REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:8303040536 DOC.DATE: 83/03/01 NOTARIZED: NO DOCKET #
FACIL:50-369 William B. McGuire Nuclear Station, Unit 1, Duke Powe 05000369
AUTH.NAME AUTHOR AFFILIATION
TUCKER,H.B. Duke Power Co.
RECIP.NAME RECIPIENT AFFILIATION
DENTON,H.R. Office of Nuclear Reactor Regulation, Director
ADENSAM,E.G. Licensing Branch 4

SUBJECT: Forwards addl info re mod to main feedwater lines to steam generators D2/D3 to alleviate certain forward flushing transients, per NRC 830218.

DISTRIBUTION CODE: A001S COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 23
TITLE: OR Submittal: General Distribution

NOTES:

	RECIPIENT ID CODE/NAME		COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME		COPIES LTTR	COPIES ENCL
	NRR LB4 BC	01	7	7				
INTERNAL:	ELD/HDS4		1	0	NRR/DE/MTEB		1	1
	NRR/DL DIR		1	1	NRR/DL/ORAB		1	0
	NRR/DSI/METB		1	1	NRR/DSI/RAB		1	1
	REG FILE	04	1	1	RGN2		1	1
EXTERNAL:	ACRS	09	6	6	LPDR	03	1	1
	NRC PDR	02	1	1	NSIC	05	1	1
	NTIS		1	1				

TOTAL NUMBER OF COPIES REQUIRED: LTTR 25 ENCL 23

ACCESSION NBR:[illegible] DOC.DATE: 85/05/01 NOTARIZED: NO DOCKET #
FACIL:50-369 William B. McGuire Nuclear Station, Unit 1, Duke Power Co. 05000369

AUTH.NAME	AUTHOR AFFILIATION
TUCKER,H.B.	Duke Power Co.
RECIP.NAME	**RECIPIENT AFFILIATION**
DENTON,H.R.	Office of Nuclear Reactor Regulation, Director
ADENSAM,E.G.	Licensing Branch 4

SUBJECT: Forwards addl info re mod to main feedwater lines to steam generators 1A/1B to alleviate certain forward flushing transients,per NRC 850218 ltr.

DISTRIBUTION CODE: A001D COPIES RECEIVED:LTR __ ENCL __ SIZE:__________
TITLE: OR Submittal: General Distribution

NOTES:

	RECIPIENT ID CODE/NAME		COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME		COPIES LTTR	COPIES ENCL
	NRR LB4 BC	01	7	7				
INTERNAL:	ELD/HDS4		1	0	NRR/DE/MTEB		1	1
	NRR/DL DIR		1	1	NRR/DL/ORAB		1	0
	NRR/DSI/METB		1	1	NRR/DSI/RAB		1	1
	REG FILE	04	1	1	RGN2		1	1
EXTERNAL:	ACRS	09	6	6	LPDR	03	1	1
	NRC PDR	02	1	1	NSIC	05	1	1
	NTIS		1	1				

TOTAL NUMBER OF COPIES REQUIRED: LTTR [illegible] ENCL [illegible]

DUKE POWER COMPANY
P.O. BOX 33189
CHARLOTTE, N.C. 28242

HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

TELEPHONE
(704) 373-4531

March 1, 1983

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Ms. E. G. Adensam, Chief
Licensing Branch No. 4

Re: McGuire Nuclear Station, Unit 1
Docket No. 50-369
D2/D3 Steam Generator Design Modification

Dear Mr. Denton:

My letter dated February 3, 1983 provided a report describing the program to be implemented by Duke Power Company at McGuire Nuclear Station-Unit 1 to address the recommendations made by the Design Review Panel in its report "Utility Design Review Panel, D2/D3 Steam Generator Design Modification", which was submitted on January 17, 1983. This program includes a modification to the main feedwater lines to the steam generator which is intended to alleviate certain forward flushing transients.

Ms. E. G. Adensam's (NRC/NRR) letter dated February 18, 1983 indicated that in order for the NRC to complete its review of this modification additional information would be required, which was requested in the form of 10 items. Please find attached the requested information.

Should there be any further questions in this matter, please advise.

Very truly yours,

Hal B. Tucker

Hal B. Tucker

PBN:jfw
Attachment

cc: Mr. James P. O'Reilly, Regional Administrator
U. S. Nuclear Regulatory Commission
Region II
101 Marietta Street, Suite 2900
Atlanta, Georgia 30303

Dr. M. W. Wambsganss
Argonne National Laboratories
9700 South Cass Avenue, Bldg. 335
Argonne, Illinois 60439

Senior Resident Inspector
McGuire Nuclear Station

A001

DUKE POWER COMPANY
McGUIRE NUCLEAR STATION, UNIT 1
D2/D3 STEAM GENERATOR DESIGN MODIFICATION
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

1. Q. Figure 2 attached to the utility's submittal dated February 3, 1983 does not indicate the normal position of the valves. Provide a figure which shows the normal position of the valves during startup (with the warmup line in operation) and a figure showing the valves in the normal full power configuration.

A. See Fig. 1 for valve positions during startup with the warmup line in operation.

See Fig. 2 for valve positions during normal full power operation.

2. Q. Provide the seismic category, quality group and code classifications of all new piping and valves.

A. The classification of all new piping and valves in the Reactor Building (RB) and the doghouses (DH) is seismic Category 1, ASME Section III, Class 2.

All new piping and valves in the Turbine Building (TB) are non-seismic ANSI B31.1.

See Fig. 1. New piping and valves shown by dotted lines. Building boundaries also shown.

3. Q. Part 1 Provide a discussion of the anticipated operating procedure using the proposed warmup modification. Include the sequence of valve operations as part of the discussion.

A. Refer to Fig. 1 for discussion of A steam generator (typical)

Main feedwater flow is supplied to the auxiliary feedwater nozzle through valves CF121 (feedwater check valve), CF160 (feedwater preheated bypass check valve) and CF126B (feedwater preheater bypass valve). Flow is regulated by the control bypass valve CF104AB below 15% power and by the control valve CF32AB otherwise. The feedwater isolation valve CF35AB is closed to prevent flow to the main nozzle. Valves CF134A, CF151B, and CF150 are closed and CF187 open to prevent flow to the main nozzle through the warmup line prior to initiating warming.

To initiate warming, valves CF134A and CF151B are opened to pressurize piping. Pressurization rate and purge flow are controlled by orifice downstream of CF188. After opening above valves, valve CF187 is opened to initiate flow. Thermocouples downstream of valve CF35AB and upstream of CF34 are monitored to determine that temperatures are satisfactory to initiate forward feedwater flow through the main nozzle.

When temperature limits are met on all steam generators valve CF187 is closed. Valve CF134A is then closed.

Valve CF150 is then opened to provide tempering flow to the auxiliary feedwater nozzles through valve CF151.

Forward feed flow can be initiated by opening feedwater isolation valve CF35AB provided all feedwater temperature and reactor power limits are met.

Valve CF126B is closed to isolate the feedwater preheater bypass. See Fig. 2.

3. Q. Part 2 Explain how the valve bypassing the containment isolation check valve on the main feedwater line will be controlled during the warmup procedure and power operation.

A. See Fig. 1 for discussion of A steam generator.

CF34 is not the containment isolation check. Containment isolation is provided by check valve CF121. CF34 originally served as the stress analysis boundary between piping analyzed for steam generator design temperature and piping analyzed for feedwater system design temperature. Piping is currently analyzed for steam generator design temperature back to isolation valves CF35AB, CF175 and CF187. Therefore an open bypass around CF34 does not adversely affect any current safety analysis.

4. Q. Identify which valves are operable from the control room, from the remote shutdown panel, and from both the control room and the remote shutdown panel. For those valves which are operable from both locations, verify that transfer switches will be installed to isolate the valves from the control room in the event of a fire in the control room. Also discuss the position indication of the above valves.

A. All motor operated valves in the warmup flow path are controlled from and have position indication in the control room only. The warmup line is a potential leak path from spurious valve operation due to a control room or cable spreading room fire; however, flow is limited by orifices to approximately 40 gpm per steam generator which when combined with previously analyzed leakage due to fires does not hinder the ability of the auxiliary feedwater pump to supply adequate flow for heat removal.

5. Q. Chapter 15 of the FSAR discusses the design basis accidents and how they would affect the McGuire Nuclear Station Unit 1. For each design basis accident for which the additional water loss due to the proposed containment isolation check valve bypass valve being open could have an effect, provide the results of revised analysis to verify that the FSAR analysis bounds the case in which this valve is open. The valve could be open either intentionally or assumed to fail in the open position.

A. Refer to Fig. 1 for discussion of steam generator A. See question 3 for discussion of check valve CF34 bypass. Valves CF134A and CF151B receive auxiliary feedwater pump automatic start signals from the corresponding trains to isolate, thus preventing water loss beyond design basis accident analyses. Therefore, check valve CF34 bypass has no adverse affect on any safety analysis.

6. Q. Provide a discussion of the procedures to test and inspect the new valves and piping in order to meet the requirements of General Design Criteria 45 and 46.

A. No additional active valves were added as a result of this modification. Existing inservice inspection procedures will therefore be used to verify operability of these valves and safety related piping added. Manual valves added will be verified in the valve check list in the operating procedures.

Eight (two per steam generator) active valves in the warmup flow path receive additional engineered safety feature signals to close on automatic start of the corresponding train motor driven auxiliary feedwater pumps. These valves will require additional testing per Technical Specifications for response times for the signal added.

7. Q. Identify which piping and valves will be provided with tornado missile and tornado protection. For those components where this protection is not provided, verify that the failure of these components concurrent with the loss of offsite power and the most limiting single failure will not result in unacceptable consequences.

A. All piping and valves inside the Doghouses and Reactor Building are protected against tornado missiles and tornadoes. Piping and valves outside doghouses and Reactor Building are not protected. See Fig. 1 for building boundaries. Loss of these unprotected components concurrent with the loss of offsite power and the most limiting single failure will not result in unacceptable consequences.

8. Q. Part 1 Since the piping valves will be subjected to a temperature higher than 200°F or a pressure greater than 275 psig, provide the results of a high energy pipe break analysis in accordance with the Standard Review Plan Section 3.6.1. Provide drawings which show the piping configurations, postulated break locations, the jet cones which result from the escaping fluid, and the location of the pipe whip restraints. For those portions of the proposed system which do not have a temperature higher than 200°F and do not have a pressure greater than 275 psig, provide the results of the moderate energy pipe crack analysis in accordance with the acceptance criteria of Standard Review Plan Section 3.6.1.

A. The piping and valves associated with the Steam Generator modification shall be subjected to approximate operating temperatures of 441° - 556°F, an approximate operating pressures of 1105 - 1052 psia. As such this piping is high energy and is analyzed in accordance with the requiremetns of the Standard Review Plan Section 3.6.1. The main and auxiliary feedwater system design parameters associated with the Steam Generator changes are delineated in the McGuire Nuclear Station modification (NSM) No. 1100.

The modifications to the feedwater system piping have resulted in seven (7) new intermediate break locations on Duke Class B feedwater piping in the outboard doghouse. There are no changes in intermediate break locations on feedwater piping in the inboard doghouse. An initially determined intermediate break location on feedwater piping in the inboard doghouse was not changed, since at the new second highest stress location none of the following conditions exist:

1. Maximum stress range exceeds the threshold level of 0.8 $(1.2\ S_H + S_A)$

2. A change is required in pipe parameters, such as major differences in pipe size or wall thickness.

3. The new highest stress location is $0.4\ (1.2\ S_H + S_A) < S \leq 0.6\ (1.2\ S_H + S_A)$, and the stress at the new stress location is > 20% higher than the original break location stress, and results in unacceptable consequences to safety related systems.

Table-1 identifies the seven (7) new break locations/designations and the new protective device requirements resulting from the Steam Generator modification.

Figures CAP-A-1 thru 4 and CAO-D-1 & 2 provide the jet cones which result from the escaping fluid issuing from the new break locations requiring protective devices.

The location of the new pipe whip restraints which are required by the Steam Generator modification are shown on isometric Math Models CAP (sh. 2 of 3) and CAO (sh. 3 of 3).

8. Q. Part 2 As part of the pipe failure analysis, include a discussion of the means to isolate the pipe break/crack, a discussion of potential flooding including the means to remove the water, and verification of the capacity of the means to remove the water, and a discussion of the affects of wetting nearby equipment due to water impingement, splashing or dripping.

A. Refer to Fig. 1 for a discussion of Steam Generator A.

All new break locations resulting from this modification are located in the doghouses. The doughouses are provided with safety related level switches which generate a feedwater isolation signal on high level to isolate forward feedwater flow into the doghouses. Valves CF151B and CF134A receive a feedwater isolation signal to close as do the feedwater isolation and control and preheater bypass isolation valves. The doghouses and safety related equipment have been analyzed for the flood level and impingement resulting from a double ended break in the largest line with complete loss of steam generator inventory and continued auxiliary feedwater addition for 30 minutes. All inventory from this break is contained in the doghouse below the level of safety related equipment.

9. Q. Verify that the new valves will not produce internally generated missiles, or verify that safety-related equipment is provided protection from these missiles, in accordance with the Standard Review Plan Section 3.5.1.1 and 3.5.1.2. For each internally generated missile, verify that no secondary missiles will be generated.

   A. Valves are not considered credible missile sources, as stated in FSAR Subsection 3.5.2.4.

10. Q. Verify that no sources of internally generated missiles are located in the same compartment as the proposed piping and valves. If this cannot be verified then either provide protection from the missile or verify the failure of the new component from the missile will not have any adverse effect on the plant.

A. We have reviewed the effect of credible missiles on the new feedwater piping and found the piping is shielded from all credible missiles listed in section 3.5.2 of the FSAR. The results of this review are listed below.

1) Turbine Building

No safety related piping was added in the Turbine Building.

2) Doghouses

No missiles from section 3.5.2 will affect piping in the doghouse with the exception of valves, subsection 3.5.2.4. For the same reasons stated in this subsection, valve failures should not be considered credible missiles. The new piping is shielded from all other credible missiles listed in 3.5.2 by the reactor building wall and the doghouse wall.

3) Reactor Building

No credible missiles from section 3.5.2 can affect the new piping in the Reactor Building. The new piping is protected by the reactor building wall and crane wall from external missiles and by the steam generators, feedwater piping and reactor vessel shield wall from internal missiles.

Fig. 1 Startup Valve Positions (with warmup line in operation)

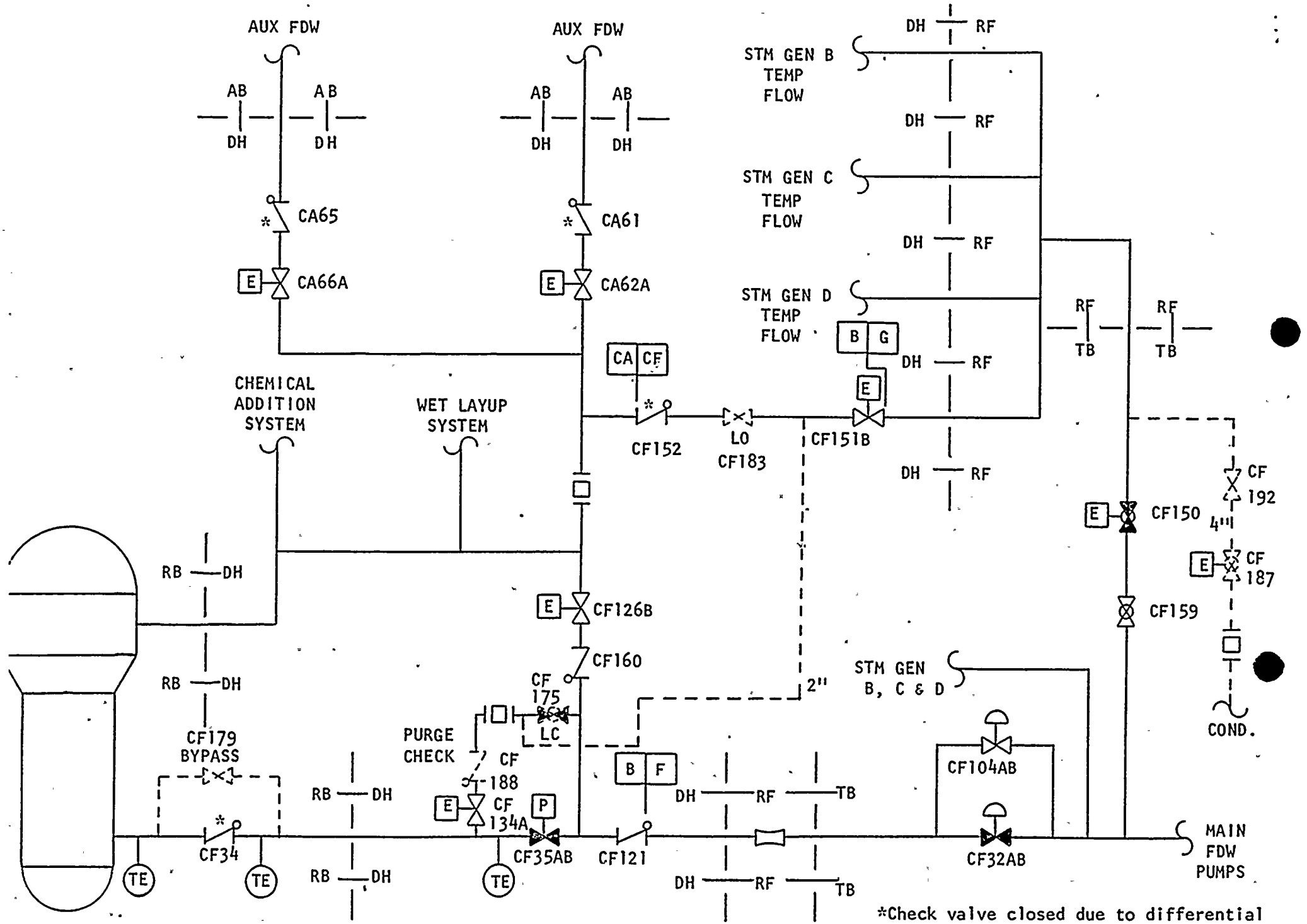


Fig. 2 Normal Valve Positions - Full Power

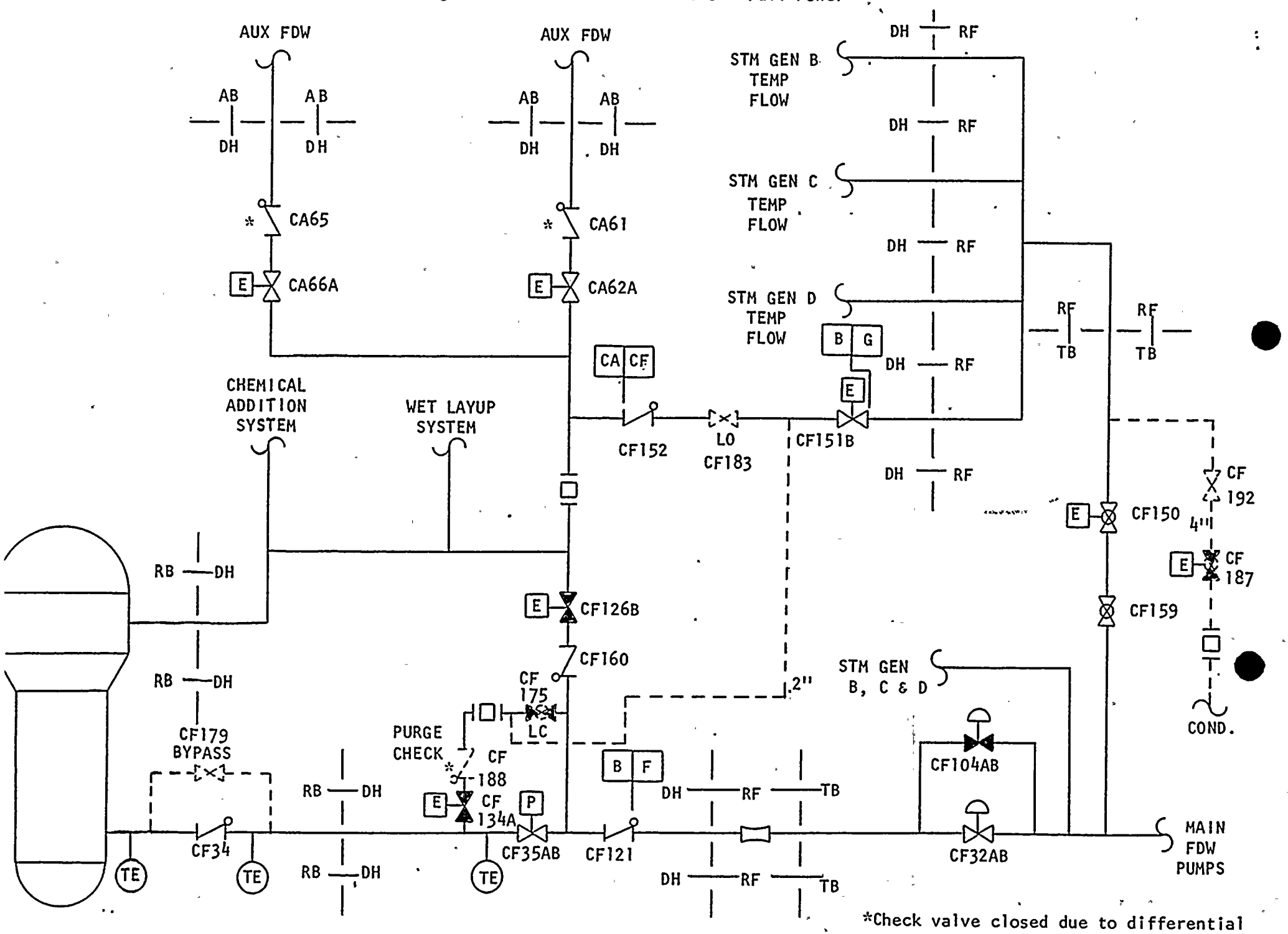
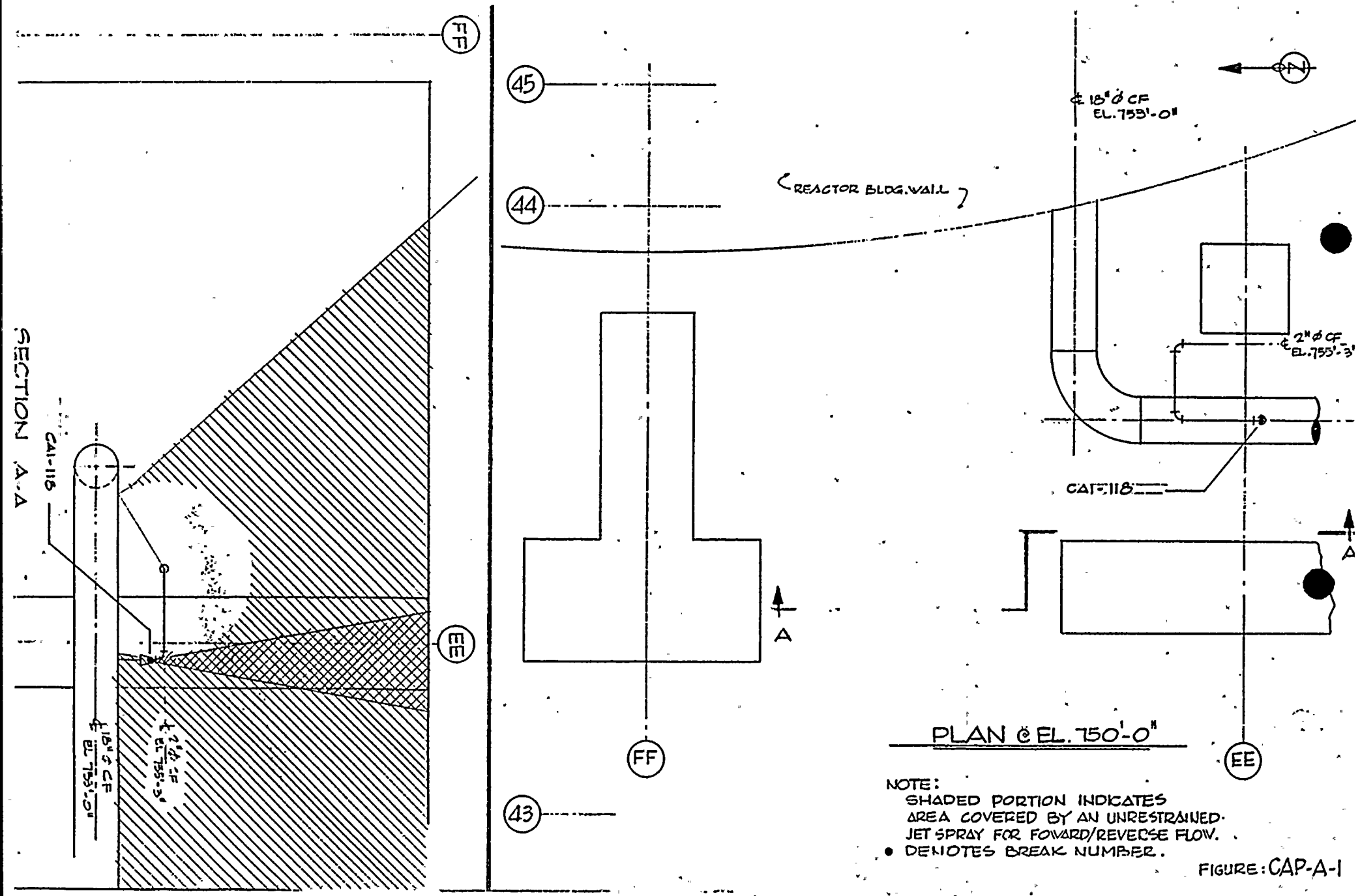


TABLE-1

HIGH ENERGY BREAK LOCATIONS

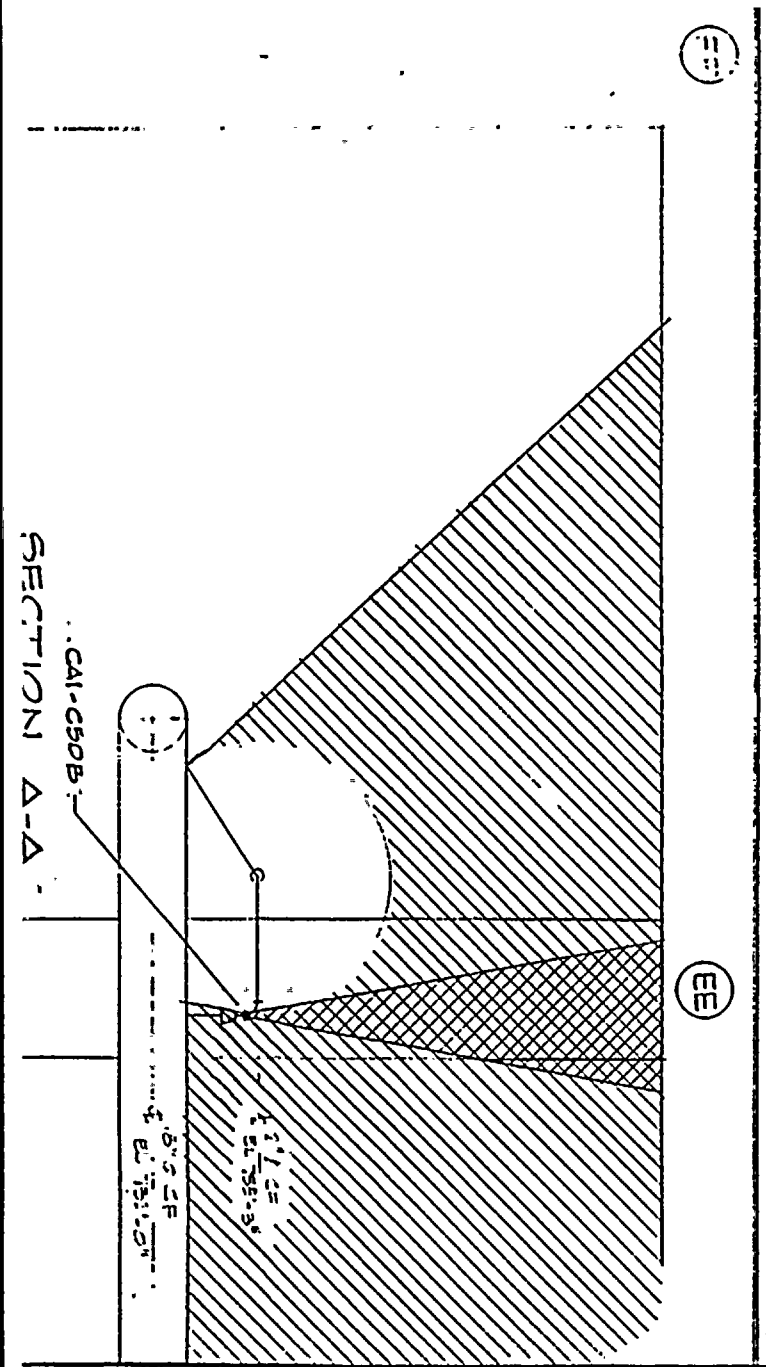
BREAK NO.	SYSTEM	DESCRIPTION	STRESS RATIO	FIGURE NO.	LOCATION	PROTECTION REQUIREMENTS
CA1-118	CA	2" CF line at the outlet of 4" x 2" reducer upstream of valve 1CF 134A.	1.033	CAP-A-1	AB-750-26	None for forward flow (FF) RR for reverse flow (RF)
CA1-C50B	CA	2" CF line at inlet of third elbow upstream of valve 1CF 134A.	1.016	CAP-A-2	AB-750-26	None for FF RR for RF
CA1-C48B	CA	2" CF line at inlet of first elbow upstream of - valve 1CF 134A.	1.095	CAP-A-3	AB-750-26	RRs for FF None for RF
CA1-C48A	CA	2" CF line at the outlet of first elbow upstream of valve 1CF 134A.	1.138	CAP-A-4	AB-750-26	RR for FF RR for RF
CA1-C58A	CA	2" CF line at the inlet of first 90° elbow upstream of valve 1CF 178.	0.860	CAO-D-1	AB-750-26	RR for FF RR for RF
CA1-C58B	CA	2" CF line at the outlet of first 90° elbow upstream of valve 1CF 178.	0.860	CAO-D-2	AB-750-26	RR for FF None for RF
CA1-C48B	CA	6" CF line at the inlet of first elbow upstream of valve 1CF 163	0.821	N/A	AB-750-26	None for FF None for RF

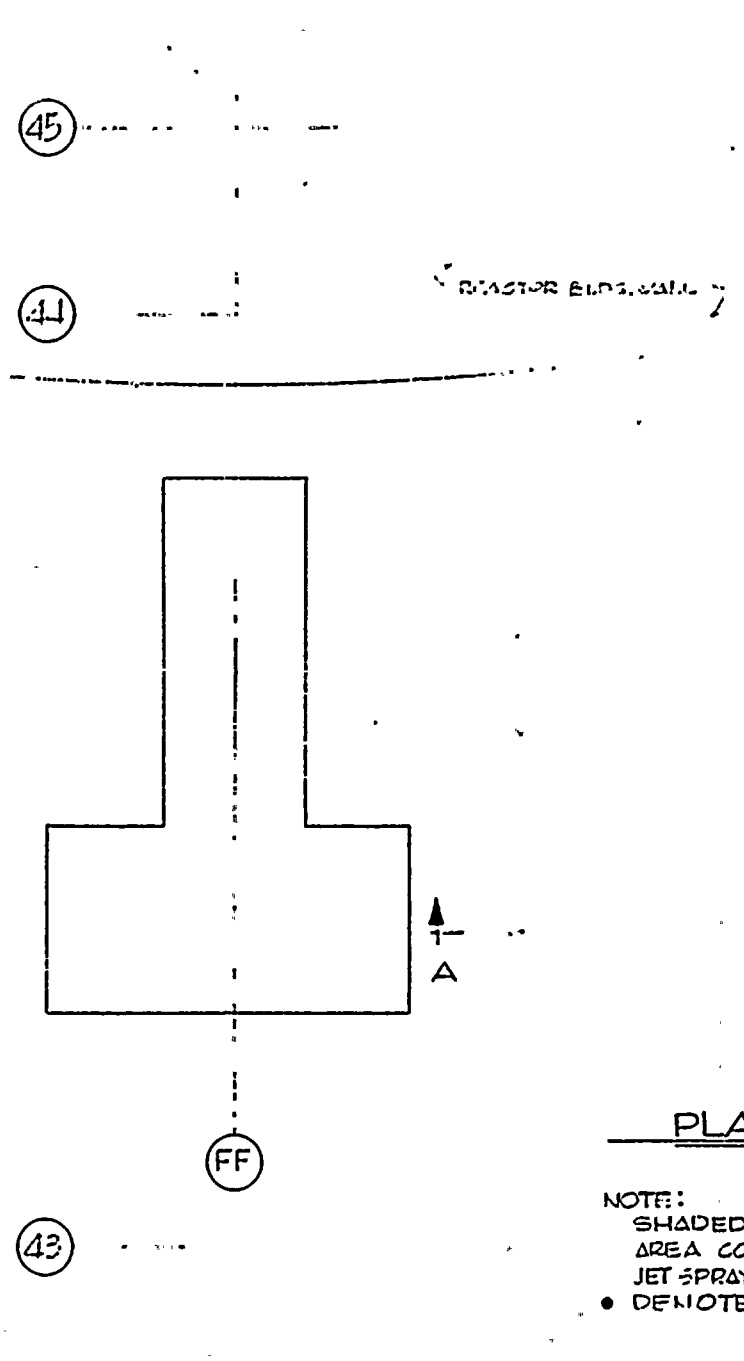
Note: RR - Rupture Restraint.

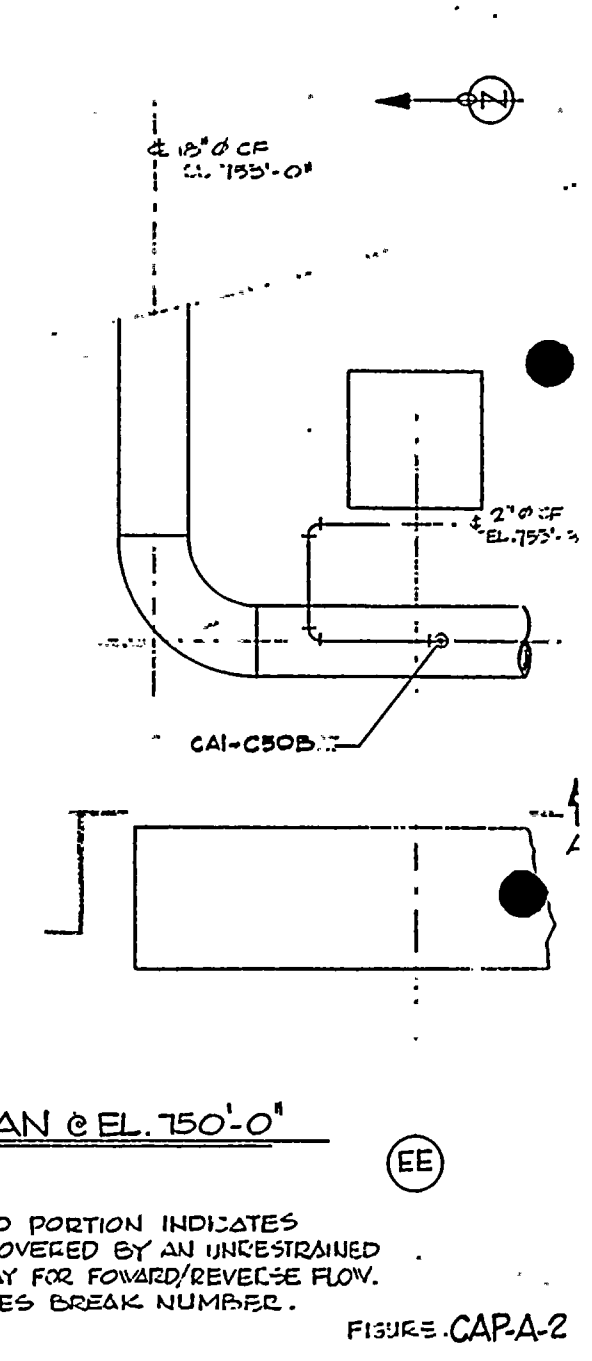


NOTE:
SHADED PORTION INDICATES AREA COVERED BY AN UNRESTRAINED JET SPRAY FOR FOWARD/REVERSE FLOW.
● DENOTES BREAK NUMBER.

FIGURE: CAP-A-1







PLAN @ EL. 750'-0"

EE

NOTE:
SHADED PORTION INDICATES AREA COVERED BY AN UNRESTRAINED JET SPRAY FOR FOWARD/REVERSE FLOW.
• DENOTES BREAK NUMBER.

FIGURE CAP-A-2

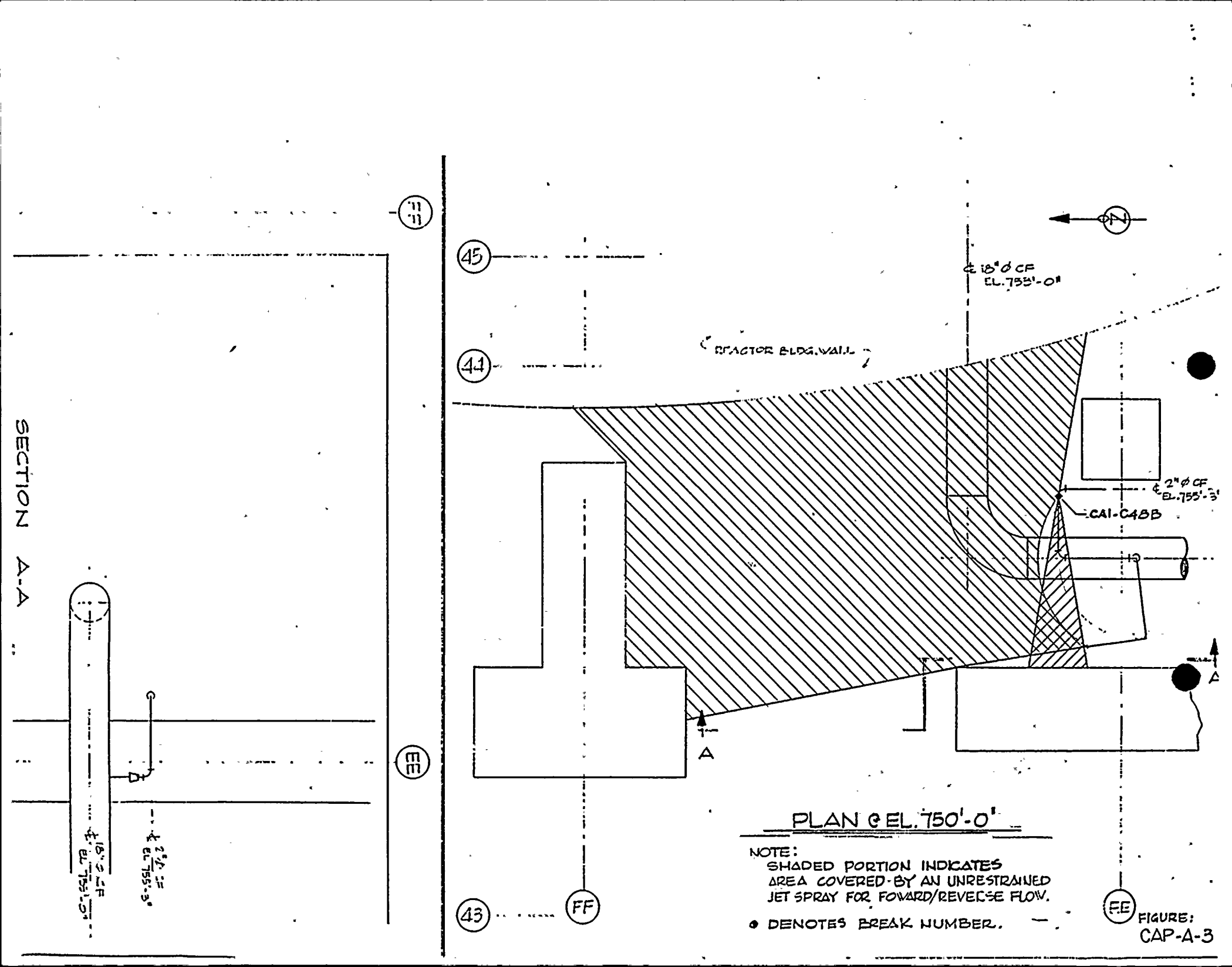


FIGURE: CAP-A-3

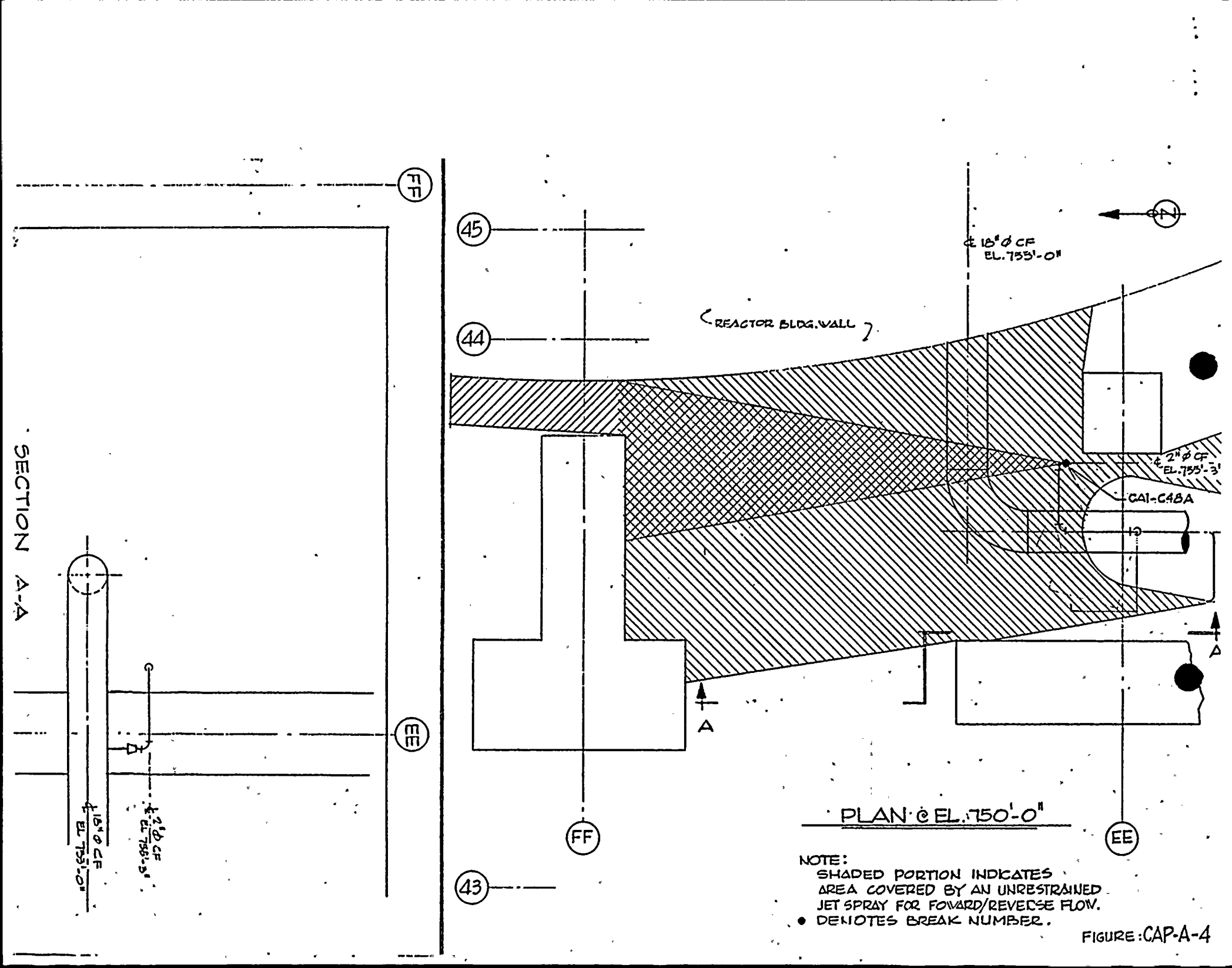
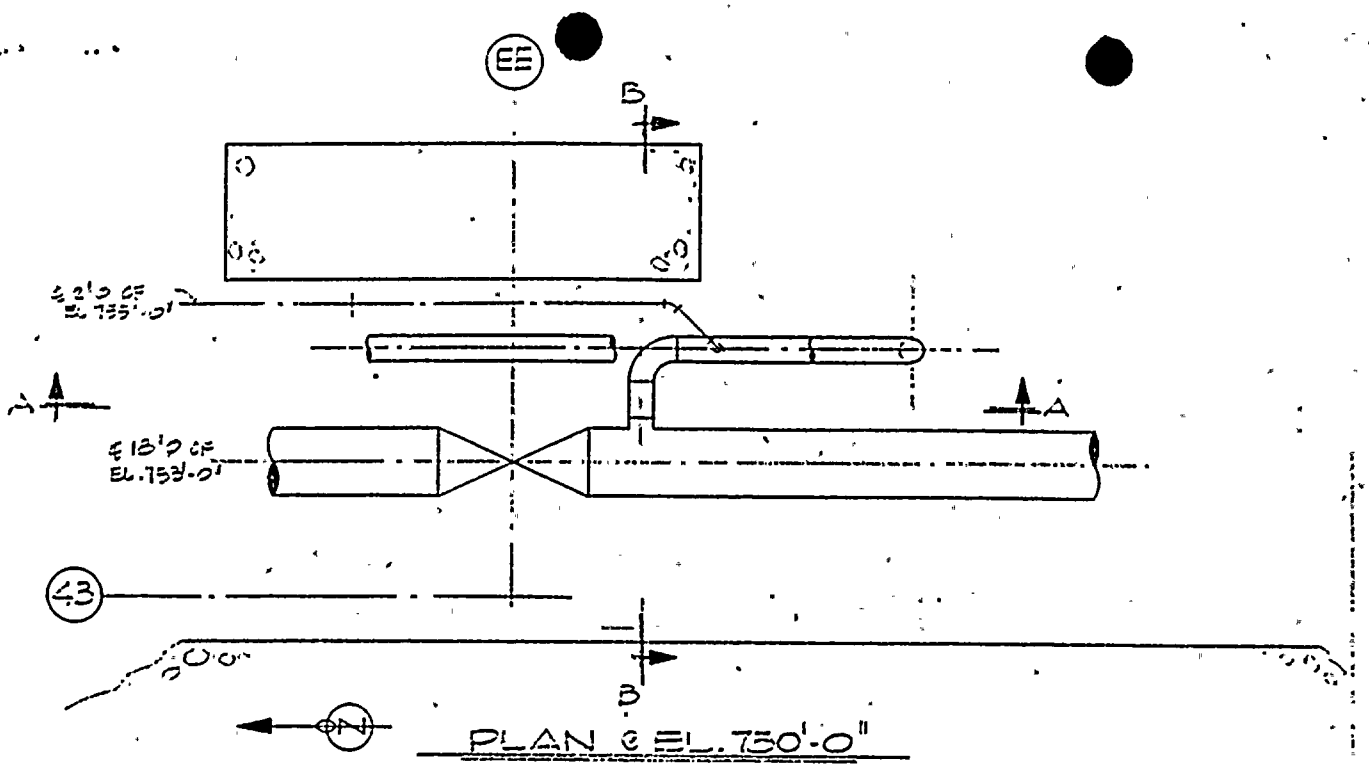
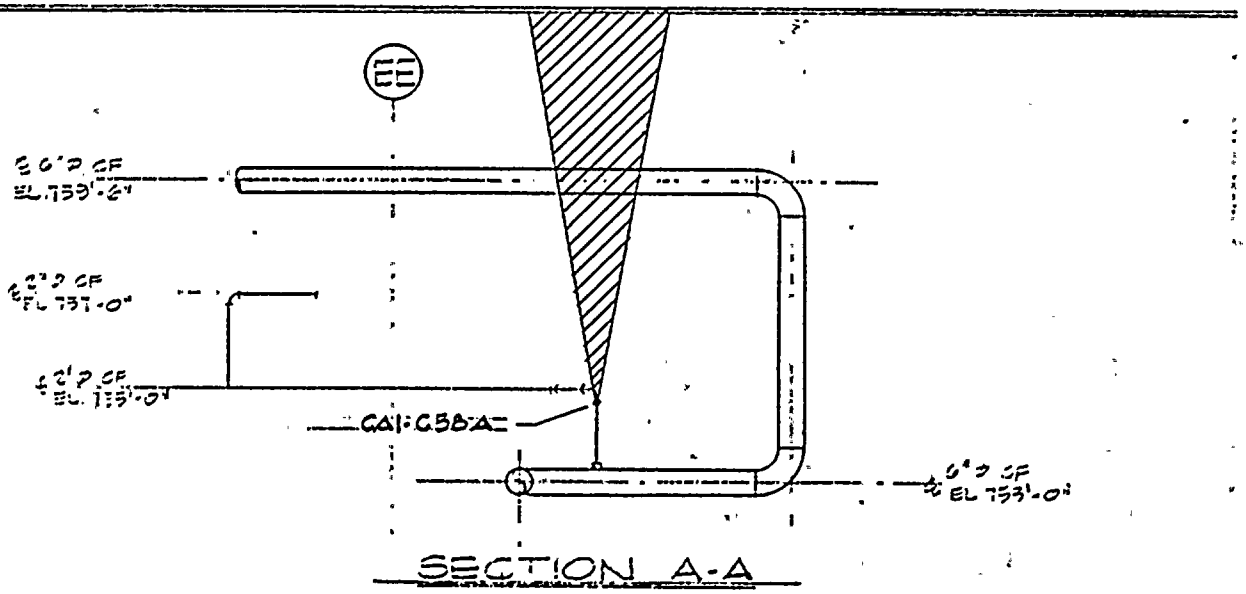
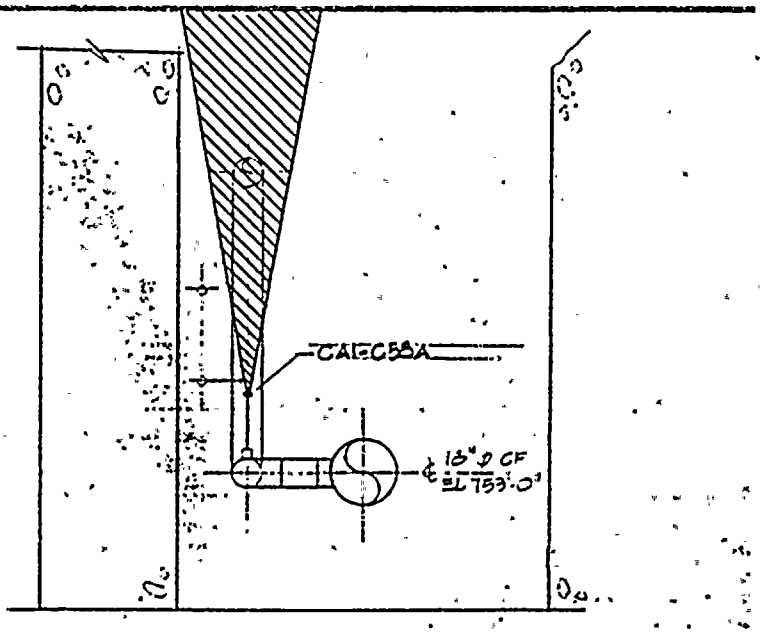


FIGURE: CAP-A-4

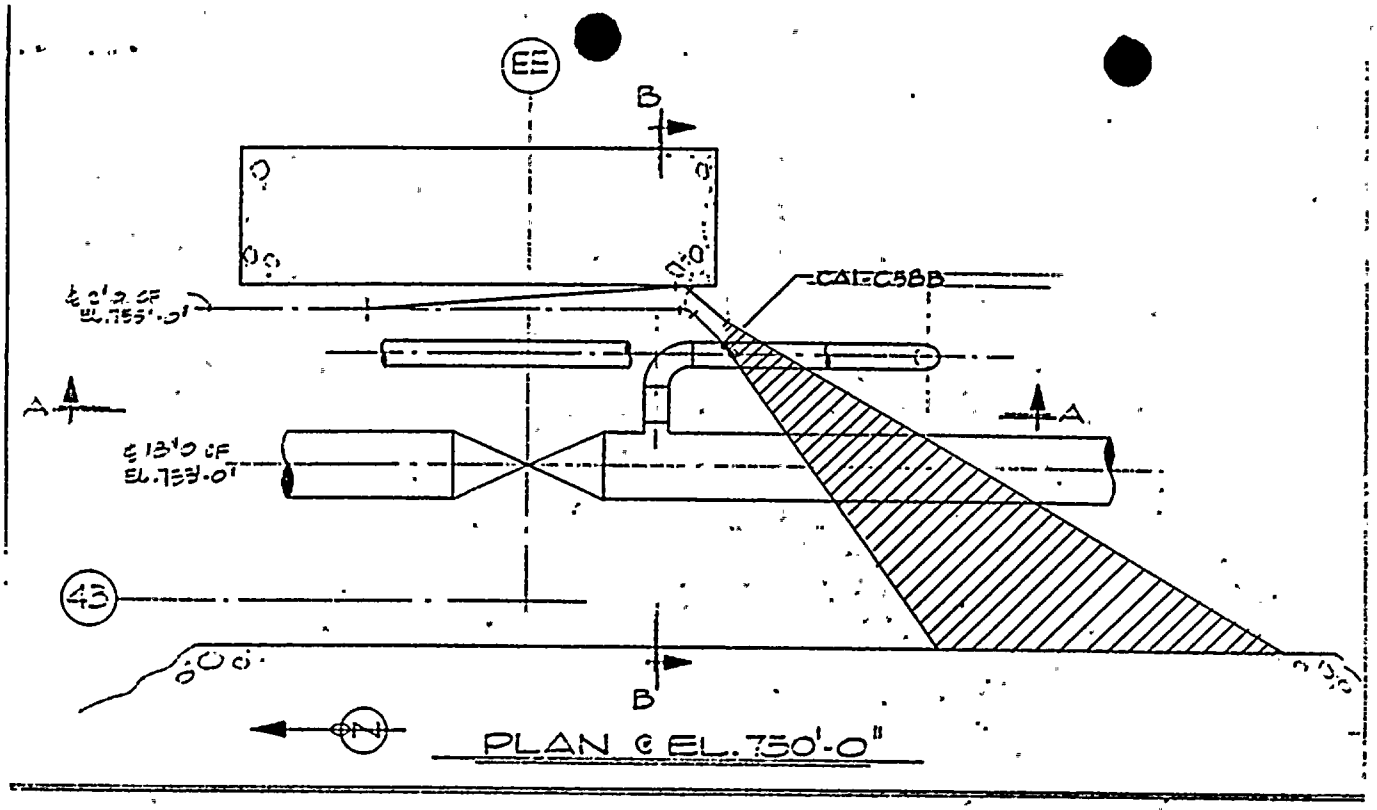


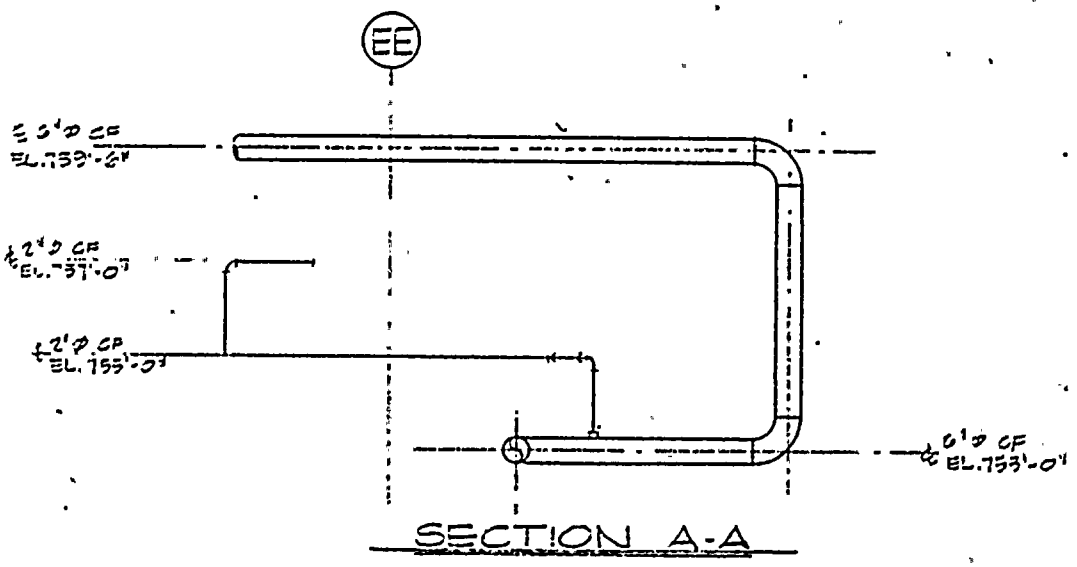


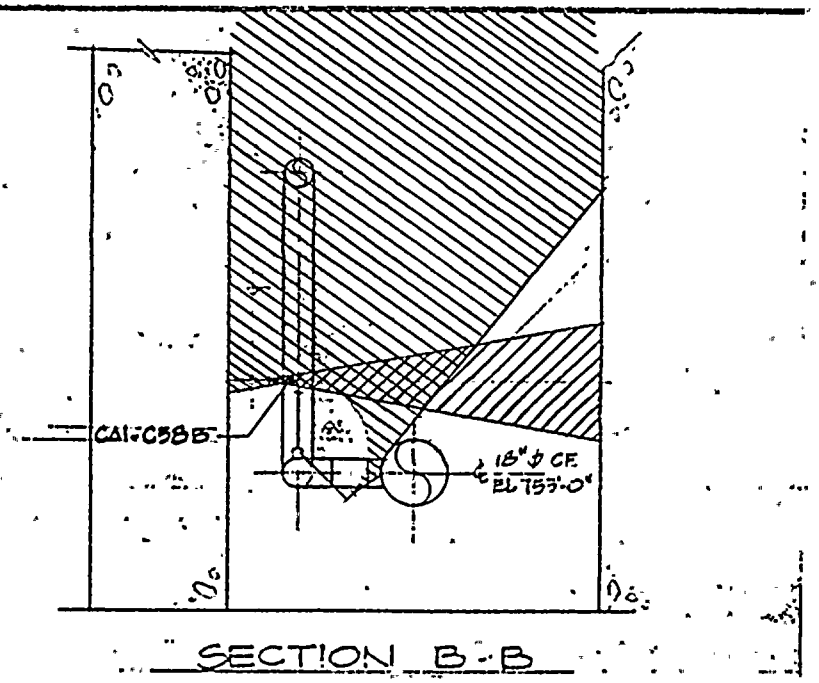


NOTE:
SHADED PORTION INDICATES AREA COVERED BY AN UNRESTRAINED JET SPRAY FOR FOWARD/REVERSE FLOW.
• DENOTES BREAK NUMBERS.

FIGURE : CAO-D-1

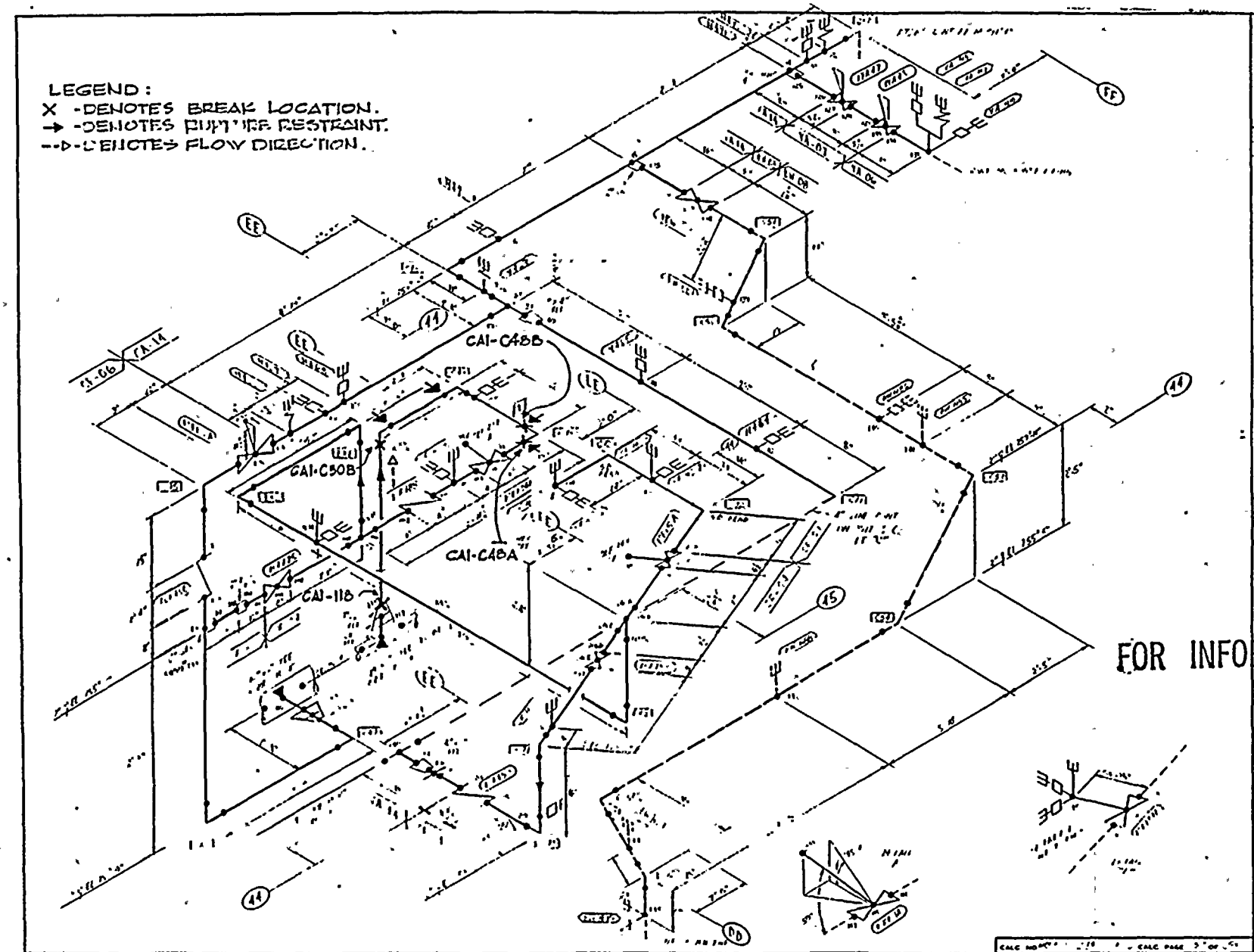






NOTE:
SHADED PORTION INDICATES AREA COVERED BY AN UNRESTRAINED JET SPRAY FOR FOWARD/REVERSE FLOW.
• DENOTES BREAK NUMBER.

FIGURE: CAO-D-2



GLOBAL COORDINATE SYSTEM REFERENCE

GLOBAL COORDINATE ORIGIN: 100.00, 0.01 @

SOURCE DATA SUMMARY

PIPE STRESS MATH MODEL REVISION →

REFERENCE DRAWINGS: PIPING, FLOW, EQUIP & VALVES

NOTES:

FOR INFORMATION ONLY.

REV NO	DESCRIPTION	STRESS ANALYSIS APPROVAL							
		DSN	DATE	PIP	DATE	CKD	DATE	APP	DATE

DUKE POWER COMPANY

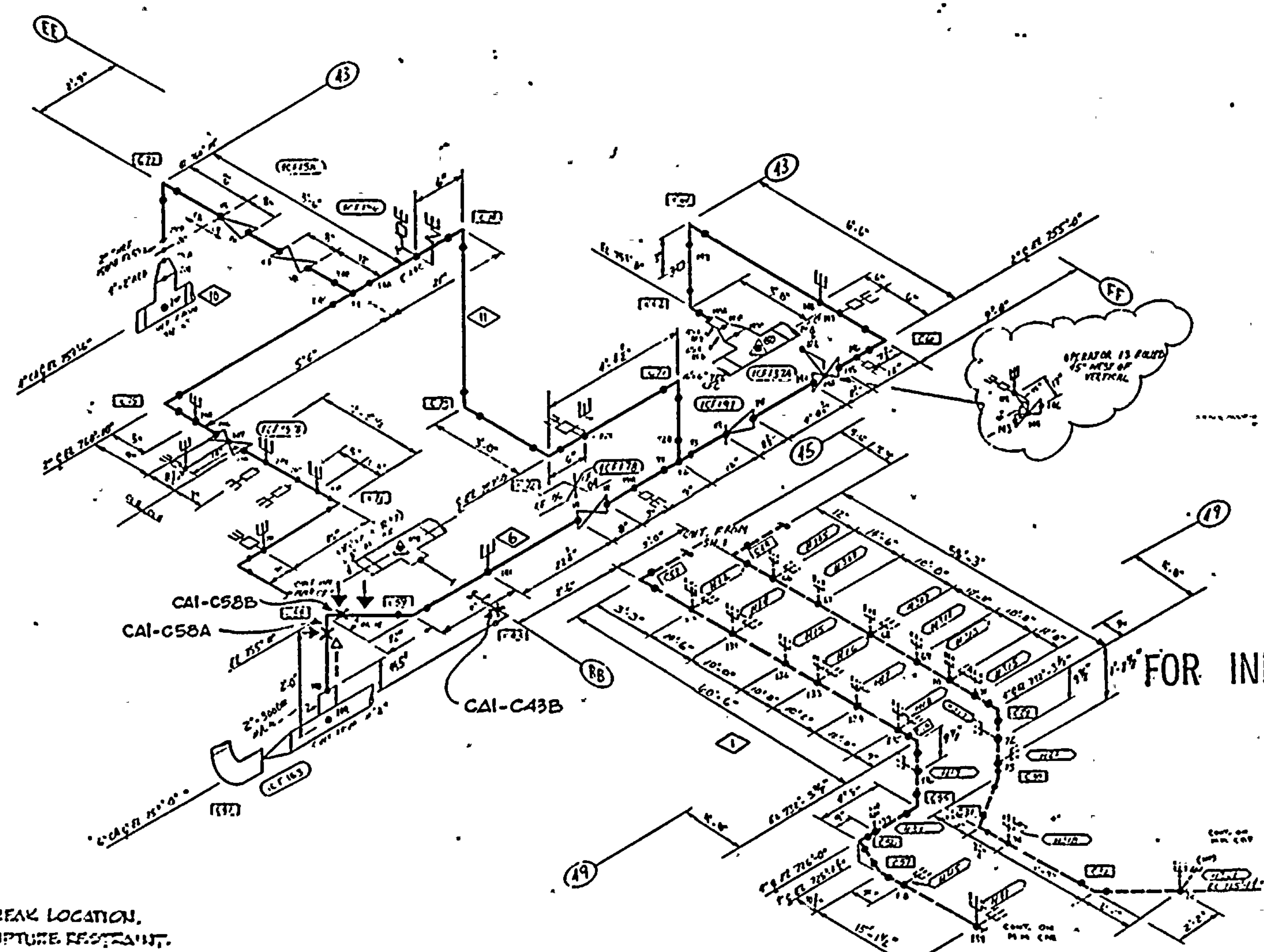
PIPE STRESS ANALYSIS MATH MODEL ISOMETRIC

QA CONDITION: 1

STATION UNIT: McGUIRE / UNIT 1

MATH MODEL: CAP

ISOMETRIC DRAWING SHEET 2 OF 8



FOR INFORMATION ONLY

LEGEND:
X - DENOTES BREAK LOCATION.
→ - DENOTES RUPTURE RESTRAINT.
--▷ - DENOTES FLOW DIRECTION.

CALC. NO MCC-1206.02.71.0005 CALC. PAGE 25B OF 39

GLOBAL COORDINATE SYSTEM REFERENCE

SOURCE DATA SUMMARY

PIPE STRESS MATH MODEL REVISION → 11

PIPING:
MC-1493-04.30-00
-01

FLOW:
MC-1591-1.1

NOTES:
1. ANCHORS AT DCP'S 150 AND 109 ARE FOR ANALYSIS PURPOSES ONLY.

DUKE POWER COMPANY
PIPE STRESS ANALYSIS MATH MODEL ISOMETRIC

QA CONDITION: 1

STATION/UNIT: McGUIRE UNIT 1

MATH MODEL: CAO

ISOMETRIC DRAWING SHEET 3 OF 3