to 2400 psi. These results were calculated using the Henry-Fauske technique developed for application to subcooled water, saturated water, and low-quality steam-water mixture blowdown through nozzles. The model is based on homogeneous non-equilibrium flow process. This method is not applicable to the case of long pipes with friction, and assumes there is negligible flashing at the exit.

Figures B-5 and B-6 give the thrust coefficient for the case of subcooled water blowdown through long pipes with friction. Smooth entrance geometry is assumed. Flashing may occur in the pipe or at the exit plane. This model is also based on a homogeneous non-equilibrium flow process. This method gives the maximum (critical) mass flowrate, exit pressure, and exit quality in terms of stagnation pressure and enthalpy of initially subcooled or saturated water by calculating the pressure losses due to acceleration at the pipe entrance, flashing in the pipe and pipe friction.

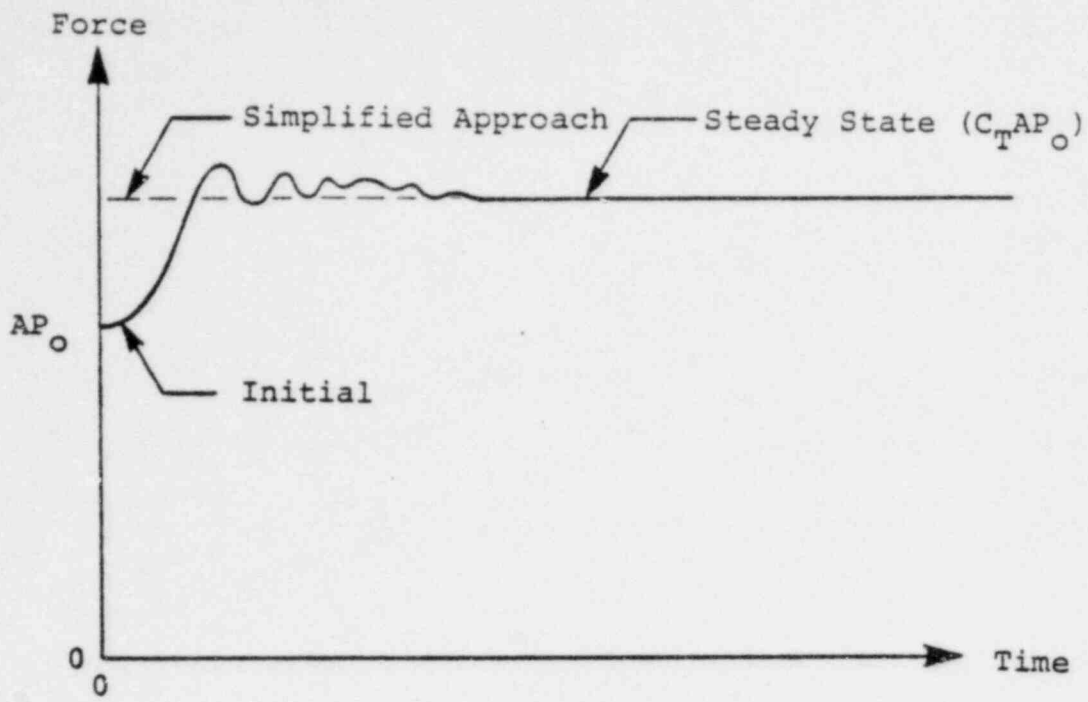
B.3.4 Incompressible Liquid

Blowdown of cold water is treated as flow of an incompressible liquid.

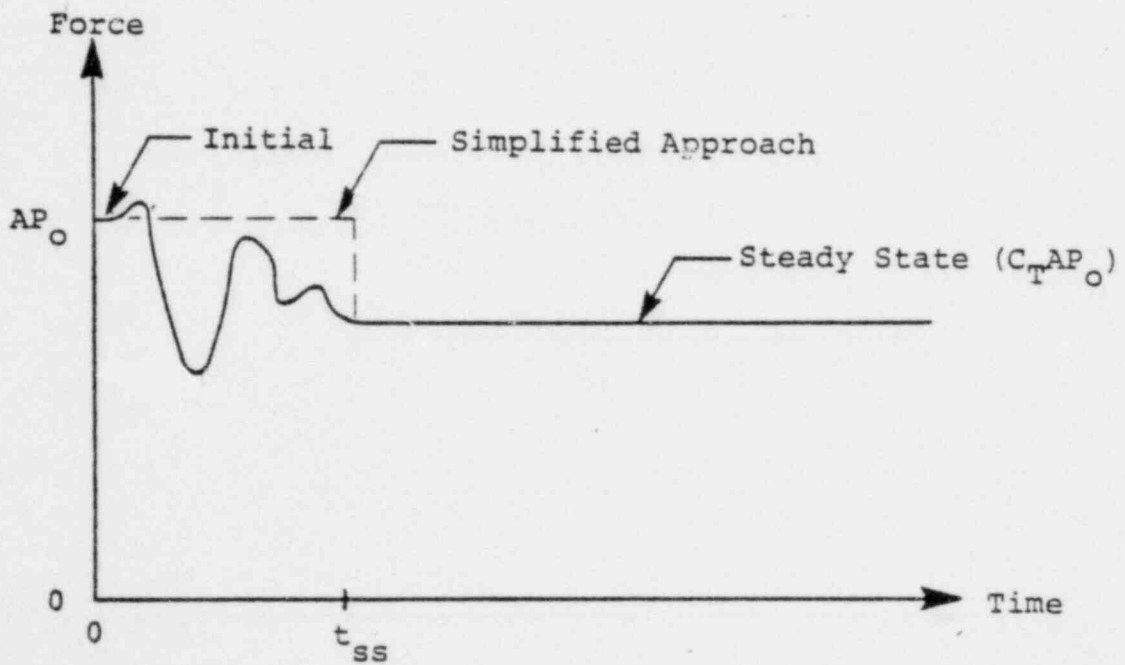
The expression for the thrust coefficient is:

$$C_T = \frac{2}{1 + fL/D} \quad (B-2)$$

This is seen as the lower curve in Figure B-3.



(a) Very Low Friction ($C_T \geq 1.0$)



(b) Friction ($C_T < 1.0$)

FIGURE B-1 PIPE RUPTURE REACTION THRUST FORCE TRANSIENT

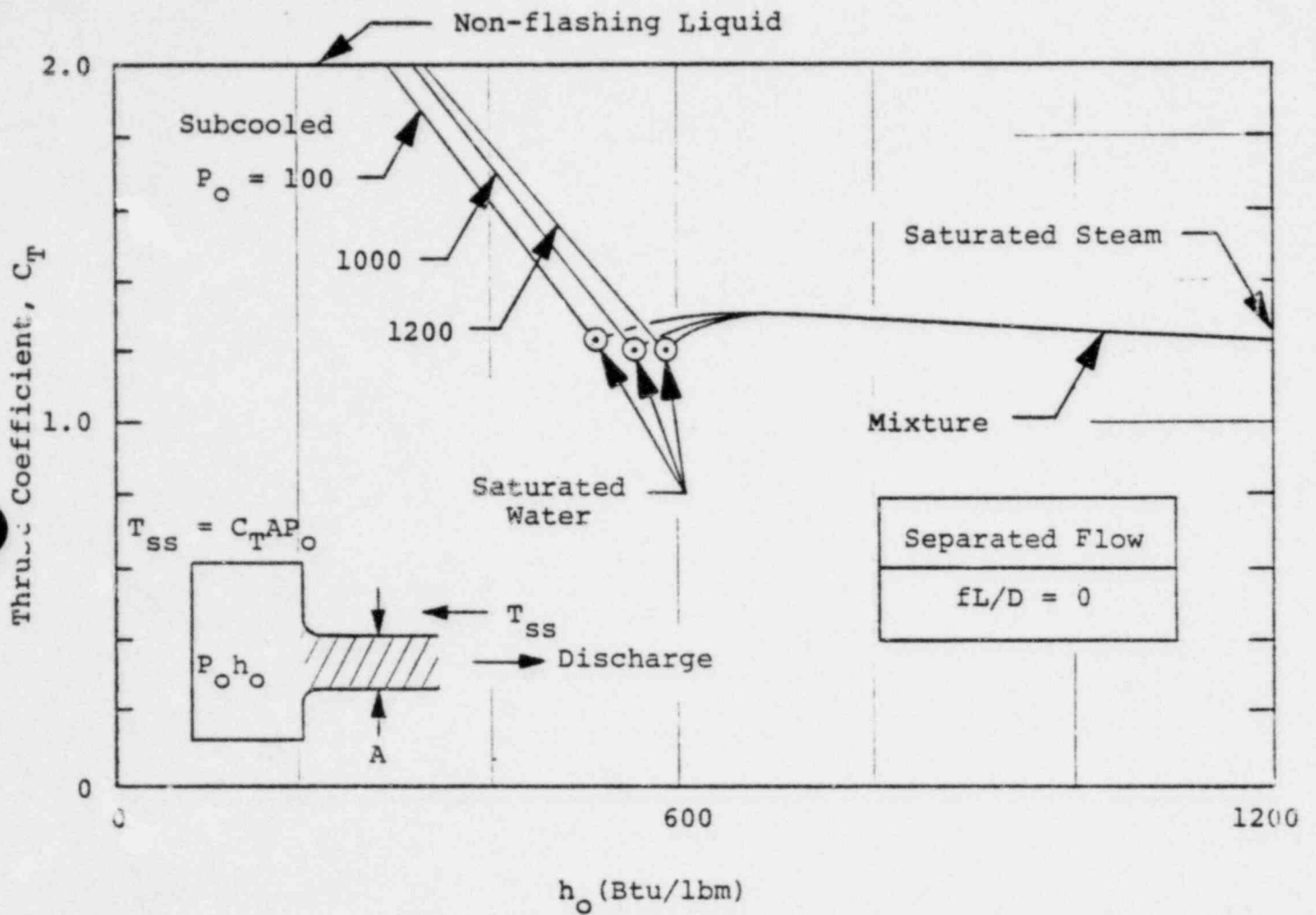


FIGURE B-2

MOODY'S STEADY-STATE SEPARATED FLOW THRUST COEFFICIENT
WITHOUT FRICTIONAL EFFECTS

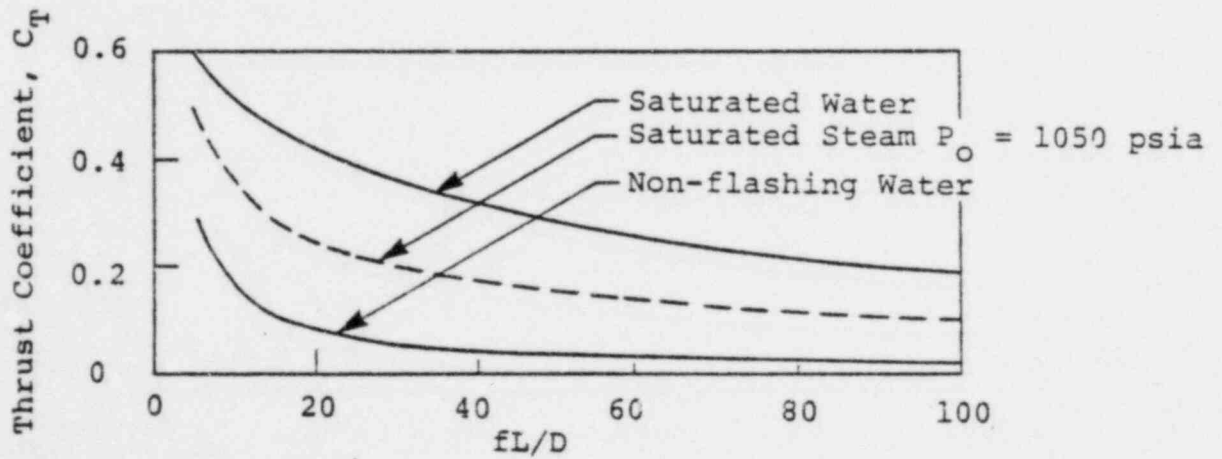
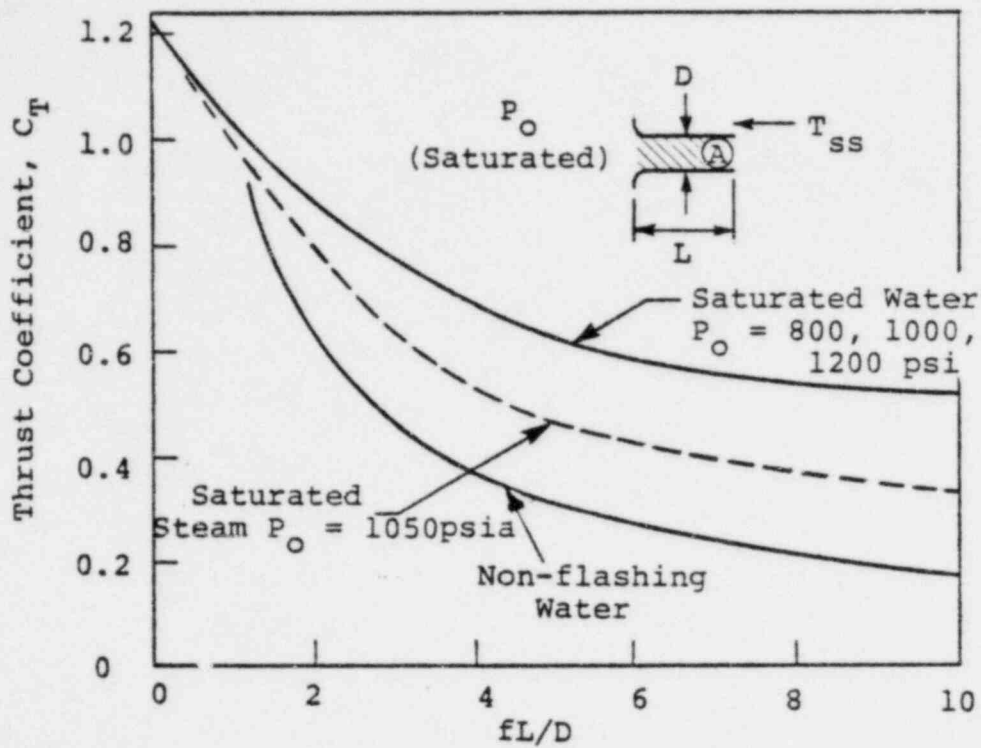


FIGURE B-3 FRICTION EFFECT ON STEADY BLOWDOWN FORCE

FIGURE B-4 THRUST FORCE AS A FUNCTION OF STAGNATION ENTHALPY, h_o , and PRESSURE, P_o
 (Henry-Fauske Model)

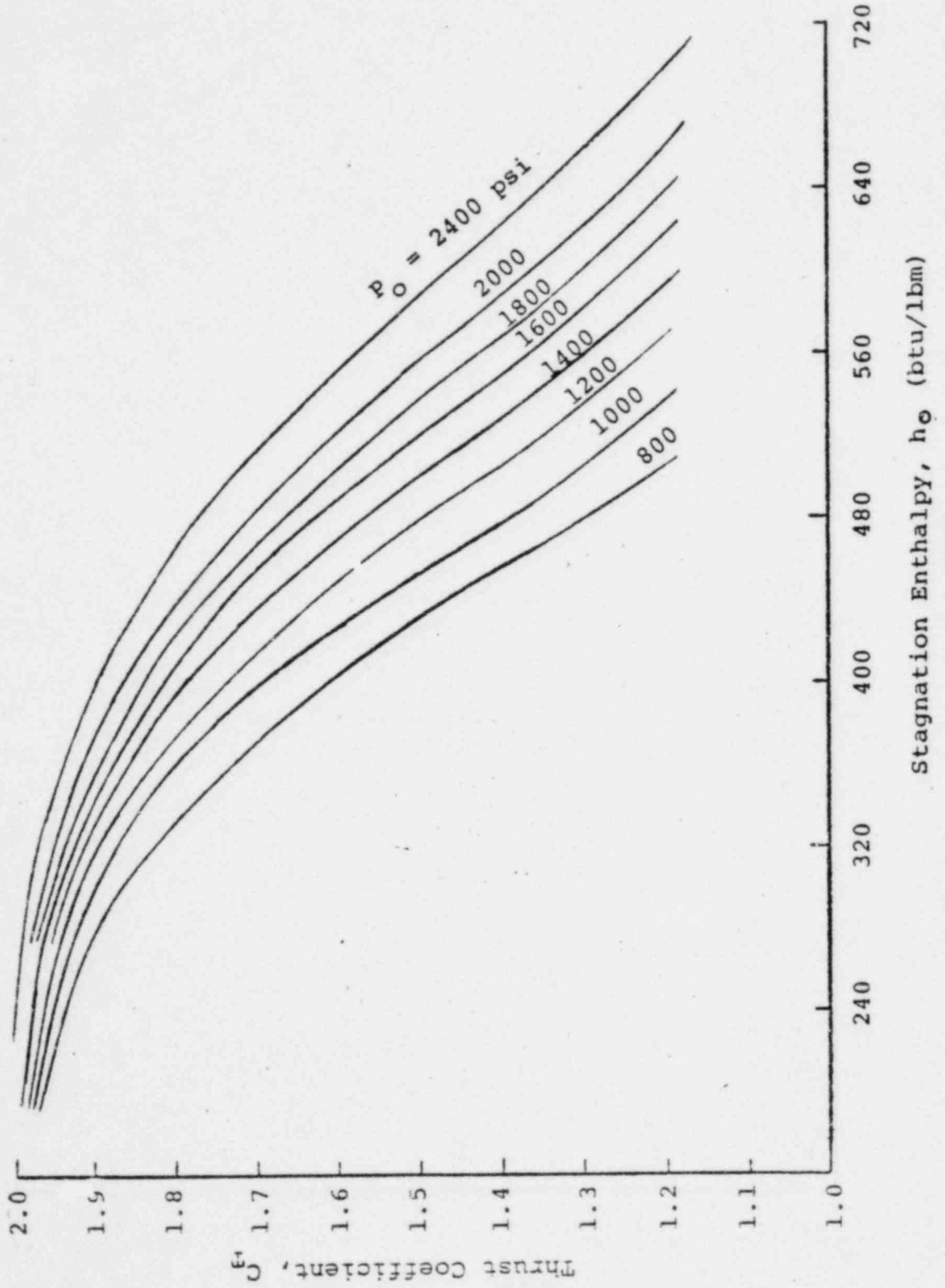


FIGURE B-5 SUBCOOLED WATER BLOWDOWN THRUST FORCE AS A FUNCTION OF STAGNATION ENTHALPHY FOR VARIOUS VALUES OF PIPE FRICTION PARAMETER, fL/D

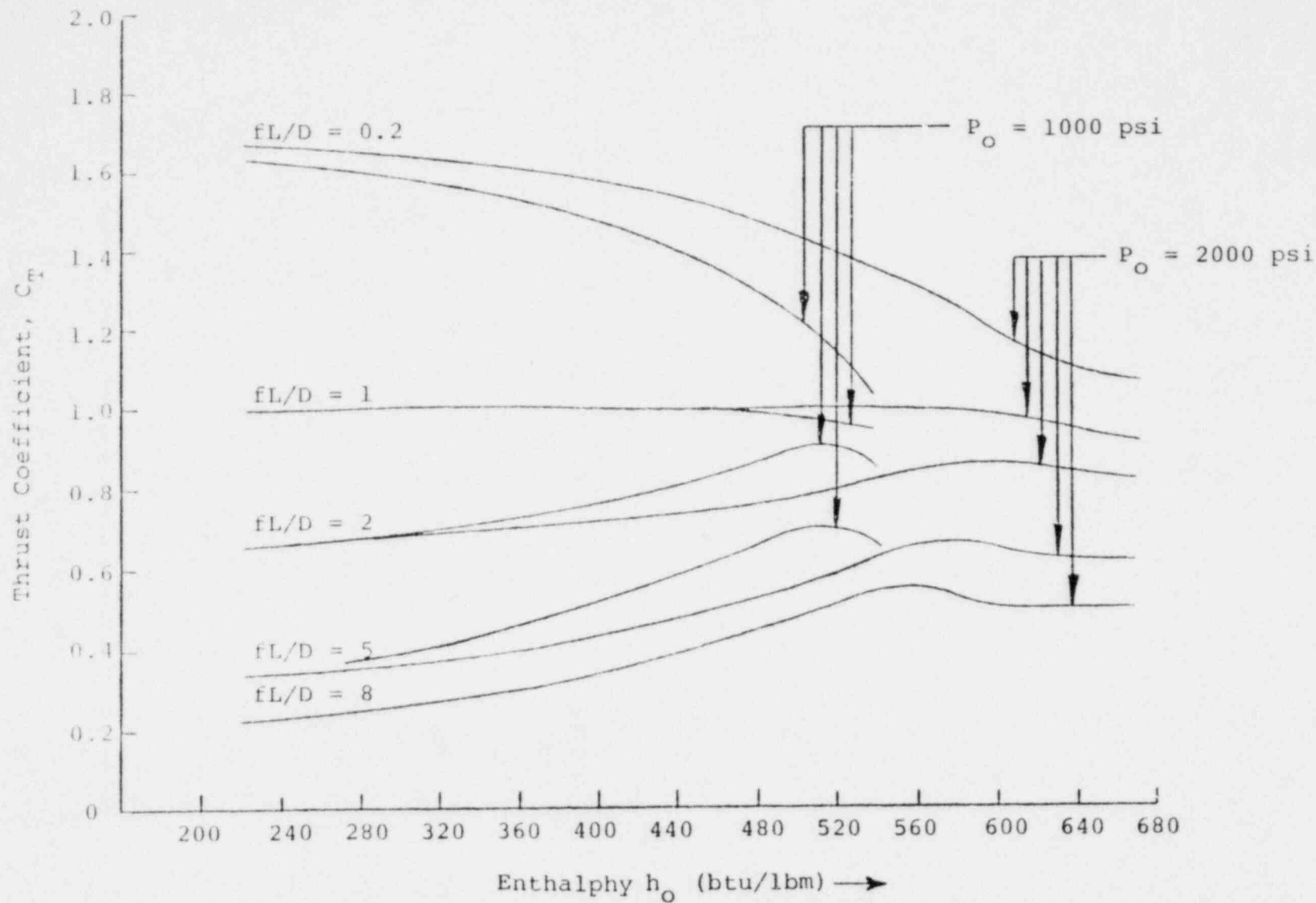
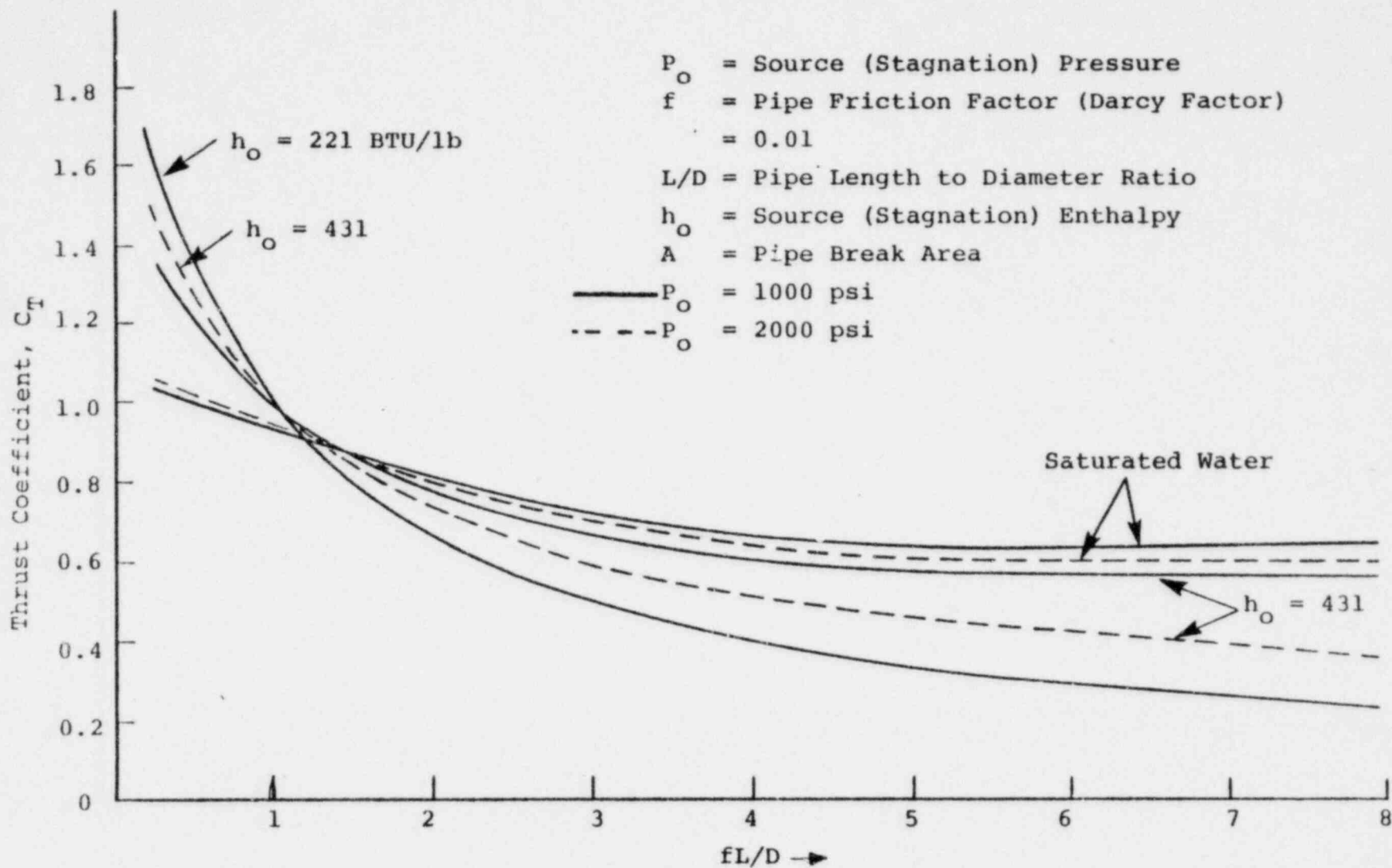


FIGURE B-6

WATER BLOWDOWN THRUST FORCE AS A FUNCTION OF PIPE FRICTION
PARAMETER, fL/D , FOR VARIOUS STAGNATION ENTHALPIES



APPENDIX C

Break Postulation on the Main Steam, Feedwater and Auxiliary Feedwater Systems Inside Containment

C.1 INTRODUCTION

This appendix presents the results of the evaluation of effects of postulated pipe failures inside containment for the Main Steam, Feedwater and Auxiliary Feedwater Systems of McGuire Nuclear Station. The appendix describes criteria and procedures used in the analysis as developed from the Nuclear Regulatory Commission Branch Technical positions APCSB 3-1 and MEB 3-1. The intent of this appendix is also to comply with Regulatory Guide 1.46. All the terminal ends and intermediate breaks in high energy piping have been analyzed. Some break locations are subject to change if as-built stress conditions are different. Postulated rupture locations and types were determined in accordance with the criteria of Section C.3. Systems and equipment essential to achieve safe plant shutdown and mitigate effects of piping failures were determined in accordance with Section C.4. Sections C.5 and C.6 provide the methods used to evaluate possible damage and the results of the evaluation. All required mitigative devices are identified in Section C.6.

C.2 PIPING LAYOUT

Main Steam and Feedwater piping represent the greatest damage potential from pipe ruptures inside containment due to their large size and high operating pressure. The Main Steam lines are enclosed within guard pipe from the steam generator nozzles to the containment penetrations. While the possibility for structural damage resulting from main steam ruptures in the containment does exist, potential consequences are minimized by this layout. Careful attention has been given to location of essential components and routing of piping systems to minimize the effect of piping failure.

C.3 CRITERIA FOR POSTULATING RUPTURES

Refer to Section 3.0 of the main body of this report for criteria for the feedwater and auxiliary feedwater systems. Breaks on the main steam system are chosen based on consequence as described in FSAR Table 3.6.5-1.

C.4 SYSTEMS AND EQUIPMENT EVALUATION

Refer to Section 4.0 of the main body of this report.

C.5 DAMAGE EVALUATION

A damage evaluation has been performed for each of the breaks postulated by application of the criteria in C.3. Main Steam, Feedwater and Auxiliary Feedwater lines have been provided with pipe rupture restraints for all postulated break locations having unacceptable targets in the containment. This ensures that essential systems, structures and components are not damaged. Pipe rupture restraints are designed using methods of Section 5.0 of the main body of the report and Appendix A.

C.6 SUMMARY AND CONCLUSIONS

In the following sections, the analysis of postulated high energy piping failure is summarized. These sections include the effects of the breaks and the plant modifications required to mitigate the effects.

C.6.1 Pipe Whip and Jet Impingement Effects

In accordance with the criteria and methods of Sections C.4 and C.5, Tables C-1 through C-3 have been prepared to identify each high energy break considered in this analysis which results in unacceptable consequences. Pipe whip restraints, jet impingement barriers, guard pipes or other mitigative devices provided are listed in Tables C-4 through C-6. The criteria used for design of these devices is presented in Appendix A.

C.6.2 Main Steam System

The main steam system has the greatest potential for damage from postulated pipe breaks. The pipe size, the operating temperature and pressure, and the large amount of stored energy dictates the most thorough review with respect to pipe rupture effects. The Main Steam system has been provided with guard pipe to limit pipe movement and restrict blowdown flow rates in the event of a process pipe failure inside the guard pipe. Main steam breaks are chosen based on consequence as described in FSAR Table 3.6.5-1. Pipe rupture restraints are provided for each postulated break to assure that essential structures and components are not damaged. Only circumferential breaks are postulated since the stress at all locations is less than 0.8 (1.2Sh + SA). Figure C-6-1 shows postulated breaks on the main steam system isometric drawing.

C.6.3 Feedwater System

Feedwater break locations are postulated at the steam generator nozzles as indicated in Figure C-6-2. No intermediate breaks are postulated inside the containment since the stresses are less than 0.4 (1.2Sh + SA) at all locations. It should be noted that main steam and feedwater penetrations are free floating and do not constitute an anchor. Pipe rupture restraints are provided for postulated feedwater breaks to ensure that essential components and structures are not damaged. Pipe break locations are listed in Table C-2 and results of the damage evaluation are given in Table C-5.

C.6.4 Auxiliary Feedwater System

Break locations for the Auxiliary Feedwater System are postulated as shown on Figure C-6-3.1 and C-6-3.2. The Auxiliary Feedwater penetrations are the floating type and do not constitute anchors. Stresses in the auxiliary feedwater lines fall between 0.4 (1.2Sh + SA) and 0.8 (1.2Sh + SA). The distribution is such that intermediate break locations for only one loop are postulated inside containment. Pipe rupture restraints are provided for all the breaks causing unacceptable interactions. Pipe break locations are listed in Table C-3 and results of the damage evaluation are given in Table C-6.

TABLE C-1

High Energy Pipe Break Locations
For Main Steam Line Inside Containment

Break Number	System	Description	Figure No.	Location	Stress Ratio To Allowable(1)
SM1-01R	Main Steam	Terminal End at Steam Generator 1A Nozzle	C-1	RB-E1.809	Terminal End Break
SM1-02R	Main Steam	At top of Main Steam Riser at outlet of second elbow downstream of Steam Generator 1A	C-1	RB-E1.813	Intermediate Break
SM1-03R	Main Steam	At outlet of third elbow downstream of Steam Generator 1A	C-1	RB-E1.760	Intermediate Break
SM1-04R	Main Steam	Terminal End at Steam Generator 1B Nozzle	C-1	RB-E1.809	Terminal Break
SM1-05R	Main Steam	At top of Main Steam Riser at outlet of second elbow downstream of Steam Generator 1B	C-1	RB-E1.813	Intermediate Break
SM1-06R	Main Steam	At outlet of third elbow downstream of Steam Generator 1B	C-1	RB-E1.760	Intermediate Break
SM1-07R	Main Steam	Terminal End at Steam Generator 1C Nozzle	C-1	RB-E1.809	Terminal Break
SM1-08R	Main Steam	At top of Main Steam Riser at outlet of second elbow downstream of Steam Generator 1C	C-1	RB-E1.813	Intermediate Break
SM1-09R	Main Steam	At outlet of third elbow downstream of Steam Generator 1C	C-1	RB-E1.760	Intermediate Break
SM1-10R	Main Steam	Terminal End at Steam Generator 1D Nozzle	C-1	RB-E1.809	Terminal Break
SM1-11R	Main Steam	At top of Main Steam Riser at outlet of second elbow downstream of Steam Generator 1D	C-1	RB-E1.813	Intermediate Break
SM1-12R	Main Steam	At outlet of third elbow downstream of Steam Generator 1D	C-1	RB-E1.760	Intermediate Break

TABLE C-2

High Energy Pipe Break Locations
For Feedwater Lines Inside Containment

Break Number	System	Description	Figure No.	Location	Stress Ratio To Allowable(1)
CF1-01R	Feedwater	Terminal End at Steam Generator 1A nozzle	C-2	RB	Terminal End Break
CF1-02R	Feedwater	Terminal End at Steam Generator 1B Nozzle	C-2	RB	Terminal End Break
CF1-03R	Feedwater	Terminal End at Steam Generator 1C Nozzle	C-2	RB	Terminal End Break
CF1-04R	Feedwater	Terminal End at Steam Generator 1D Nozzle	C-2	RB	Terminal End Break

TABLE C-3

High Energy Pipe Break Locations
For Auxiliary Feedwater Lines Inside Containment

Break Number	System	Description	Figure No.	Location	Stress Ratio To Allowable(1)
CA1-01R	Aux. Feedwater	Terminal End at Steam Generator 1A Nozzle	C-6-3.1	RB	Terminal End Break
CA1-02R	Aux. Feedwater	Terminal End at Steam Generator 1B Nozzle	C-6-3.1	RB	Terminal End Break
CA1-03R	Aux. Feedwater	Terminal End at Steam Generator 1C Nozzle	C-6-3.2	RB	Terminal End Break
CA1-04R	Aux. Feedwater	Terminal End at Steam Generator 1D Nozzle	C-6-3.2	RB	Terminal End Break
CA1-05R	Aux. Feedwater	Intermediate Break at the inlet of elbow C08	C-6-3.1	RB	Intermediate Break
CA1-06R	Aux. Feedwater	Intermediate Break at the outlet of elbow C08	C-6-3.1	RB	Intermediate Break

TABLE C-4

Results of Damage Evaluation

Unacceptable Pipe Whip/Jet Impingement Effects
For Main Steam Line Inside the Containment

Break Number	Type Break	Effect on Required Components	Resolution
SM1-01R	Circumferential (Forward Flow) (Reverse Flow)	Unacceptable damage to Structural Concrete and essential Piping Systems and Components located Inside Containment	Add Pipe Whip Restraint
SM1-02R	Circumferential (Forward Flow) (Reverse Flow)	Unacceptable damage to Structural Concrete and essential Piping Systems and Components located Inside Containment	Add Pipe Whip Restraint
SM1-03R	Circumferential (Forward Flow) (Reverse Flow)	Unacceptable damage to Structural Concrete and essential Piping Systems and Components located Inside Containment	Add Pipe Whip Restraint
SM1-04R	Circumferential (Forward Flow) (Reverse Flow)	Unacceptable damage to Structural Concrete and essential Piping Systems and Components located Inside Containment	Add Pipe Whip Restraint
SM1-05R	Circumferential (Forward Flow) (Reverse Flow)	Unacceptable damage to Structural Concrete and essential Piping Systems and Components located Inside Containment	Add Pipe Whip Restraint
SM1-06R	Circumferential (Forward Flow) (Reverse Flow)	Unacceptable damage to Structural Concrete and essential Piping Systems and Components located Inside Containment	Add Pipe Whip Restraint
SM1-07R	Circumferential (Forward Flow) (Reverse Flow)	Unacceptable damage to Structural Concrete and essential Piping Systems and Components located Inside Containment	Add Pipe Whip Restraint

TABLE C-4

(Continued)

Results of Damage Evaluation
Unacceptable Pipe Whip/Jet Impingement Effects
For Main Steam Line Inside the Containment

Break Number	Type Break	Effect on Required Components	Resolution
SM1-08R	Circumferential (Forward Flow) (Reverse Flow)	Unacceptable damage to Structural Concrete and essential Piping Systems and Components located Inside Containment	Add Pipe Whip Restraint
SM1-09R	Circumferential (Forward Flow) (Reverse Flow)	Unacceptable damage to Structural Concrete and essential Piping Systems and Components located Inside Containment	Add Pipe Whip Restraint
SM1-010R	Circumferential (Forward Flow) (Reverse Flow)	Unacceptable damage to Structural Concrete and essential Piping Systems and Components located Inside Containment	Add Pipe Whip Restraint
SM1-011R	Circumferential (Forward Flow) (Reverse Flow)	Unacceptable damage to Structural Concrete and essential Piping Systems and Components located Inside Containment	Add Pipe Whip Restraint
SM1-012R	Circumferential (Forward Flow) (Reverse Flow)	Unacceptable damage to Structural Concrete and essential Piping Systems and Components located Inside Containment	Add Pipe Whip Restraint

TABLE C-5

Results of Damage Evaluation
Unacceptable Pipe Whip/Jet Impingement Effects
For Feedwater Line

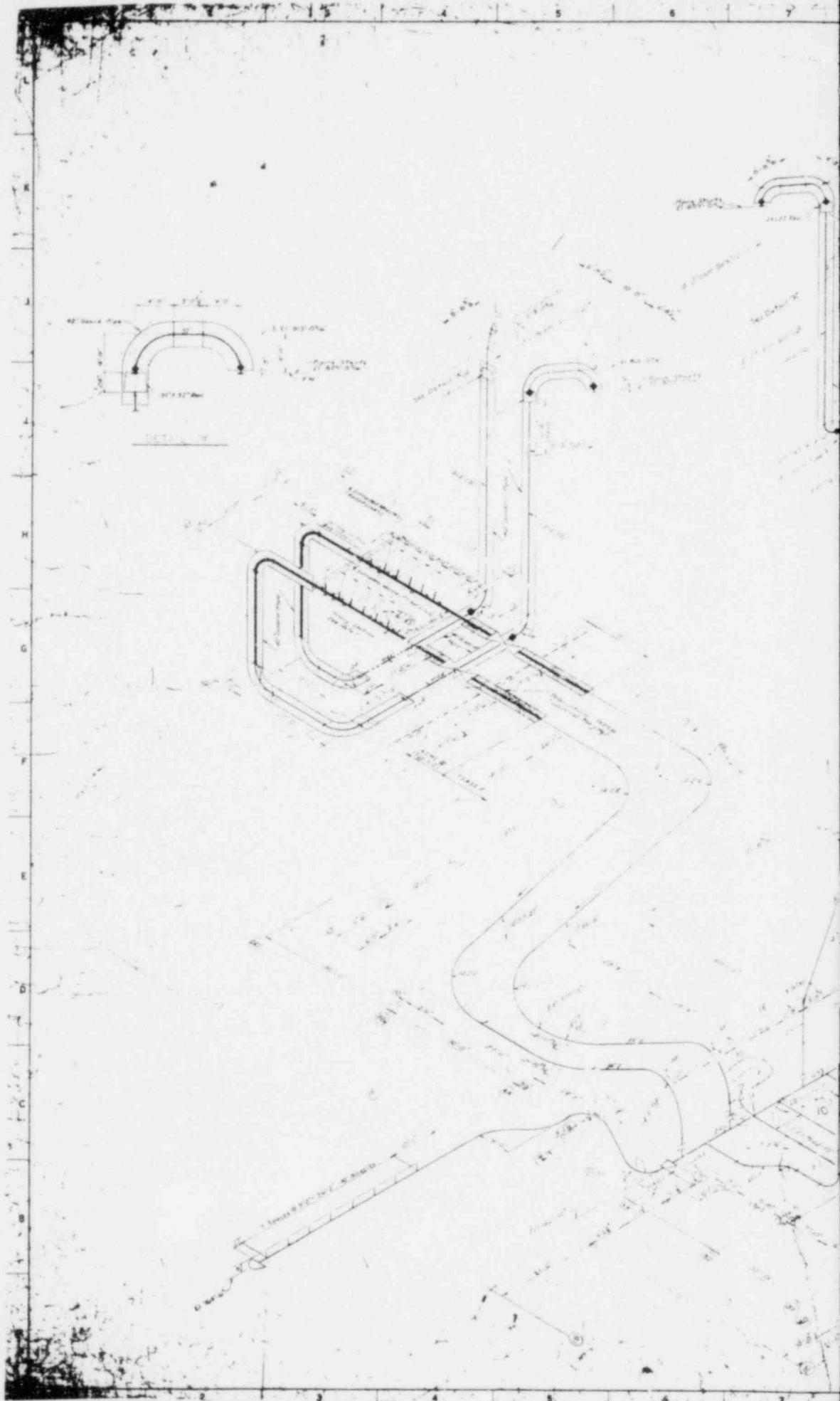
Break Number	Type Break	Effect on Required Components	Resolution
CF1-01R	Circumferential (Forward Flow) (Reverse Flow)	Unacceptable damage to Crane Wall and essential Piping Systems and Components located Inside Containment	Add Pipe Whip Restraint
CF1-02R	Circumferential (Forward Flow) (Reverse Flow)	Unacceptable damage to Crane Wall and essential Piping Systems and Components located Inside Containment	Add Pipe Whip Restraint
CF1-03R	Circumferential (Forward Flow) (Reverse Flow)	Unacceptable damage to Crane Wall and essential Piping Systems and Components located Inside Containment	Add Pipe Whip Restraint
CF1-04R	Circumferential (Forward Flow) (Reverse Flow)	Unacceptable damage to Crane Wall and essential Piping Systems and Components located Inside Containment	Add Pipe Whip Restraint

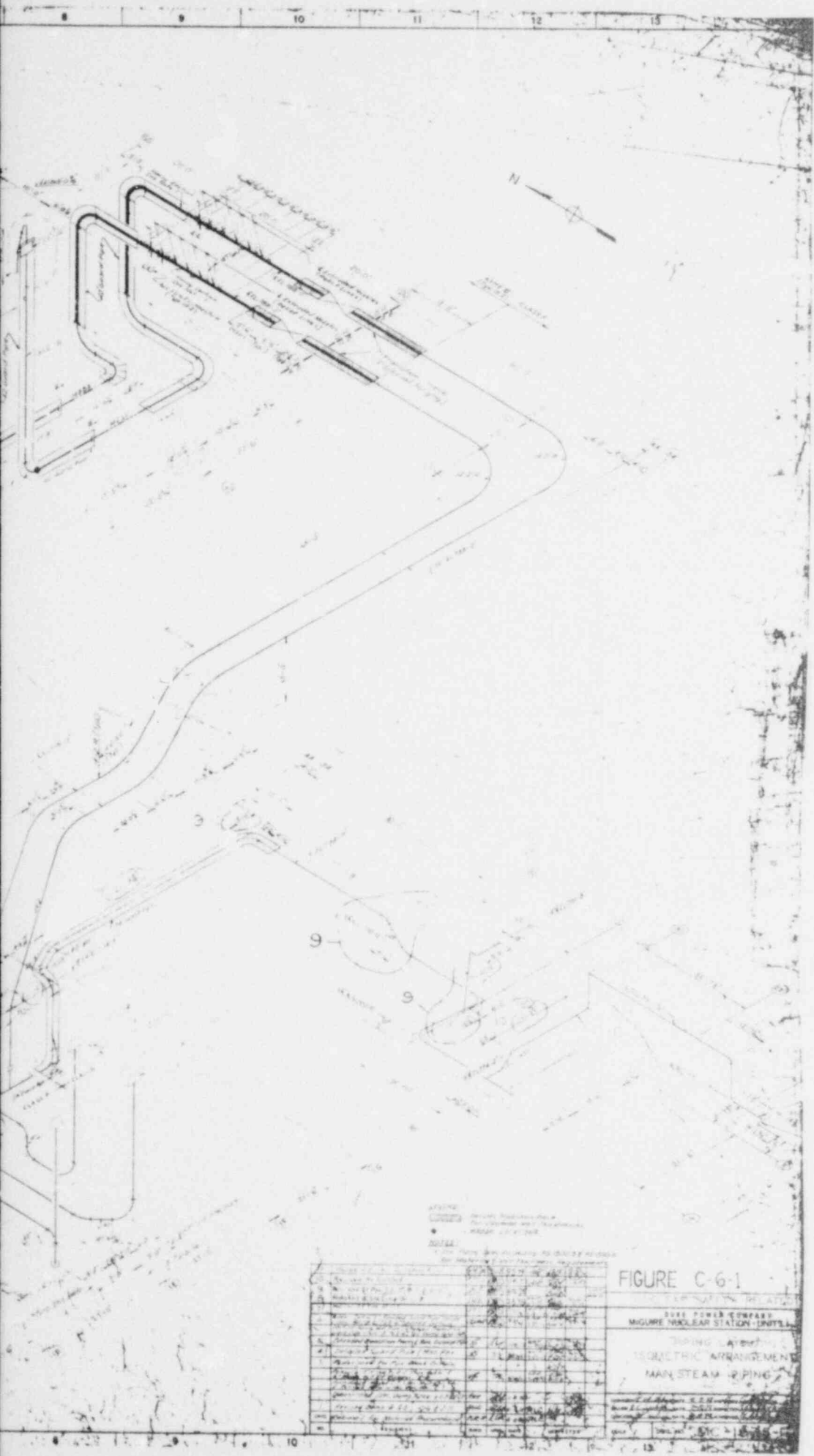
TABLE C-6

Results of Damage Evaluation

Unacceptable Pipe Whip/Jet Impingement Effects
For Auxiliary Feedwater Line

Break Number	Type Break	Effect on Required Components	Resolution
CA1-01R	Circumferential (Forward Flow) (Reverse Flow)	Unacceptable damage to Steel Enclosure of Steam Generator	Add Pipe Whip Restraint
CA1-02R	Circumferential (Forward Flow) (Reverse Flow)	Unacceptable damage to Steel Enclosure of Steam Generator	Add Pipe Whip Restraint
CA1-03R	Circumferential (Forward Flow) (Reverse Flow)	Unacceptable damage to Steel Enclosure of Steam Generator	Add Pipe Whip Restraint
CA1-04R	Circumferential (Forward Flow) (Reverse Flow)	Unacceptable damage to Steel Enclosure of Steam Generator	Add Pipe Whip Restraint
CA1-05R	Circumferential (Forward Flow) (Reverse Flow)	Electrical Cable Tray segments 11202 and 11149, 3" NC Line including valve 1NC64, 2" NI Lines including valves 1NI157 and 1NI160	Add Pipe Whip Restraints
CA1-06R	Circumferential (Forward Flow) (Reverse Flow)	Electrical Cable Tray segments 11202 and 11149, 3" NC Line including valve 1NC64, 2" NI Lines including valves 1NI157 and 1NI160	Add Pipe Whip Restraint





1. All piping shall be designed to meet the requirements of the ASME Code, Section III, Division 1, Subsection NB, and shall be fabricated in accordance with the requirements of the ASME Code, Section III, Division 1, Subsection NB.

NO.	DESCRIPTION	DATE	BY	CHECKED
1	Issue for Review	10/1/68	J. H. [unclear]	[unclear]
2	Issue for Construction	10/1/68	J. H. [unclear]	[unclear]
3	Issue for [unclear]	10/1/68	J. H. [unclear]	[unclear]
4	Issue for [unclear]	10/1/68	J. H. [unclear]	[unclear]
5	Issue for [unclear]	10/1/68	J. H. [unclear]	[unclear]
6	Issue for [unclear]	10/1/68	J. H. [unclear]	[unclear]
7	Issue for [unclear]	10/1/68	J. H. [unclear]	[unclear]
8	Issue for [unclear]	10/1/68	J. H. [unclear]	[unclear]
9	Issue for [unclear]	10/1/68	J. H. [unclear]	[unclear]
10	Issue for [unclear]	10/1/68	J. H. [unclear]	[unclear]
11	Issue for [unclear]	10/1/68	J. H. [unclear]	[unclear]
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14	Issue for [unclear]	10/1/68	J. H. [unclear]	[unclear]
15	Issue for [unclear]	10/1/68	J. H. [unclear]	[unclear]

FIGURE C-6-1
 MAIN STEAM PIPING
 ISOMETRIC ARRANGEMENT
 SURE POWER COMPANY
 MORGAN NUCLEAR STATION - UNIT 1

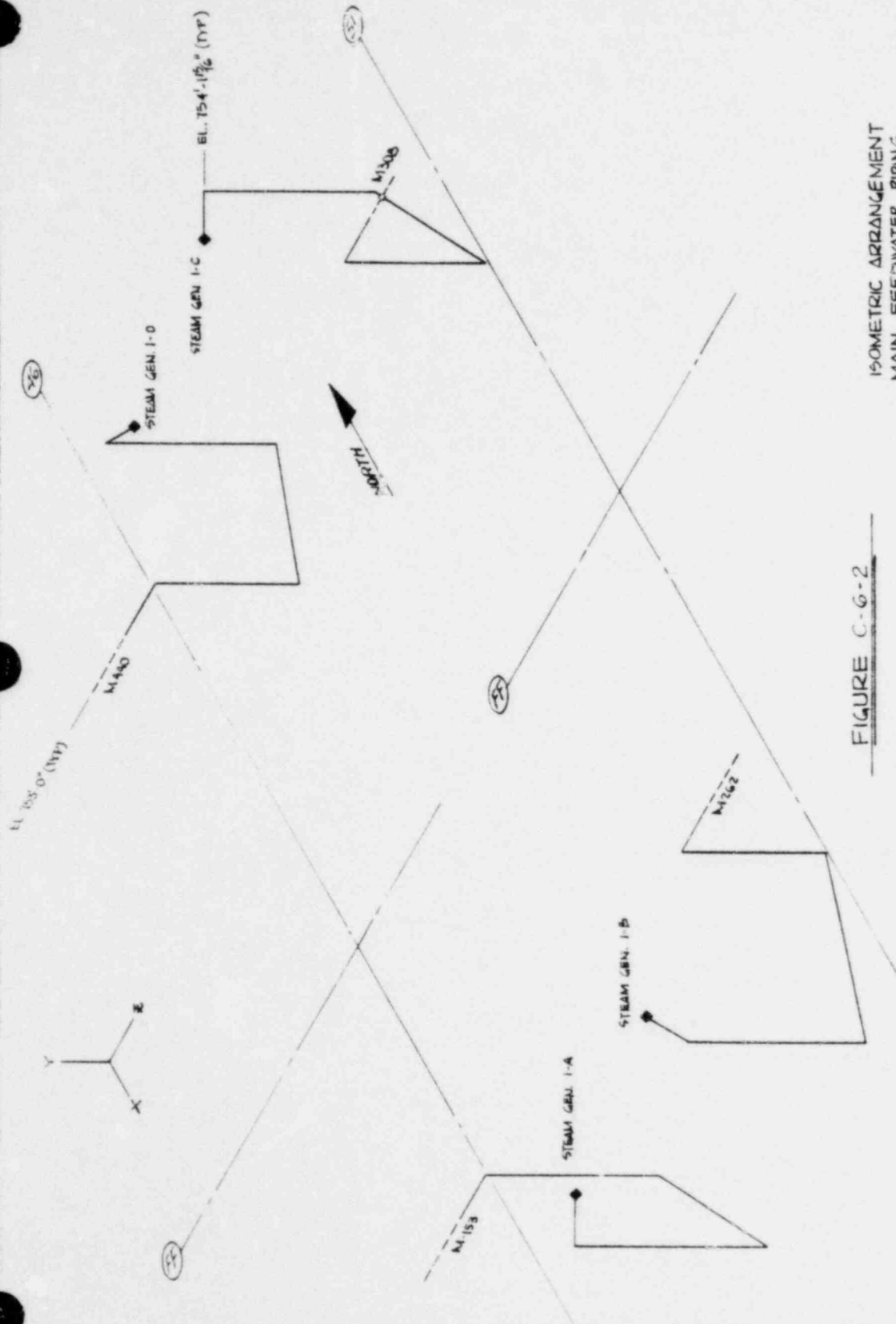
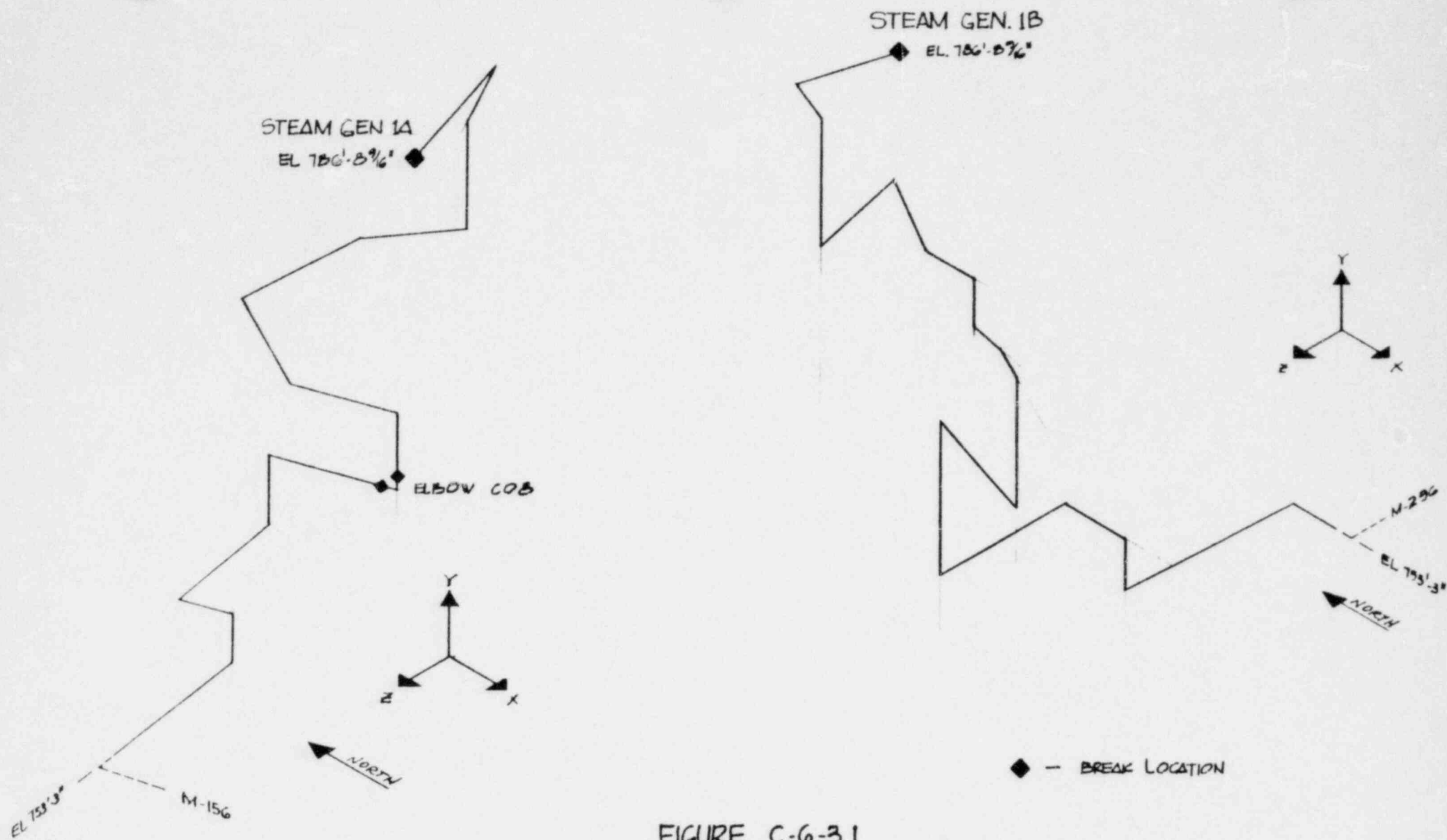


FIGURE C-6-2

ISOMETRIC ARRANGEMENT
MAIN FEEDWATER PIPING

◆ BREAK LOCATION



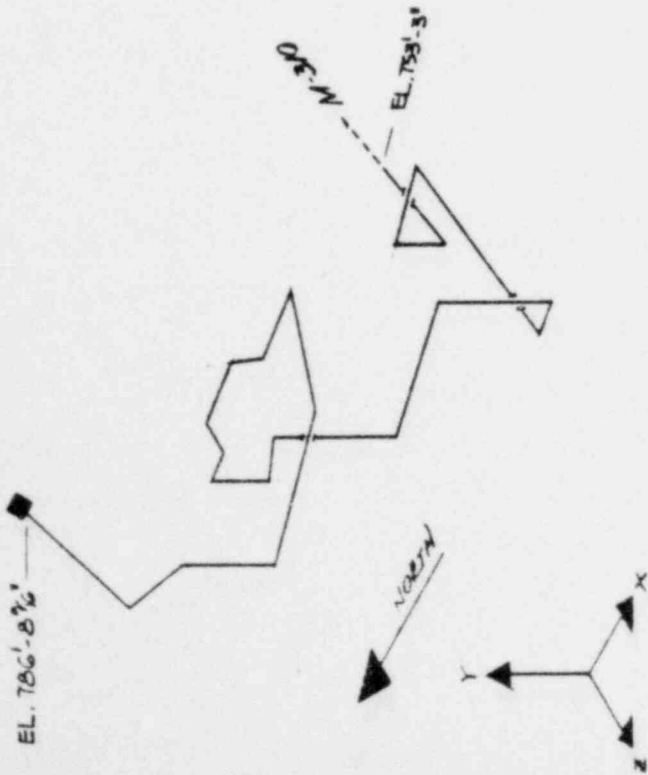
◆ - BREAK LOCATION

FIGURE C-6-3.1

ISOMETRIC ARRANGEMENT
 AUXILIARY FEEDWATER PIPING
 6-17-80 REV. 0

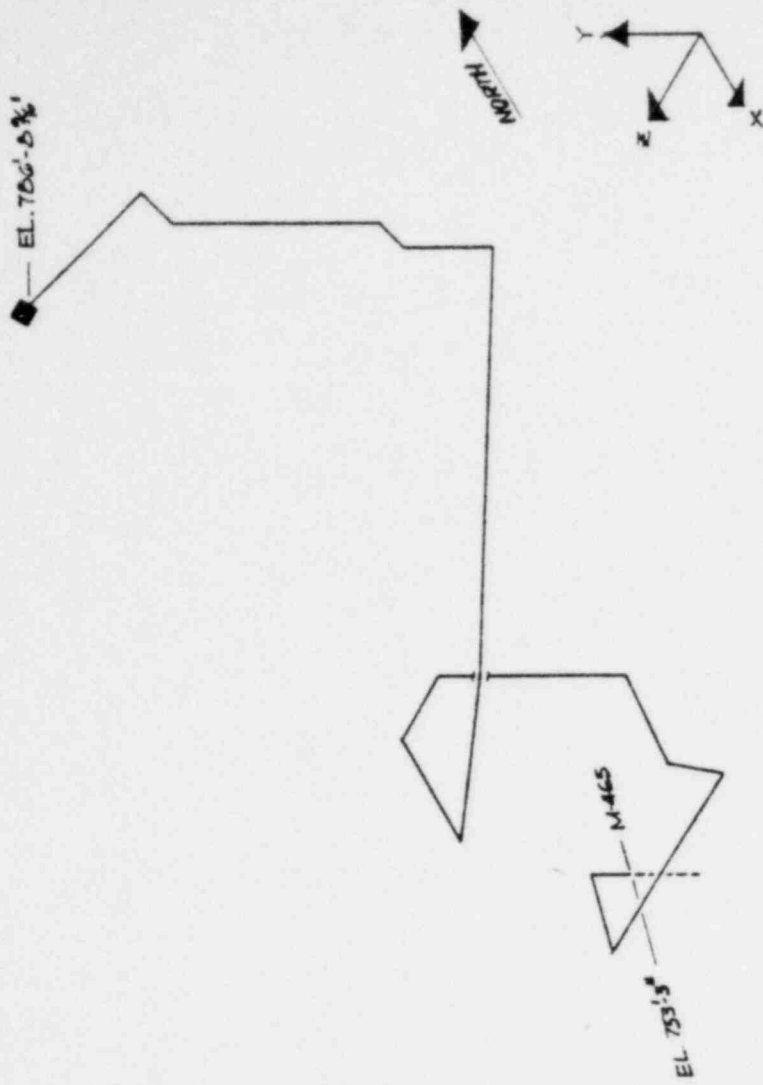
STEAM GEN. 1C

EL. 706'-8 3/4"



STEAM GEN. 1D

EL. 706'-8 3/4"



◆ BREAK LOCATION

FIGURE C-6-3.2

ISOMETRIC ARRANGEMENT
AUXILIARY FEEDWATER PIPING
G-10-00 REV. 0