Met-Ed
GPU ELECTRIC POWER COMPANIES
POWER FOR PROGRESS

METROPOLITAN EDISON COMPANY SUBSIDIARY OF GENERAL PUBLIC UTILITIES CORPORATION

POST OFFICE BOX 542 READING, PENNSYLVANIA 19603 TELEPHONE 215 – 929-3601

July 1, 1971

Mr. R. W. Kirkman, Director
Atomic Energy Commission
Division of Compliance
Region I
970 Broad Street
Newark, New Jersey 07102

Subject: D/C letter of April 20, 1971
Three Mile Island Nuclear Station
Unit No. 1
Construction Permit No. CPRH-40

Dear Mr. Kirkman:

Please accept my apologies for the inadvertent delay in responding to your letter of April 20, 1971 concerning the cable pull slips and the 480 volt power centers.

Met-Ed offers the following comments with regard to these two items:

Item I Computer routing. Gilbert Associates Inc. Drawing S-211-002 sh. 2 (an explanation document, not a design drawing) states, "the computer will select the path between the beginning and end trays of the tray system shown on G.A.I. cable tray drawings, and will print the route on the computer circuit routing summary and the pull slips". Thus, the computer is being used as an engineering tool to maintain a fix on the tray loading by GAI as the cables are routed, and is also being used as the basis for the generation of a "pull slip".

This path information does not provide complete terminal to terminal cable routing, therefore, additional information such as conduit, underground duct bank and 6" field tray numbers is added manually in the proper place on the safeguard pull slips by GAI. (All numbers are being assigned by GAI, including those of "field run" trays). By adding this information, GAI is now issuing a safeguard circuit "pull slip" to the field which has complete routing, terminal to terminal. All safeguard pull slips have been returned to GAI for this complete routing.

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Item 1
Cont'd. GAI Inc. Drawing S-211-002 sheet 2 also refers to "Forced or Hand Routed Circuits". These are circuits which are entirely manually routed. The routing data are then fed into the computer in order to be included in the total tray loading, and a "pull slip" is generated reflecting these data. Again if any additional information is required, it will be handled as previously outlined.

Item II The 480 volt power centers cited have been placed in accordance with the engineers design specification. GAI has located all redundant equipment with adequate separation as a design criterion. Met-Ed has reviewed with GAI the location of the cited equipment and agrees that the separation is adequate.

Due to the nature of certain types of equipment and/or its required location, the type and amount of physical separation will vary. Certain electrical equipment is separated by a fireproof barrier because this is an insurance requirement for partitions in this building. If, during construction, a modification or additional protection of redundant equipment is required, the criteria for physical separation would be re-evaluated by Met-Ed and the Engineer and necessary changes would be made. For example: Met-Ed and GAI are presently evaluating the presence of water lines in the vicinity of the 480 volt power centers cited as a potential cause of common mode failure of this equipment to determine whether additional protection is required. If this is the case, it will be provided.

Yours very truly,

J. G. Miller

J. G. Miller
Vice President

kg

CC: Mr. J. H. Tillou

Met Ed
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METROPOLITAN EDISON COMPANY SUBSIDIARY OF GENERAL PUBLIC UTILITIES CORPORATION
POST OFFICE BOX 542 READING, PENNSYLVANIA 19603 TELEPHONE 215 – 929-3601

September 28, 1972

Mr. J. P. O'Reilly
Director, Region 1
Directorate of Regulatory Operations
U. S. Atomic Energy Commission
970 Broad Street
Newark, New Jersey 07102

Subject: Three Mile Island Nuclear Station Resolution of AEC Findings Regarding Document Control and Storage of Non-conforming Material

Dear Mr. O'Reilly:

Your letter of August 29, 1972, refers to inspections performed by AEC-DRO personnel at the TMI site during July 11 to 14, 1972. Your letter indicates that there were two areas noted where activities at the TMI site were in apparent nonconformance with the AEC quality assurance criteria in Appendix B to 10 CFR 50. These apparent nonconformances, and our resolution of them, are as follows:

1. Finding

A random check of drawings in the field indicated that 31% of the drawings were superseded and 26% were illegible as to title, revision number, or drawing number, and their status could not be determined.

Resolution

Earlier this Spring, our own quality assurance surveillance system had alerted us to the need to upgrade the site drawing control system, and we had initiated corrective action with United Engineers and Constructors (UE&C). Our review of drawings in the field has indicated that many of the out-of-date or illegible drawings were being retained for informational purposes rather than for control work. We recognize that this should have been indicated on the

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drawings. We have essentially completed a thorough upgrading of the drawing control system at the TMI site to ensure that all drawings used to control work are up-to-date and all informational drawings are clearly identified as not being for construction. This upgrading included:

a. Detailed auditing of documentation control at the TMI site was performed to identify the areas requiring correction.

b. Revisions and clarifications have been issued to the UE&C procedures for drawing control and for control of QC procedures.

c. The Gilbert Associates and Burns & Roe master drawing lists are being clarified to facilitate drawing control.

d. Each UE&C construction organization has thoroughly reviewed their drawings to assure that only up-to-date drawings are in use.

e. UE&C has re-emphasized to their construction supervision the need to follow document control procedures. In this regard, the responsibilities for keeping documents up-to-date in each construction department have been clarified and documented in the applicable UE&C procedures. Additional document control personnel have been assigned to this work.

f. Re-audits have been and will again be performed to assure that the upgraded document control system is effective.

2. Finding

The designated quarantine storage area at the site contained items which were not in a hold status. In addition, it was noted that tagged nonconforming items were in storage outside of the quarantine area.

Resolution

The applicable UE&C storage procedure has been revised to emphasize that only "Reject" items shall be stored in the "Reject" storage area. The procedure has also been revised to emphasize that, while items on "Hold" are not given segregated storage, their storage locations are logged and they may not be released from storage except on a waiver basis as permitted by the procedure for control of nonconforming material.

In regard to storing all reject items in a segregated "Reject" storage area, this is sometimes not practical, e.g., for large vessels. The applicable UE&C procedure requires use of segregated storage when practicable.

We consider that the above actions satisfactorily resolve the findings in your letter of August 29, 1972.

Very truly yours,

J. G. Miller
Vice President

Distribution List: Mr. B. G. Avers
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Mr. M. J. Stromberg

THREE MILE ISLAND NUCLEAR STATION

UNIT 1

GENERAL INFORMATION ON THE EFFECTS OF FLUID FLOW INSTABILITY IN THE MAIN AND EMERGENCY FEEDWATER SYSTEMS OF THREE MILE ISLAND NUCLEAR STATION UNIT 1

METROPOLITAN EDISON COMPANY

SUBSIDIARY OF GENERAL PUBLIC

UTILITIES CORPORATION

Gilbert Commonwealth Companies

Gilbert Associates, Inc.

engineers and consultants Reading, Pennsylvania

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25 pp.

SEPTEMBER, 1975 GAI REPORT NO. 1881

GENERAL INFORMATION ON THE EFFECTS OF FLUID FLOW INSTABILITY IN THE MAIN AND EMERGENCY FEEDWATER SYSTEMS OF THREE MILE ISLAND NUCLEAR STATION UNIT 1

METROPOLITAN EDISON COMPANY

Subsidiary of General Public Utilities Corporation

Prepared by: R.A. Snow

R.A. Snow, PE, Piping Engineer, Piping Engineering Dept.

Reviewed and Approved by:

A.P. Rochino, PE, PhD, Supervisory Engineer, Piping Engineering Dept.

Gilbert Associates, Inc.
525 Lancaster Avenue
Reading, Pennsylvania

JWF

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FIGURES

1.0 INTRODUCTION

This report presents the results of a design review and evaluation performed in response to the United States Nuclear Regulatory Commission (NRC) request, "General Information Required for Consideration of the Effects of Secondary System Fluid Flow Instability," dated May 15, 1975, for Three Mile Island Nuclear Station, Unit 1 (TMI-1).

Included are complete responses to specific NRC questions (Reference 1) considering design and reference data requested by the NRC. Most of these responses are based heavily upon information provided in Reference 2.

2.0 NUCLEAR REGULATORY COMMISSION QUESTIONS AND CONSEQUENT RESPONSES

Question No. 1

Describe all operating occurrences that could cause the level of the water/steam interface in the steam generator to drop below the feedwater sparger or inlet nozzles, and allow steam to enter the sparger and/or feedwater piping.

Response

The design of the Babcock and Wilcox (B&W) once through steam generator (OTSG) requires that the level of the water/steam interface remain below the feedwater inlet nozzles during operation. However, the arrangement of the feedwater nozzles, external ring distribution header and feedwater piping leading up to the header is such that steam cannot enter the feedwater piping. The piping immediately outside the steam generator contains a "gooseneck" or trap arrangement which is always filled with water. This precludes any steam from entering the feedwater piping.

At TMI-1 there is a small one-inch bypass line around the feedwater control valves which maintains a continuous flow rate to the steam generators prior to power operation. This small flow rate (which begins when the unit is cold) will, in itself, keep the feedwater lines leading up to the steam generator full of water and preclude steam from entering the pipe. Once power operation begins, the normal feedwater flow rate fills the pipes, feedwater distribution header and nozzles with water and they remain filled throughout power operation.

Question No. 2

Describe and show by isometric diagrams, the routing of the main and auxiliary feedwater piping from the steam generators outwards through containment up to the outer containment isolation valve and restraint. Note all valves and provide the elevations of the sparger and/or inlet nozzles and all piping runs needed to perform an independent analysis of drainage characteristics.

Response

The routing of the main and emergency feedwater piping from the steam generators to the anchors past the outer containment isolation valves is discussed in Sections 3.0 and 4.0. The isometric diagrams referenced in the discussions contain information relative to valve locations, elevations of spargers or inlet nozzles and the elevations of the different parts of the piping runs.

It should be noted that the main and emergency feedwater systems are completely separated.

Question No. 3

Describe any "water hammer" experiences that have occurred in the feedwater system and the means by which the problem was permanently corrected.

Response

No "water hammer" experiences have occurred in the main or emergency feedwater systems at TMI-1 (Reference 4).

Question No. 4

Describe all analyses of the feedwater and auxiliary systems for which dynamic forcing functions were assumed. Also, provide the results of any test programs that were carried out to verify that either uncovering of the feedwater lines could not occur at your facility or if it did occur, that "water hammer" would not occur.

a. If forcing functions were assumed in your analysis, provide the technical bases that were used to assure that an appropriate choice was made and that adequate conservatisms were included in the analytical model.

b. If a test program was followed, provide the basis for assuring that the program adequately tracked and predicted the flow instability event that occurred, and further, that the test results contained adequate conservatisms and an acceptable factor of safety, e.g., range of parameters covered all conceivable modes of operation.

c. If neither a nor b have been performed, present your basis for not requiring either and your plans to investigate this potential transient occurrence.

Response

Items a and b, above, have not been performed nor are future investigations concerning the potential occurrence of this transient planned for the following reasons:

a. No "water hammer" phenomena have occurred in TMI-1 main or emergency feedwater lines from the steam generator inlet nozzles or spargers to the pipe anchors past the outer containment isolation valves.

b. Similar fluid flow transients have not been observed during operations at Oconee Unit 1, a nuclear unit similar to TMI-1.

c. System design and complete separation of main and emergency feedwater piping preclude the possibility of steam entering the feedwater lines. Consequently, the flow instability that generates the "water hammer" transient will not occur.

d. A one-inch bypass line is provided around each feedwater control valve which maintains a continuous flow rate to the steam generators at all power levels. This ensures that the feedwater piping is filled at all times.

Question No. 5

Discuss the possibility of a sparger or nozzle uncovering and the consequent pressure wave effects that could occur in the piping following a design basis loss-of-coolant accident, assuming concurrent trip and loss of offsite power.

Response

The steam generator water level is below the feedwater inlet nozzles during power operation. However, steam will not enter the feedwater piping and no pressure wave effects will occur in the piping following a design basis accident with concurrent turbine trip and loss of offsite power due to the "gooseneck" arrangement of the feedwater piping directly outside the steam generator. Even when the main feedwater pumps trip and feedwater flow rate is zero, the trap remains full of water and precludes the possibility of any steam entering the piping.

A test was run at Oconee Unit 1 (a unit similar to TMI-1) from 40 percent power during which the main feedwater pumps and the turbine generator were tripped and auxiliary feedwater flow was initiated. The auxiliary feedwater system on the B&W OTSG (including distribution header and piping) is completely separated from the main feedwater system. The auxiliary feedwater enters the unit through a separate header at the top of the tube section. This test very closely simulates the effects of a loss of offsite power on the secondary plant. The steam generator and feedwater piping directly adjacent to the steam generator were monitored for noises using the B&W loose parts monitoring system. No unusual noises were heard, confirming the fact that no "water hammer" in the feedwater piping occurred during the test.

Question No. 6

If plant system design changes have been or are planned to be made to preclude the occurrence of flow instabilities, describe these changes or modifications, and discuss the reasons that made this alternative superior to other alternatives that might have been applied. Discuss the quality assurance program that was or will be followed to assure that the planned system modifications will have been correctly accomplished at the facility. If changes are indicated to be necessary for your plant, consider and discuss the effects of reduced auxiliary feedwater flow as a possible means of reducing the magnitude of induced pressure waves, including positive means (e.g., interlocks) to assure sufficiently low flow rates and still meet the minimum requirements for the system safety function.

Response

It has been shown in the preceding responses that system design already precludes the possibility of the occurrence of "water hammer". Therefore, no system design changes are planned for further protection against this occurrence. Furthermore, TMI-1 is provided with a one-inch bypass line around the feedwater control valves which maintains a continuous low flow rate to the steam generators at all power levels. This feature, in conjunction with other inherent system designs, precludes the possibility of the occurrence of the "water hammer" phenomenon.

3.0 MAIN AND EMERGENCY FEEDWATER TO STEAM GENERATOR 1A

Figure 3-1 is an isometric representation of those portions of the main feedwater (FW) and emergency feedwater (EFW) systems associated with steam generator 1A. These systems are completely separated from each other. All valves are noted and the height above (+) or below (-) the datum lines for the different parts of the piping runs are indicated. The datum lines are the elevations of the injection nozzles which are as follows:

a. For FW 324 ft-7-11/16 in

b. For EFW 337 ft-5-3/8 in

Since the two systems are independent of each other, the indicated elevations for parts of the piping runs of each system are referenced to the elevations of the associated steam generator injection nozzles.

The information provided by Figure 3-1 is sufficient to perform an independent analysis of the drainage characteristics of the entire feedwater system associated with steam generator 1A. This isometric diagram depicts the piping runs for each system (FW and EFW) from the steam generator inlet nozzles to the pipe anchor past the containment isolation valves outside the reactor building.

Figures 3-2 and 3-3 show the details inside and outside the reactor building, respectively, of the FW and EFW system piping. In the following discussion the height, or elevation, of each part of the piping run with respect to the datum lines will be indicated within parentheses using the sign convention noted, i.e., (+) for above and (-) for below the datum line elevations.

3.1 MAIN FEEDWATER SYSTEM

The FW system piping enters steam generator 1A through 32 injection nozzles at elevation 324 ft-7-11/16 in. Each of the 32 injection nozzles is connected by a 3 ft-9 in long, vertical section of 3 in pipe down to a 14 in diameter horizontal ring header (-3 ft-9 in).

The ring header consists of two half rings as shown by Figure 3-2. Each half ring connects to a 14 in supply pipe which descends 4 ft-9 in (-8 ft-6 in) from the centerline of the ring. The two sections of 14 in pipe run horizontally for 19 ft-4-1/2 in and 20 ft-6 in, respectively, and then connect through 20x14 in reducers to a common 20x20x20 in tee. The 20 in diameter feedwater supply pipe then rises 40 ft-10-5/16 in (+32 ft-4-5/16 in), runs horizontally for 29 ft-1 in (+32 ft-4-5/16 in), descends 28 ft-0 in (+4 ft-4-5/16 in) and runs horizontally for 17 ft-9 in (+4 ft-4-5/16 in) as shown by Figure 3-2.

The FW line is anchored to the reactor building at penetration no. 227.

In summary, there are 145 feet of FW piping inside the reactor building and there are no valves in the line. There are, however, 8 rupture restraints, 3 hydraulic snubbers and 7 spring and constant support hangers.

Figure 3-3 shows the FW piping run outside of the reactor building. The 20 in pipe runs horizontally from penetration no. 227 for 16 ft-10 in (+4 ft-4-5/16 in), descends 5 ft-0 in (-0 ft-7-11/16 in) and runs horizontally for 2 ft-6 in (-0 ft-7-11/16 in) before passing through an isolation check valve, FWV-12A.

The 20 in pipe continues to run horizontally for 12 ft-4 in (-0 ft-7-11/16 in) to the electric motor operated (EMO) 20 in feedwater control valve, FWV-17A. The FW line then rises 22 ft-0 in (+21 ft-4-5/16 in). The main feedwater isolation valve, FWV-5A, is located in this vertical piping run. A 6 in bypass line around the feedwater control valve, FWV-17A, and the main feedwater isolation valve, FWV-5A, is installed to facilitate control of low and intermediate FW flow. This bypass line is provided with two control valves, FWV-92A and FWV-16A.

The feedwater control valve, FWV-17A, is provided with a 1 in bypass line to ensure continuous feedwater flow during all modes of FW system operation.

The remainder of the 20 in FW line runs horizontally for 105 ft-10 in (+21 ft-4-5/16 in) before descending 21 ft-9 in (-0 ft-4-11/16 in) and then runs horizontally for 4 ft-9-1/2 in (-0 ft-4-11/16 in) to a pipe anchor.

3.2 EMERGENCY FEEDWATER SYSTEM

As shown in Figure 3-2, the flow from the EFW system enters steam generator 1A through seven injection nozzles at an elevation of 337 ft-5-3/8 in (the EFW datum elevation), 12 ft-9-11/16 in above the FW injection nozzles. Each of the seven injection nozzles connects to a 3 in pipe which descends 1 ft-8 in and joins a 6 in diameter, horizontal, 300 degree ring header (-1 ft-8 in). This ring header is supplied by a single 6 in pipe which descends 5 ft-9-3/8 in (-7 ft-5-3/8 in) from the ring centerline. This 6 in pipe then runs horizontally for 6 ft-3 in (-7 ft-5-3/8 in), rises 6 ft-0 in

(-1 ft-5-3/8 in), runs horizontally for 34 ft-6 in (-1 ft-5-3/8 in), descends 4 ft-0 in (-5 ft-5-3/8 in), runs horizontally for 60 ft-2 in (-5 ft-5-3/8 in), descends 14 ft-0 in (-19 ft-5-3/8 in), runs horizontally for 18 ft-0 in (-19 ft-5-3/8 in) and is anchored to the reactor building at penetration no. 110.

In summary, there is 150 feet of EFW piping inside the reactor building and there are no valves in the line. There are, however, 2 guides, 8 hydraulic snubbers, 5 sway struts, 8 rupture restraints and 9 spring hangers and constant load supports.

Outside the reactor building, as shown by Figure 3-3, the 6 in EFW pipe runs horizontally from penetration no. 110 for 13 ft-10 in (-19 ft-5-3/8 in) passing through check valve EFV-12A. The pipe then descends 10 ft-0 in (-29 ft-5-3/8 in), runs horizontally for 34 ft-11 in (-29 ft-5-3/8 in), descends 9 ft-8-9/16 in (-39 ft-1-15/16 in) and runs horizontally to the EFW control valve, EFV-30A. After the control valve, the pipe descends 2 ft-3-7/16 in and runs horizontally 8 ft-0 in to a pipe anchor.

4.0 MAIN AND EMERGENCY FEEDWATER TO STEAM GENERATOR 1B

Figure 4-1 is an isometric representation of those portions of the main feedwater (FW) and emergency feedwater (EFW) systems associated with steam generator 1B. These systems are completely separated from each other. Presentation of the heights of various parts of the piping runs follows the conventions discussed in Section 3.0. The datum line elevations (injection nozzle elevations) are as follows:

a. For FW 324 ft-7-11/16 in

b. For EFW 337 ft-5-3/8 in

Figures 4-2 and 4-3 show the details, inside and outside the reactor building, respectively, of the FW and EFW piping.

4.1 MAIN FEEDWATER SYSTEM

The FW system piping enters steam generator 1B through 32 injection nozzles at elevation 324 ft-7-11/16 in. Each of the 32 injection nozzles is connected by a 3 ft-9 in long, vertical section of 3 in pipe down to a 14 in diameter horizontal ring header (-3 ft-9 in).

The ring header consists of two half rings, each of which is connected to a 14 in supply pipe which descends 4 ft-9 in (-8 ft-6 in) from the centerline of the ring. The two sections of 14 in pipe run horizontally for 14 ft-11 in and 17 ft-11 in, respectively, and then connect through 20x14 in reducers to a common 20x20x20 in tee. The 20 in diameter feedwater supply pipe then rises 14 ft-10-5/16 in (+6 ft-4-5/16 in), runs horizontally for 9 ft-9 in (+6 ft-4-5/16 in), descends 14 ft-10-5/16 in (-8 ft-6 in), runs horizontally for 58 ft-4 in (-8 ft-6 in), descends 13 ft-5-11/16 in (-21 ft-11-11/16 in), runs horizontally for 46 ft-5 in (-21 ft-11-11/16 in), rises 13 ft-5-11/16 in (-8 ft-6 in) and runs horizontally for 5 ft-8-3/4 in to an anchor at reactor building penetration no. 103 as shown by Figure 4-2.

In summary, there are 186 feet of FW piping inside the reactor building and there are no valves in the line. There are, however, 9 rupture restraints, 5 hydraulic snubbers and 13 spring and constant support hangers.

Figure 4-3 shows the FW piping outside of the reactor building. The 20 in pipe runs horizontally from penetration no. 103 for 21 ft-9-11/16 in (-8 ft-6 in) and rises vertically 7 ft-10-5/16 in (-0 ft-7-11/16 in) to an isolation check valve FWV-12B.

The 20 in pipe then runs horizontally for 10 ft-7-3/8 in (-0 ft-7-11/16 in) to the electric motor operated (EMO) 20 in feedwater control valve, FWV-17B. The FW line then rises 22 ft-0 in (+21 ft-4-5/16 in). The main feedwater isolation valve, FWV-5B is located in this vertical piping run. A 6 in bypass line around the feedwater control valves, FWV-17B, and the main feedwater isolation valve, FWV-5B, is installed to facilitate control of low and intermediate FW flow. This bypass line is provided with two control valves, FWV-92B and FWV-16B.

The feedwater control valve, FWV-17B, is provided with a 1 in bypass line to ensure continuous feedwater flow during all modes of FW system operation.

The remainder of the 20 in FW line runs horizontally for 52 ft-9 in (+21 ft-4-5/16 in), descends 21 ft-9 in (-0 ft-4-11/16 in) and runs horizontally for 7 ft-9-1/2 in (-0 ft-4-11/16 in) to an anchor.

4.2 EMERGENCY FEEDWATER SYSTEM

As shown in Figure 4-2, the flow from the EFW system enters steam generator 1B through seven injection nozzles at an elevation of 337 ft-5-3/8 in (the EFW datum elevation), 12 ft-9-11/16 in above the FW injection nozzles. Each of the seven injection nozzles connects to a 3 in pipe which descends 1 ft-8 in and joins a 6 in diameter, horizontal, 300 degree ring header (-1 ft-8 in). This ring header is supplied by a single 6 in pipe which descends 5 ft-9-3/8 in (-7 ft-5-3/8 in), runs horizontally for 6 ft-3 in (-7 ft-5-3/8 in), rises 6 ft-0 in (-1 ft-5-3/8 in), runs horizontally for 11 ft-9-7/16 in (-1 ft-5-3/8 in), descends 4 ft-3 in (-5 ft-8-3/8 in), runs horizontally for 92 ft-3 in (-5 ft-8-3/8 in), descends 13 ft-9 in (-19 ft-5-3/8 in), runs horizontally for 40 ft-6 in (-19 ft-5-3/8 in) and is anchored to the reactor building at penetration no. 111.

In summary, there is 156 feet of EFW piping inside the reactor building and there are no valves in the line. There are, however, 3 guides, 2 hydraulic snubbers, 9 sway struts, 9 rupture restraints and 11 spring, constant load and rigid hangers.

Outside the reactor building, as shown by Figure 4-3, the 6 in EFW pipe runs horizontally from penetration no. 111 for 14 ft-0 in (-19 ft-5-3/8 in) passing through check valve, EFV-12B. The pipe then descends 19 ft-8-9/16 in (-39 ft-1-15/16 in), runs horizontally for 9 ft-3 in (-39 ft-1-15/16 in) to the EFW control valve, EFV-30B, descends 2 ft-3-9/16 in (-41 ft-5-1/2 in) and runs horizontally for 2 ft-3 in (-41 ft-5-1/2 in) to a pipe anchor.

5.0 REFERENCES

1. United States Nuclear Regulatory Commission letter and attachment to Metropolitan Edison Company, Docket No. 50-289; requesting general information; May 15, 1975.
2. Babcock and Wilcox, "Prevention of Waterhammer", presentation meeting with the United States Nuclear Regulatory Commission; April 1, 1975.
3. Babcock and Wilcox letter to Metropolitan Edison Company from Mr. J. D. Phinney to Mr. R. M. Klingaman; June 30, 1975.
4. Telecon memo, Mr. A. P. Rochino of Gilbert Associates, Inc., to Mr. G. Kunder of Three Mile Island Nuclear Station Unit 1, Operations; August 19, 1975.

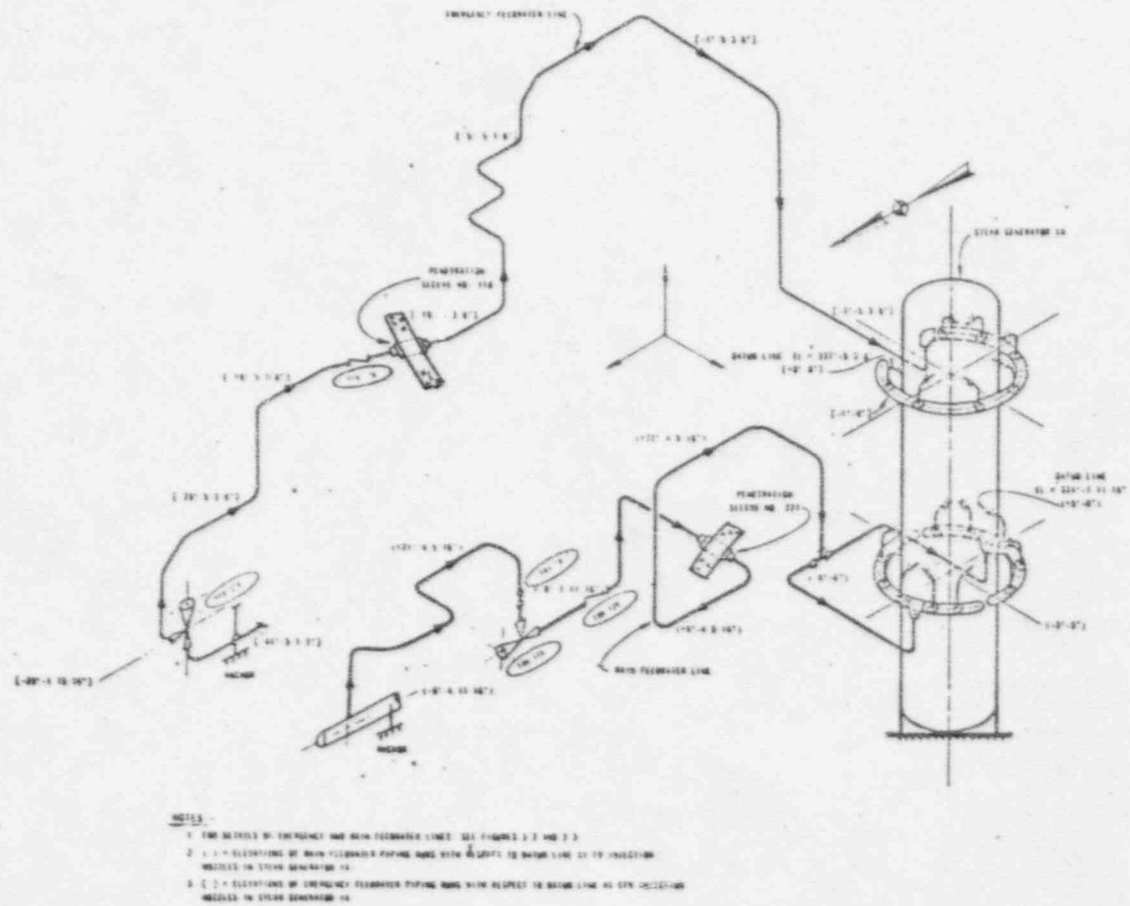


FIGURE 3-1

MAIN AND EMERGENCY FEEDWATER ELEVATIONS FROM STEAM GENERATOR 1A TO ANCHOR PAST CONTAINMENT ISOLATION VALVES

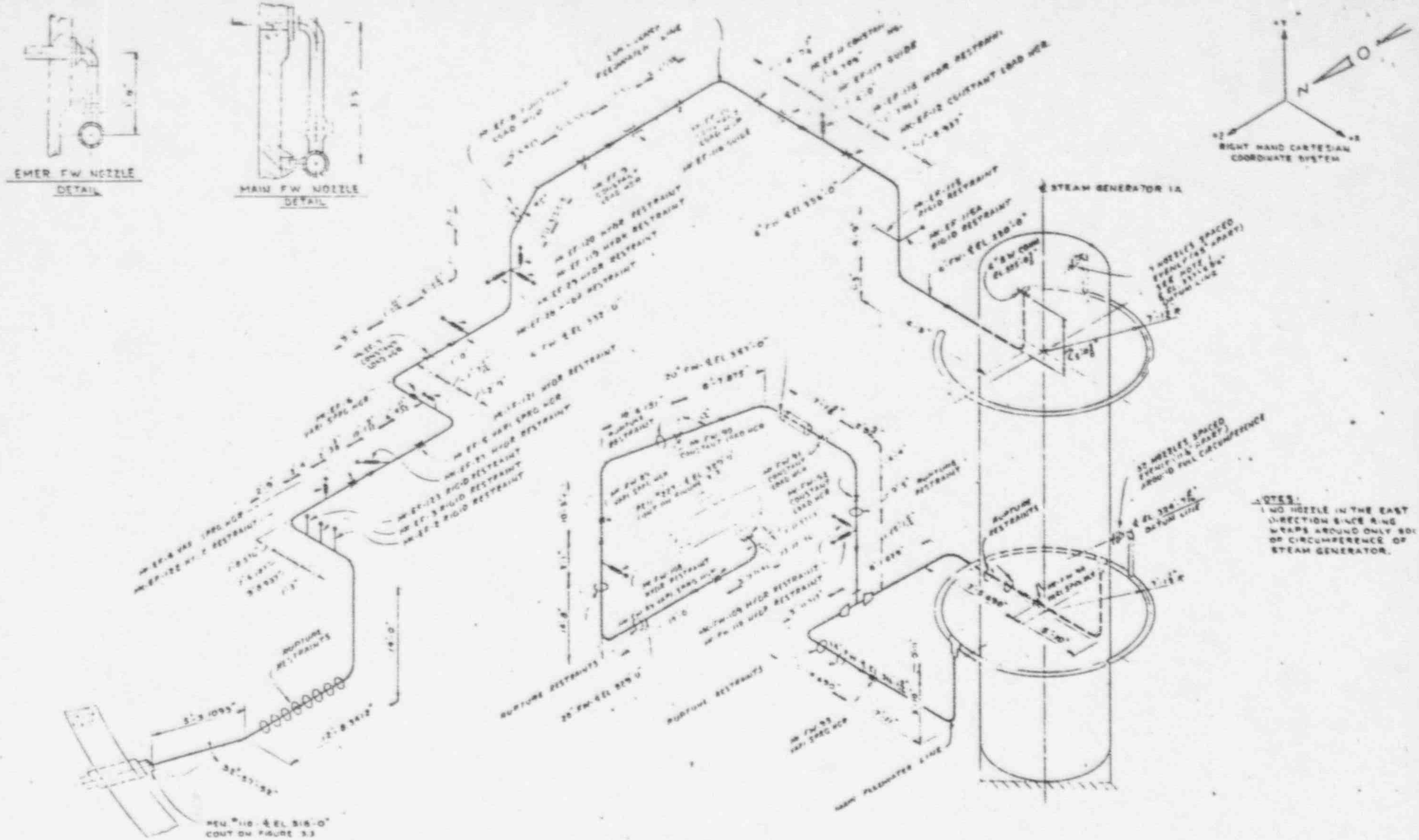


FIGURE 3-2

MAIN AND EMERGENCY FEEDWATER DETAILS FROM STEAM GENERATOR 1A TO REACTOR BUILDING PENETRATIONS NO. 227 AND NO. 110

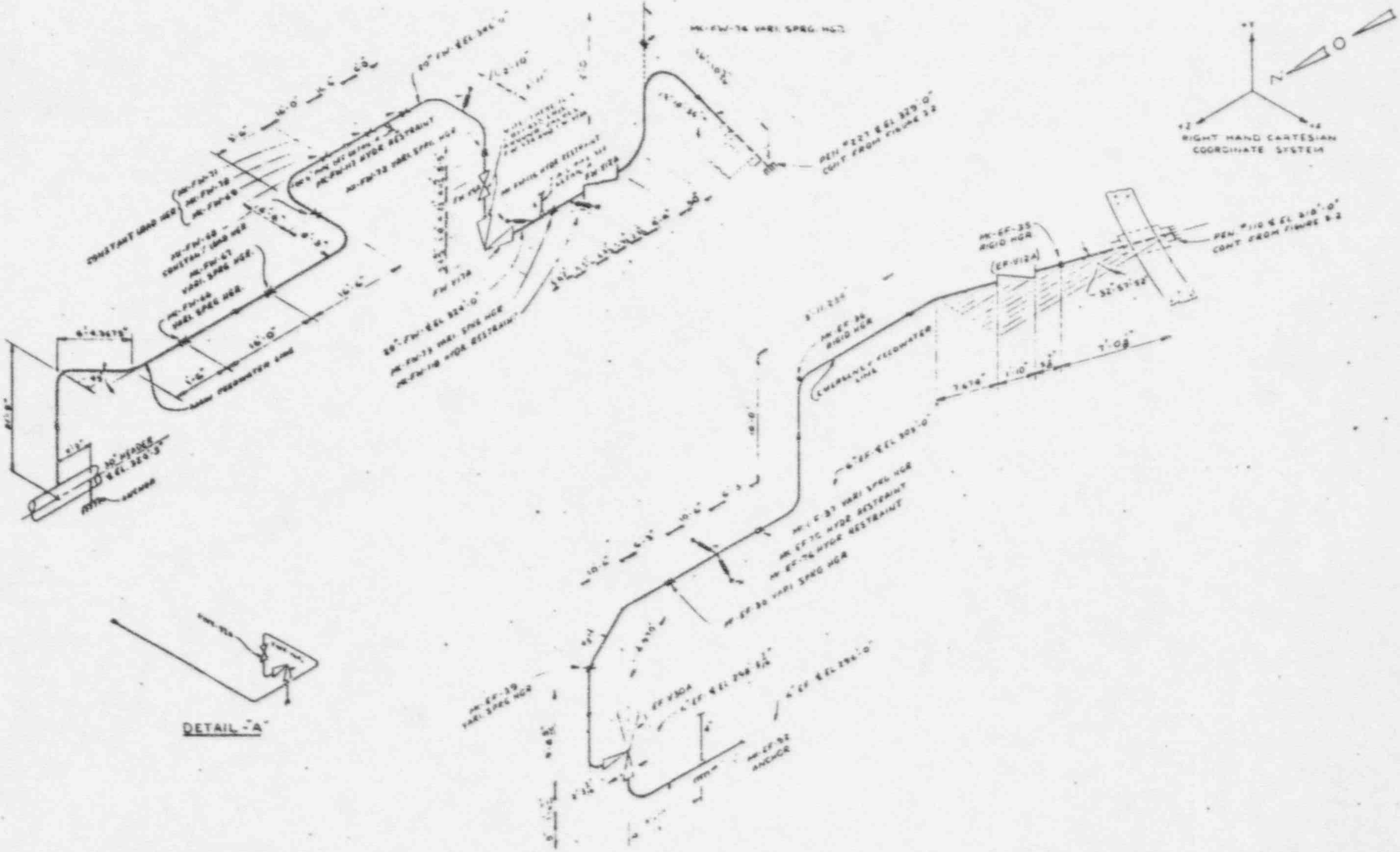
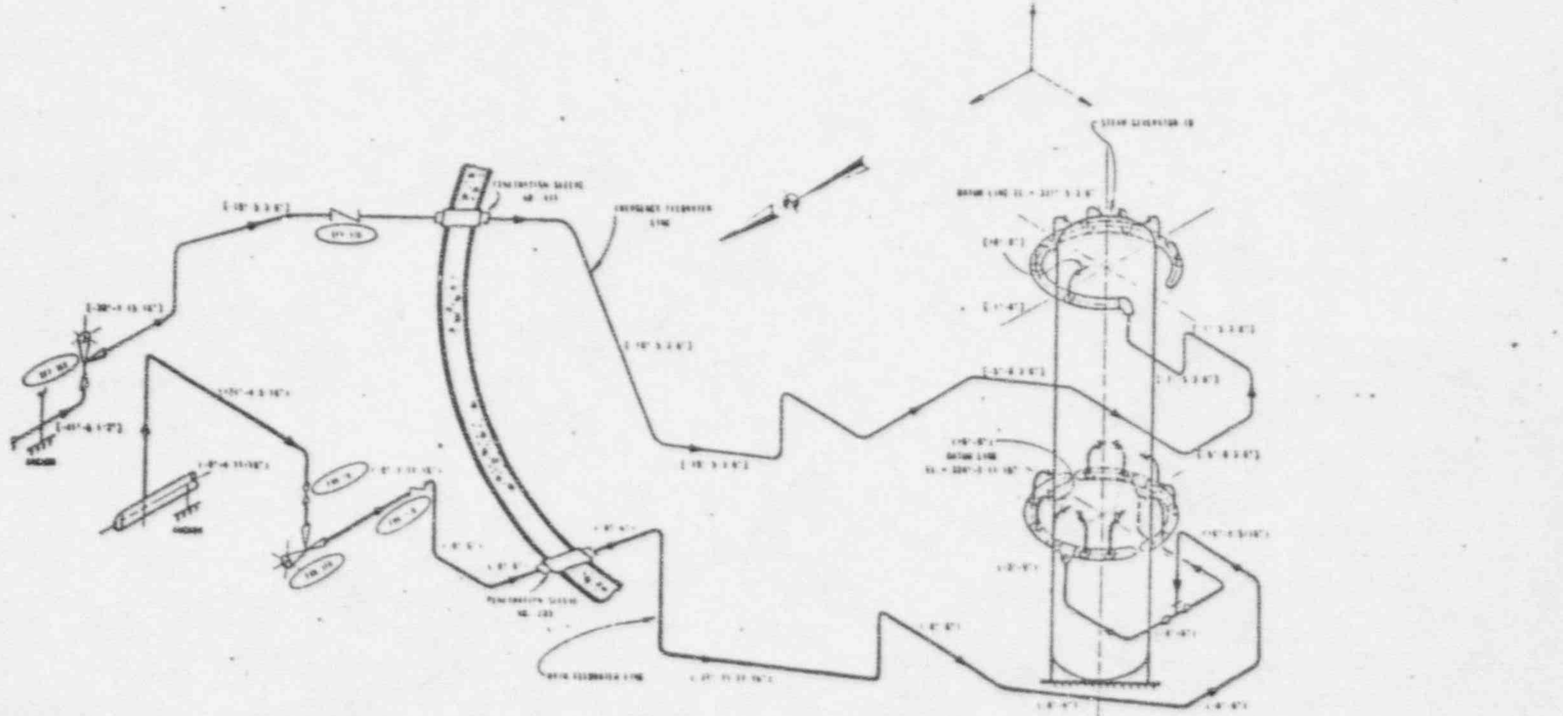


FIGURE 3-3

MAIN AND EMERGENCY FEEDWATER DETAILS FROM REACTO BUILDING PENETRATIONS NO. 227 AND NO. 110 TO ANCHORS PAST CONTAINMENT ISOLATION VALVES



NOTES

FIGURE 4-1

MAIN AND EMERGENCY FEEDWATER ELEVATIONS FROM STEAM GENERATOR 1B TO ANCHORS PAST CONTAINMENT ISOLATION VALVES

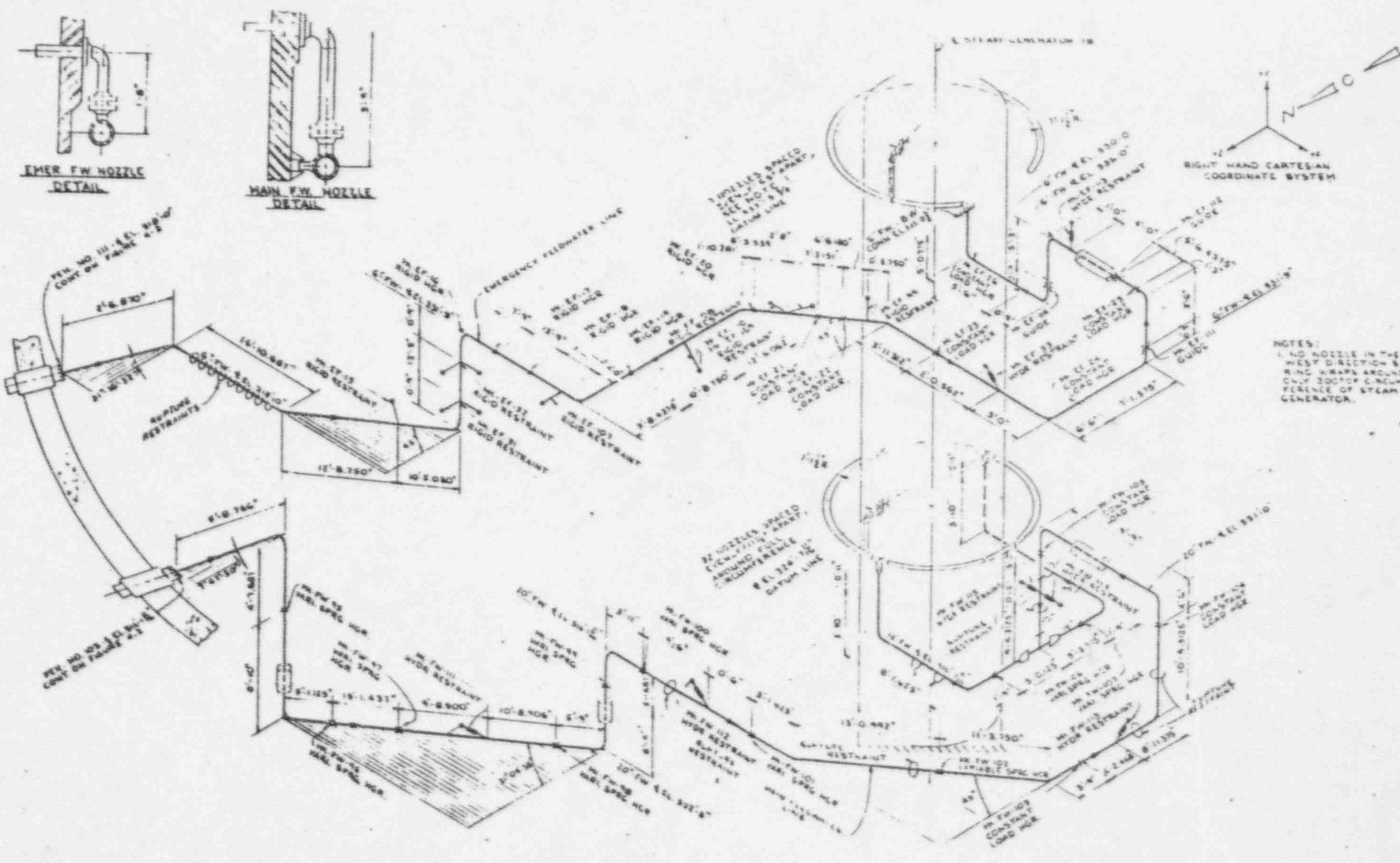


FIGURE 4-2

MAIN AND EMERGENCY FEEDWATER DETAILS FROM STEAM GENERATOR 1B TO REACTOR BUILDING PENETRATIONS NO. 103 AND NO. 111

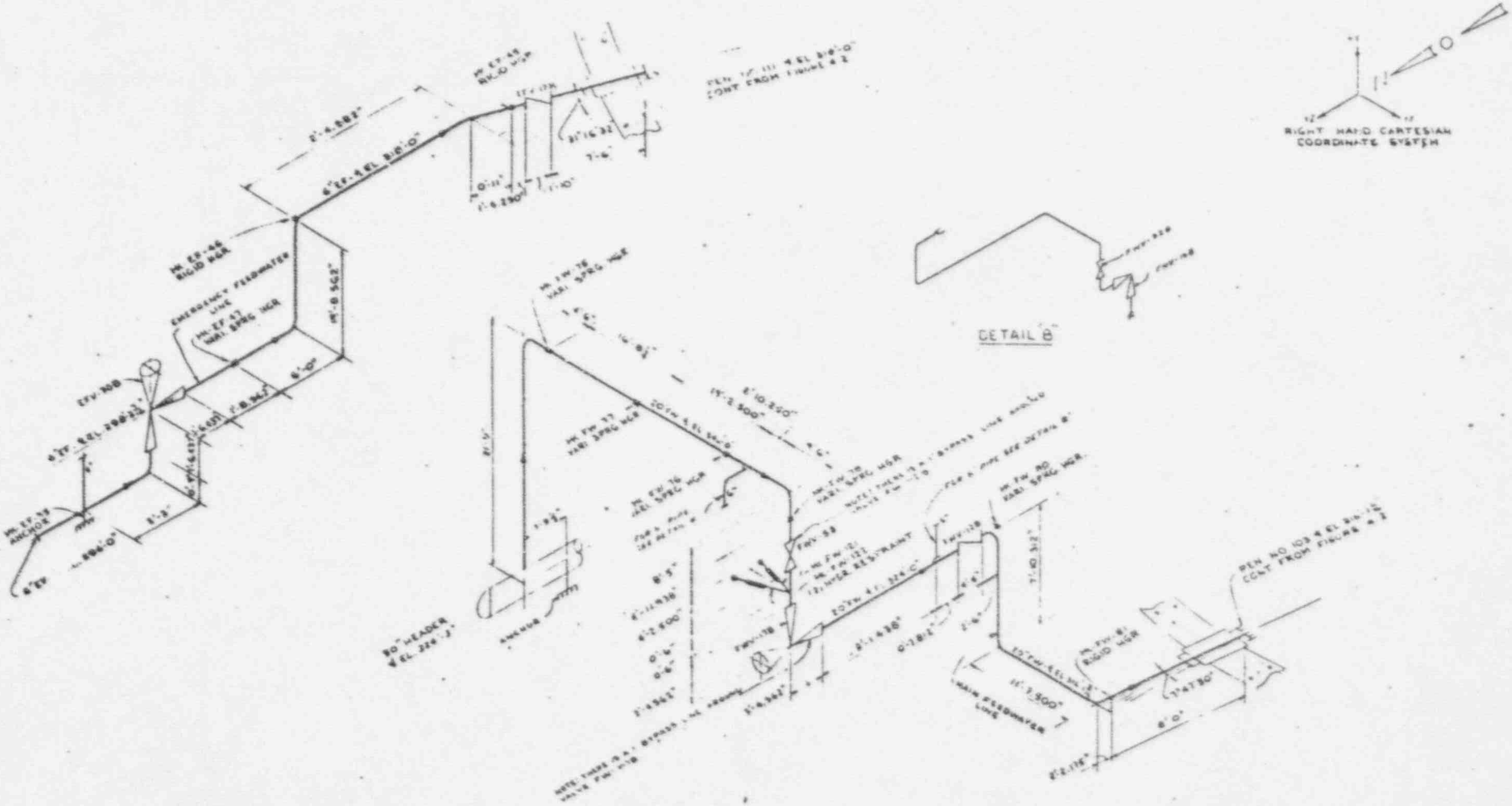


FIGURE 4-3

MAIN AND EMERGENCY FEEDWATER DETAILS FROM REACTC BUILDING PENETRATIONS NO. 103 AND NO. 111 TO ANCHORS PAST CONTAINMENT ISOLATION VALVES