AUXILIARY FEEDWATER FLOW

RATE FOLLOWING A

LOSS OF MAIN

FEEDWATER

(177 FA PLANTS)

86-1102587-00

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Prepared By [signature] 5/22/79

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1.0 PURPOSE

Several studies have been performed (References 1 and 2) for the B&W 177 FA plants on a loss of main feedwater transient. The studies to date have looked at normal feedwater actuation times and delayed auxiliary feedwater actuation times. The purpose of this sensitivity study is to evaluate the response of 177 FA plants to minimum and maximum AFW flow rates following a LOFW event from 100% full power. This study then defines the minimum auxiliary feedwater flow rate that can be initiated to one or both OTSG's and return the plant to a stabilized cooling mode without filling the pressurizer, reaching saturation conditions in the primary side, or lifting the PORV following a LOFW event.

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2.0 ANALYSIS ASSUMPTIONS

A survey of B&W 177 Fuel Assembly plants shows that the auxiliary feedwater flow rates for the 2772 Mwt units can vary from 500 gpm (one AFW pump) to 1600 gpm (all AFW pumps). The smallest capacity at any of the B&W 177 FA plants is a one AFW pump flow of 370 gpm at a 2568 Mwt plant.

The computer code used for this analysis is the CADDS code (See Section 3.2.1.1 of Reference 1). The model and analysis assumptions of Reference 1, Sections 3.2.1.2, 4.2, 4.3, and 4.4 are used for this analysis. Volume III of Reference 1 verifies that this analysis is applicable to raised as well as lowered loop plants.

CADDS is a single loop model and therefore, the case where the minimum AFW flow is to one OTSG only is not explicitly studied. The ability of the code to model this situation is the subject of a separate report (Reference 2). The auxiliary feedwater flow rate in gpm is input as a heat demand. The % heat demand (fraction of BTU/sec removed assuming 100% is 2772 Mwt) is derived by considering the enthalpy change in 80°F, 1050 psig auxiliary feedwat[illegible] saturated steam at 1050 psig.

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3.0 ANALYSIS RESULTS

A spectrum of auxiliary feedwater flow rate from 370 to 1500 gpm were evaluated, for both trip on high RC pressure (2300 psig) and for the anticipatory trip at zero. The results, in terms of time to reach steam generator level control, are given in Table 3-1. The cooling rate for each case is also presented.

In each case, the system parameters remained within normal ranges for a reactor trip. For example, the most severe initial undercooling case of 370 gpm AFW flow with a reactor trip on high RC pressure of 2300 psig (at 8 sec), resulted in a peak RCS temperature following trip of 583.5°F. This case also showed that AFW flow balanced the decay heat addition at about 8 minutes, with system temperatures dropping slowly (see cooling rate in Table 3-1) thereafter. Overcooling continues until 1000 psig is reached in the steam generator and OTSG level control occurs at about 40 minutes (2400 sec). For the maximum overcooling case (1500 gpm with anticipatory trip at time zero), the steam generator pressure of 1000 psig is reached within about 2 minutes. In no instance does the pressurizer empty prior to establishing steam generator level control. In each of these cases, it was conservatively assumed that pressurizer spray did not operate.

The attached figures of RCS temperature and pressure, and pressurizer level, show the effect of different AFW flow rates following a LOFW event. Each case was evaluated to that point in time where it is indicated that steam generator level control would initiate control of AFW to maintain ~1000 psig in the steam generators, with 30" startup level.

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TABLE 3-1

AUXILIARY FEEDWATER SENSITIVITY STUDY

TIME OF REACTOR TRIP, SEC.	AFW FLOW RATE, GPM	TIME TO OTSG LEVEL CONTROL, SEC	COOLING RATE, F/MIN
0	370	1900	0.5
0	500	745	2
0	1000	215	3.5
0	1500	125	11
~8 (Reactor trips	370	2400	0.5
~8 on 2300	500	1020	2
~8 psig RC	1000	320	3.5
~8 Pressure)	1500	150	11

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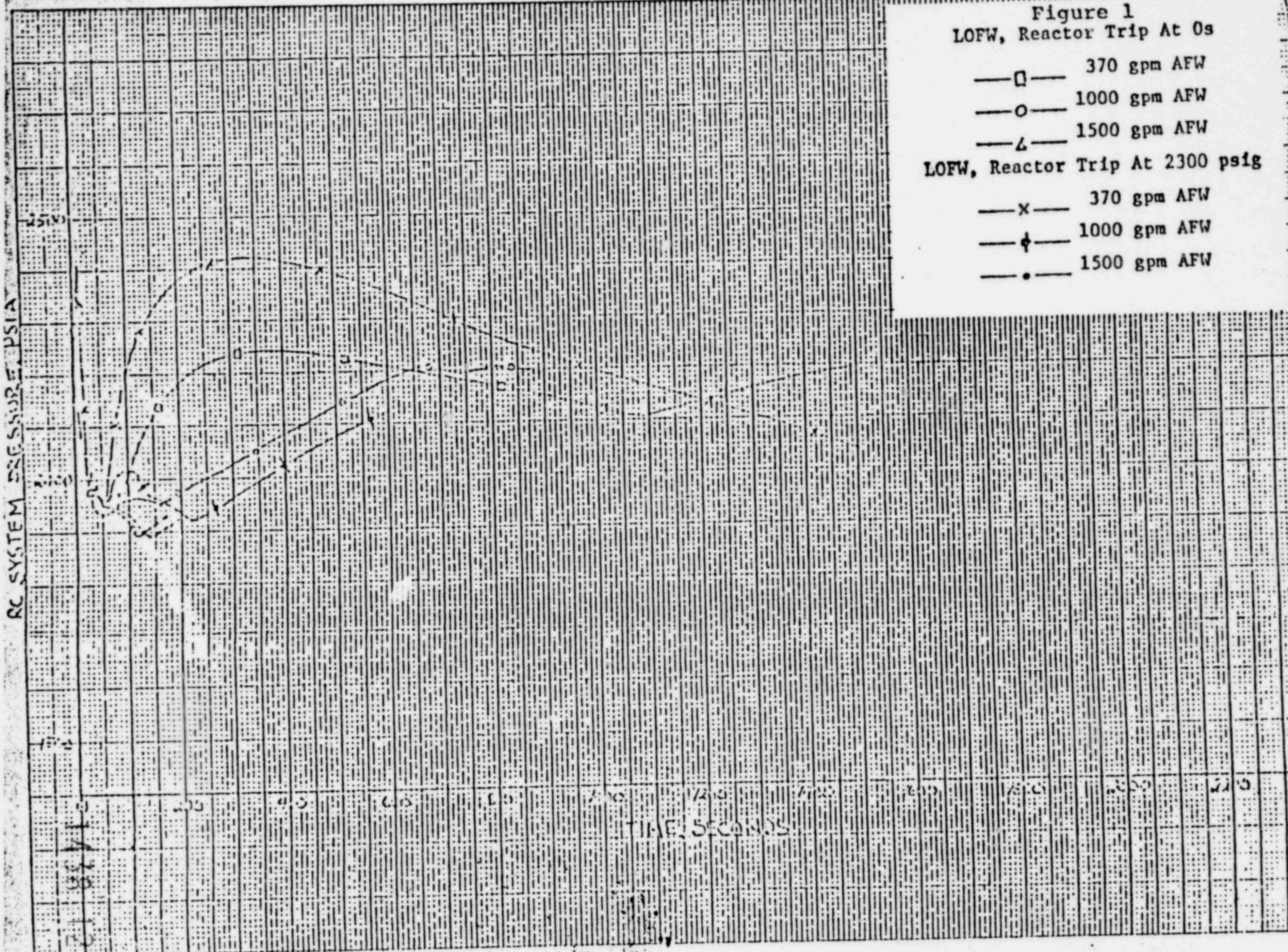

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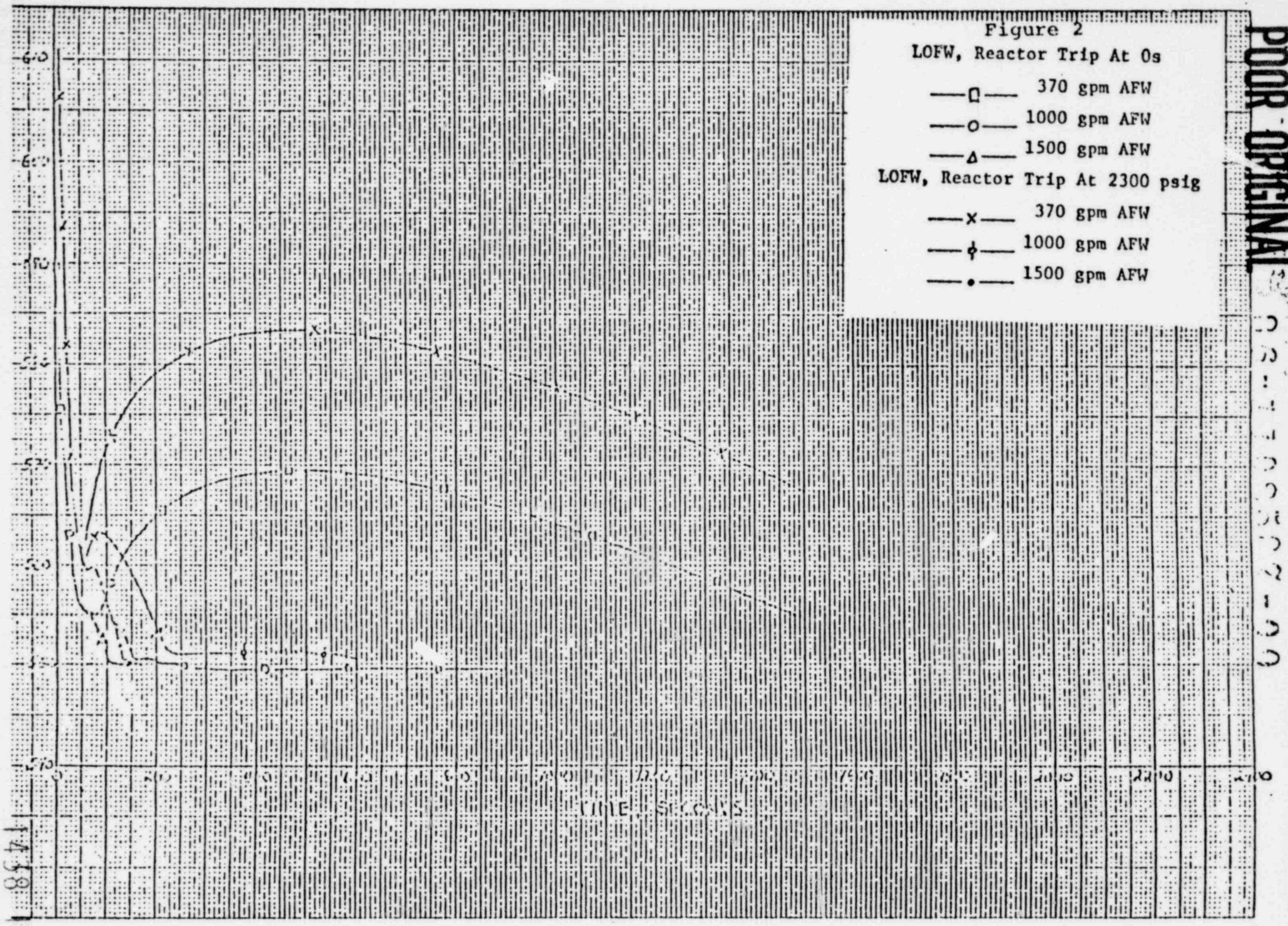
Figure 2
LOFW, Reactor Trip At 0s
370 gpm AFW
1000 gpm AFW
1500 gpm AFW
LOFW, Reactor Trip At 2300 psig
370 gpm AFW
1000 gpm AFW
1500 gpm AFW
TIME, SECONDS

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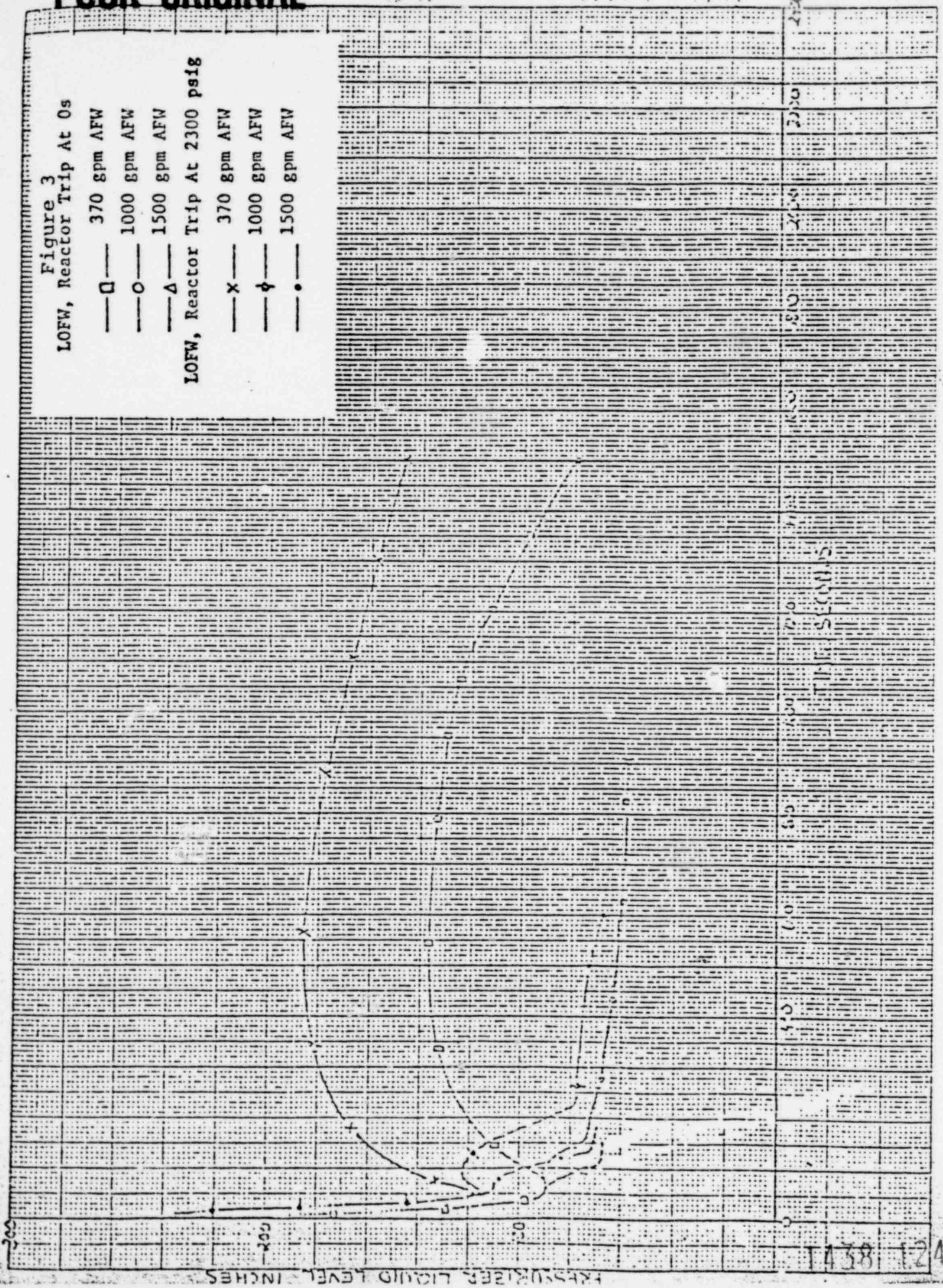

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4.0 SUMMARY AND CONCLUSIONS

Auxiliary feedwater flow rates of 370 to 1500 gpm provide satisfactory performance in that RCS parameters remain within normal ranges. For undercooling flow rates between 370 and 750 gpm, the RCS will not fill nor lift the PORV allowing heat production to decay to a level where level control can take place. For overcooling flow rates (>750 gpm), the pressurizer does not empty prior to the control point being reached.

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REFERENCES

1. B&W Report to the NRC, May 7, 1979, "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 Fuel Assembly Plant".

2. B&W Report, "Anticipatory Trip Sensitivities", May 21, 1979.

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QUESTION

4. Your response to Question 4 is not complete. Provide "as built" drawings and qualification documentation for the existing steam generator level instrumentation.

RESPONSE

The "as-built" drawings and qualification documentation forthe existing steam generator level instrumentation will be submitted at a later date under separate cover.

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QUESTION

5. Your response to Question 6 is not complete. Provide qualification documentation for the new "safety related" control on EFW flow measuring devices.

RESPONSE

The qualification documentation for the EFW flow measuring devices will be submitted to the NRC under separate cover.

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QUESTION

6. Provide design drawings for the modifications which provide for control room annunciation of all automatic start conditions of the EFW system.

RESPONSE

The design drawings for the modifications which provide for control room annunication of all automatic start condition of the EFW system will be submitted under separate cover during the first week in January, 1980.

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QUESTION

7. Your response to Question 10b is not complete. Provide the test plan (procedure) for the proposed 72-hour endurance test on all EFW pumps, including your acceptance criteria.

RESPONSE

The test procedure, including acceptance criteria, for the 72-hour endurance test of the EFW pumps will be provided for your review and concurrence prior to conducting the test. The test will be completed prior to start-up and the results of the test made available to the on-site NRC inspector.

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QUESTION

8. Your response to Question 10e is not complete. Provide the revised procedures as indicated for assuring an EFW water supply.

RESPONSE

Procedure revisions detailing the method of assuring an EFW water supply are in the draft stage and have not received the approvals required by Administrative Procedure #1001 and are not presently available. These procedures will provide the methods for utilizing the following sources of EFW supply:

1.) Condensate storage tanks
2.) Demineralized water storage tank
3.) Reactor Building Emergency Cooling River Water system via EF-V4 and EF-V5.

These procedure revisions will be provided upon completion of the review and approval process.

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QUESTION

9. Your response to Question 10g is not complete. Provide "as-built" drawings and a description in the Restart Report for the existing design for terminating EFW flow to a steam generator when low steam line pressure is revised.

RESPONSE

The steam line rupture detection is shown on elementary wiring diagram SS-209-143 and SS-209-144. Each steam generator is equipped with four (4) pressure switches, specifically PS 600, PS 601, PS 602 and PS 603 for steam Generator "A" and PS 604, PS 605, PS 606 and PS 607 for steam generator "B".

The steam line rupture detection system is designed to actuate when the steam pressure drops below approx. 600 psig. The actuation is on a steam generator basis. The actuation system associated with one steam generator is independent of the other steam generator.

The following is a description of the actuation associated with steam generator "A". The actuation associated with the steam generator "B" is similar.

Referring to electrical elementary diagram SS 209-143, the steam line A1 is monitored by PS 600 and the steam line A2 is monitored by PS 601. These two pressure switches control solenoid valves EF-V30A (SV1), FW-V17A (SV1), FW-V16A (SV1), EF-V30A (SV3), FW-V17A (SV3) and FW-V16A (SV3). On low steam generator pressure these solenoid valves energize. These valves are piped in series in the "signal air line" between the E/P converters and valve positioner of the EF and FW valve.

Energizing both solenoids results in dumping the "signal air" to the valve positioner to atmosphere, resulting in valve closure. This pneumatic anding of the solenoid provides for reliability and on-line testing one solenoid at a time.

To meet the single failure, a redundant set of pressure switches and solenoid valves are provided as shown on elementary wiring diagram SS 209-144.

Separate D.C. power supplies are provided. Separation of cable and seismic qualification are also provided.

The design meets the single failure requirement of IEEE 279-1971 and allows for on-line testing.

Bypass/defeat of the signal is required for normal shutdown. Switches 43A (#952) and 43A (#953) are provided. An alarm soundsif any defeat switch is actuated.

An alarm is also provided to indicate any pressure switch actuation. Indicating lights monitor each pressure switch circuit.

Limit switches on valve FW-V16A and FW-V16B are used in the control circuit of valve FW-V5A and FW-V5B.

QUESTION

10. You have committed to modify the EFW system by providing automatic loading of the motor driven EFW pumps on their respective diesel generators during a loss of offsite power with coincident ESAS actuation condition. You have not provided sufficient justification to support this coincident logic. It is our position that adequate emergency feedwater flow for postulated accident conditions can best be assured if the motor driven EFW pumps are automatically loaded to the diesel generator on all loss of offsite power conditions.

RESPONSE

Emergency feedwater flow will be automatically loaded to the diesel generator on all loss of offsite power conditions. See Section 2.1.1.7.3, Amendment 6.

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QUESTION

11. You have committed to provide a new Limiting Condition for Operation (LCO) in the plant technical specifications requiring an EFW flow path to each steam generator be available at 100% capacity. Your commitment states:

> "If a flow path becomes unavailable or if capacity drops below 100% to each steam generator, the plant shall be shutdown within 48 hours ...."

To meet the intent of IE Bulletin 79-05A, Item B, it is our position that the word "each" be changed to the word "either" for this part of the LCO.

RESPONSE

See Section 11.2.1 as revised by Amendment No. 6.

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QUESTION

12. We have noted that the EFW pump discharge line cross tie contains two normally open motor operated valves. Upon loss of offsite power or loss of main feedwater, a single passive failure, such as pipe rupture in one EFW discharge line, could render both EFW trains inoperable. Provide the necessary modifications and/or procedural revisions to correct this condition and mitigate its potential adverse effects to plant safety.

RESPONSE

The subject piping is not high energy piping within the meaning of the Standard Review Plan Sections 3.6.1 and 3.6.2, therefore, rupture of the pipe is only possible during system use or during system surveillance. Such a passive failure during system use (which is only under emergency conditions) is not a design basis for the EFW system and is considered too improbable to warrant mitigation. EFW system surveillance is performed monthly and includes pressurizing this piping to maximum EFW pump discharge head. If a failure were to occur, it would therefore be discovered during testing and not under emergency conditions.

The EFW system is not used for plant startup.

(Also, see the response to Question 14 in Supplement 1, part 2).

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QUESTION

13. Your response to Question 10a provided in Amendment 5 is not complete. You have not indicated whether the redundant condensate storage tanks level indication and alarms are powered from separate power supplies nor have you indicated that the low level set point provides at least 20 minutes for operator action to assure an EFW water supply assuming that the largest capacity EFW pump is operating.

RESPONSE

At present, each condensate storage tank has a Low Level Alarm which is set to ensure that each tank level will be maintained above tech spec limits. A Low-Low Level Alarm will be added to each tank to give the operator at least 20 minutes warning before operator action is required to transfer to an alternate water supply. The indicators and alarms for the redundant tanks will be powered from separate power supplies. These modifications are planned as part of the long term upgrading of the Emergency Feedwater System. See response to Question 14.

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QUESTION

14. Your response to question 10j provided in Amendment 5 is not complete. Provide legible arrangement drawings for the EFW system showing the location of all system pumps, piping and valves. Provide qualification documentation which assures that the motor driven EFW pumps will start and remain operational under the environmental conditions (humidity and temperature) resulting from a postulated break in the main steam supply line to the turbine driven EFW pumps. Further, verify that the EFW control valves and actuators are qualified to function under these environmental conditions. Also, provide an analysis which justifies the environmental conditions (323°F) assumed as a result of the postulated steam line break.

RESPONSE

As described in response to Question 10j (Supplement 1, part 1) the subject break was not considered probable enough to warrant detailed design consideration at the time TMI-1 was licensed. Since TMI-1 was licensed, NRC acceptance criteria for EFW systems has been modified and the EFW system has taken on new importance. In recognition of this fact, Met-Ed has initiated a complete design review of the EFW system to upgrade it to the current licensing criteria to the extent practicable. This review will consider and resolve the type of concerns raised by questions 12, 13 and 14 above. We believe that this approach is preferred over an item resolution of issues. Nevertheless, a response to your specific concerns is given below.

The EFW pumps have been certified to withstand the calculated environment. A copy of the motor qualification certification is attached, together with the calculations which support the environmental conditions (323°F). Environmental qualification of EF-V30A/B to 323°F was not invoked as part of the original purchase order for these valves, however, efforts are under way to determine if these valves can be certified to withstand the accident environment.

Arrangement drawings showing the location of important EFW valves and piping will be provided in about one week.

RECEIVED
JUL 9 1973
W. F. SAILER

July 5, 1973
GIL-03-13

ANALYSIS OF INTERMEDIATE BUILDING ENVIRONMENT FOLLOWING A POSTULATED MAIN STEAM BREAK FOR THREE MILE ISLAND NUCLEAR PLANT

Prepared for

Gilbert Associates Incorporated

by

Nuclear Services Corporation

CAMPBELL, CALIFORNIA

Prepared by: R. A. Uffer
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Date: 7-5-73

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Nuclear Services Corporation

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1.0 INTRODUCTION

This report, prepared for the Gilbert Associates Incorporated, presents the results of analyses predicting the environmental effects of a postulated main steam line break inside the intermediate building for the Three Mile Island Nuclear Station - Unit 1. The analyses included thermal hydraulic blowdown analyses to calculate the mass flow rate and enthalpy of the fluid leaving the break, and time history building temperature and relative humidity calculations. The analyses considered only circumferential breaks, since this type of break gives the maximum total flow and most severe conditions in the building.

The configuration of the piping and the postulated rupture locations are shown on Figure 1-1. The compartment configuration in the intermediate building is shown on Figure 1-2. A description of the analytical procedures and the analytical results, in terms of building temperature and relative humidity vs. time after the break, are presented on the following pages.

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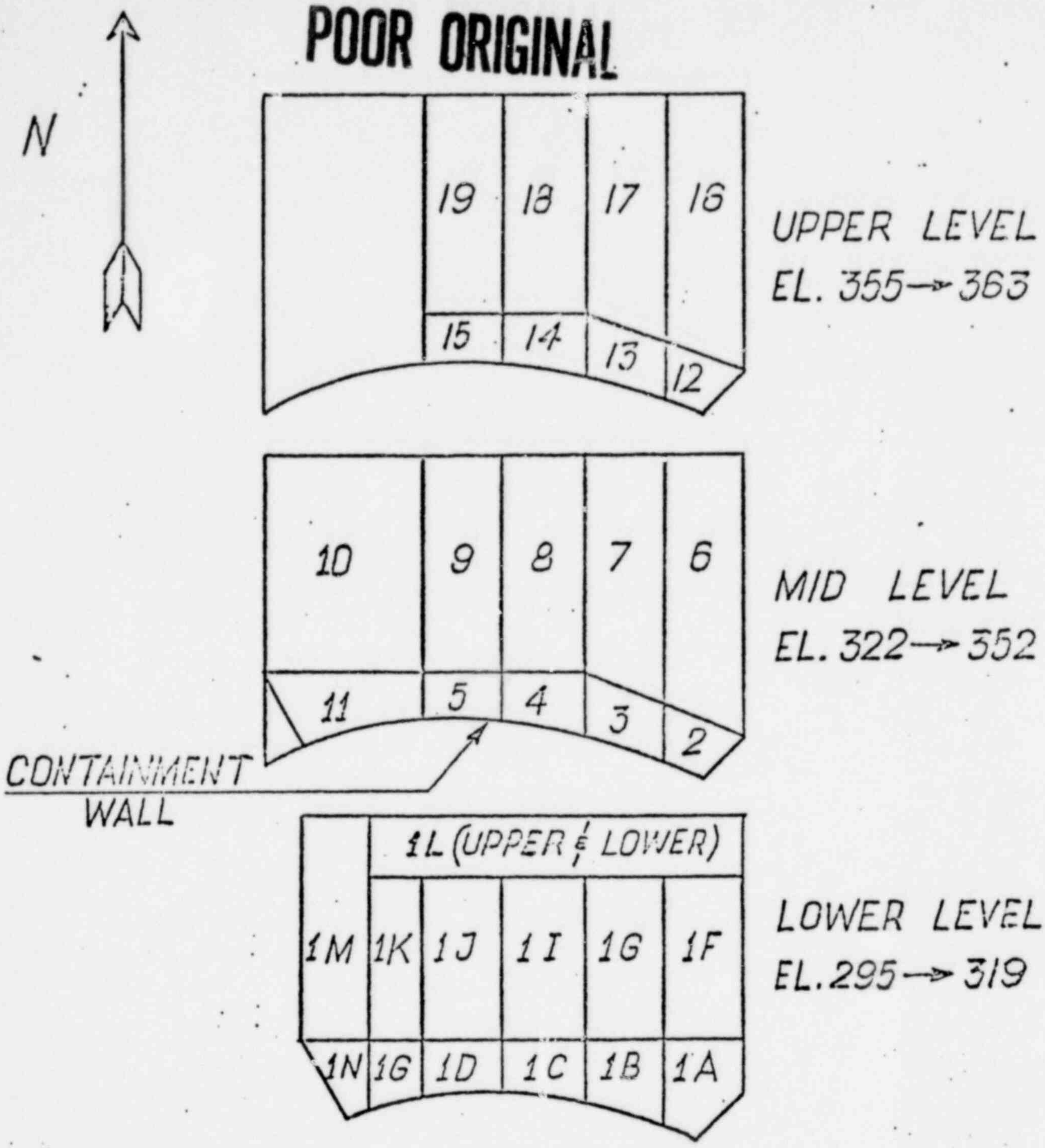

FIGURE 1-2
THREE MILE ISLAND
INTERMEDIATE BUILDING
SUBCOMPARTMENT LAYOUT

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2.0 ANALYTICAL DISCUSSION

2.1 System Description

2.1.1 Main Steam System

The main steam system consists of the piping used to deliver steam from the two steam generators to the main generator high-pressure turbine. It also includes piping to provide steam during startup, shutdown, and normal operation to the main feedwater pump turbines, and during blackout or failure of both main feed pumps, to the emergency feedwater pump turbine which exhausts to atmosphere. There are two main steam lines from each steam generator to the main generator turbine or a total of four lines. The only cross connection between lines is in the turbine steam chest between the turbine stop valves and control valves (main steam isolation valves). The turbine stop valves are both quick and tight closing; the control valves are quick closing. The piping arrangement is such that the rupture of a line from one steam generator will not blow down the other steam generator,thus ensuring that a steam supply is available to the emergency feedwater pump turbines.

2.1.2 Main Steam Piping Analyzed

The main steam piping considered in the rupture analysis consists of the sections of the four main steam lines (lines ME-38, -39, -40, -41) between penetrations numbers 419, 114, 113, 112 respectively, and the main steam stop valves in the turbine building. All four lines enter the

intermediate building at the reactor building containment wall. Inside the intermediate building the piping passes through several wall openings, then penetrates the intermediate building wall to the turbine building through 36 inch diameter wall sleeves.

2.2 Analytical Methods

The blowdown analyses were performed using the transient thermal hydraulic computer program PRTHRUST (Reference 1). This program models a physical system as an assemblage of volumes interconnected by flow paths. Characteristics of a volume include state of the fluid and possible energy addition. Volumes are used to model the steam generators, feedwater heaters and piping volumes. Flow paths are used to interconnect volumes, and can include operable valves, check valves, fills, and pumps. The program allows the operation of the valves, heat exchangers and fills to be triggered by time or by a physical signal such as pressure. Subcooled break flow is calculated using the unsteady state Bernoulli equation and saturated break flow is calculated using Moody's critical flow model. Results of the analysis include time history values of break flow, enthalpy and other thermodynamical quantities.

The temperature and humidity of the intermediate building are evaluated by another computer program CONTEMPT (Reference 3). CONTEMPT predicts the pressure-temperature response of a building due to a fluid line rupture. The input conditions for the program are the flow rate and the enthalpy of the leaking steam. The building volume is separated into a liquid region

and a vapor region. Each region is assumed to have a uniform temperature, but the temperatures of the two regions may be different. The building is represented as consisting of several heat-conducting structures whose thermal behavior can be described by the one-dimensional multi-region heat-conduction equation. The program also includes the building leakage through venting areas.

2.3 Assumptions

2.3.1 PRTHRUST analysis

The main steam line rupture model for blowdown analysis using PRTHRUST is shown in Figure 2-1.

The conditions assumed for the analysis were:

a. The reactor is operating at full power until reactor trip occurs, after which the reactor power is assumed to be given by the decay heat curve (Table 2-1).

b. A reactor trip occurs nine seconds after break.

c. The turbine stop-valves close 1.1 seconds after reactor trip.

d. The steam generator pressure is 925 psia and the steam is saturated vapor.

e. The feedwater pressure is 1050 psia and its temperature is 462°F

f. The break was assumed to be a guillotine severance of one steam line, such that flow issued from both sides of the broken line.

g. Only the circumferential breaks were considered because this type of break results in maximum flow into the intermediate building.

h. Flow into the steam generator was based upon the feedwater pump performance curve.

i. Failure of the feedwater control valve to close was assumed as the single active component failure.

j. Flow stops 600 seconds after the break when the condenser hotwell, which is the source of feedwater flow to the steam generator, is emptied.

k. The effect of both steam generators and their associated lines is considered.

l. Both sensible and decay heat in the primary system are considered.

2.3.2 CONTEMPT Analysis

The conditions assumed for the analysis were:

a. The intermediate building was modeled as one complete volume.

b. The walls and floors were modeled as exposed concrete surface to simulate their heat sink effect. The interior walls and floors were assumed heated on both sides while exterior walls were heated on the interior side only.

c. Mass flow rate and enthalpy results from the PRTHRUST analysis were used for input to the CONTEMPT analysis.

d. A relative humidity of 60% exists in the intermediate building at the time of the break with an initial temperature of 90°F. and pressure 14.7 psia.

e. The building vent area is 167 square feet and has a discharge coefficient of 0.60.

f. Outside air conditions are 90°F.

g. The evaporation rate from the condensed liquid in the building to the vapor in the building at the end of steam generator blowdown is 28 lbs/second.

h. The heat rate between the atmosphere and the liquid in the building is 4.5×10^6 BTU/°F HR.

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TABLE 2-1

REACTOR DECAY HEAT vs TIME

Time after Trip (sec.)	Decay Heat ($H(t)/H_o$)
.0	1.0
.75	1.0
.85	.95
1.0	.92
1.2	.80
1.4	.60
1.75	.30
2.0	.27
4.0	.22
5.0	.205
6.0	.20
7.0	.195
8.0	.190
100.0	0.0

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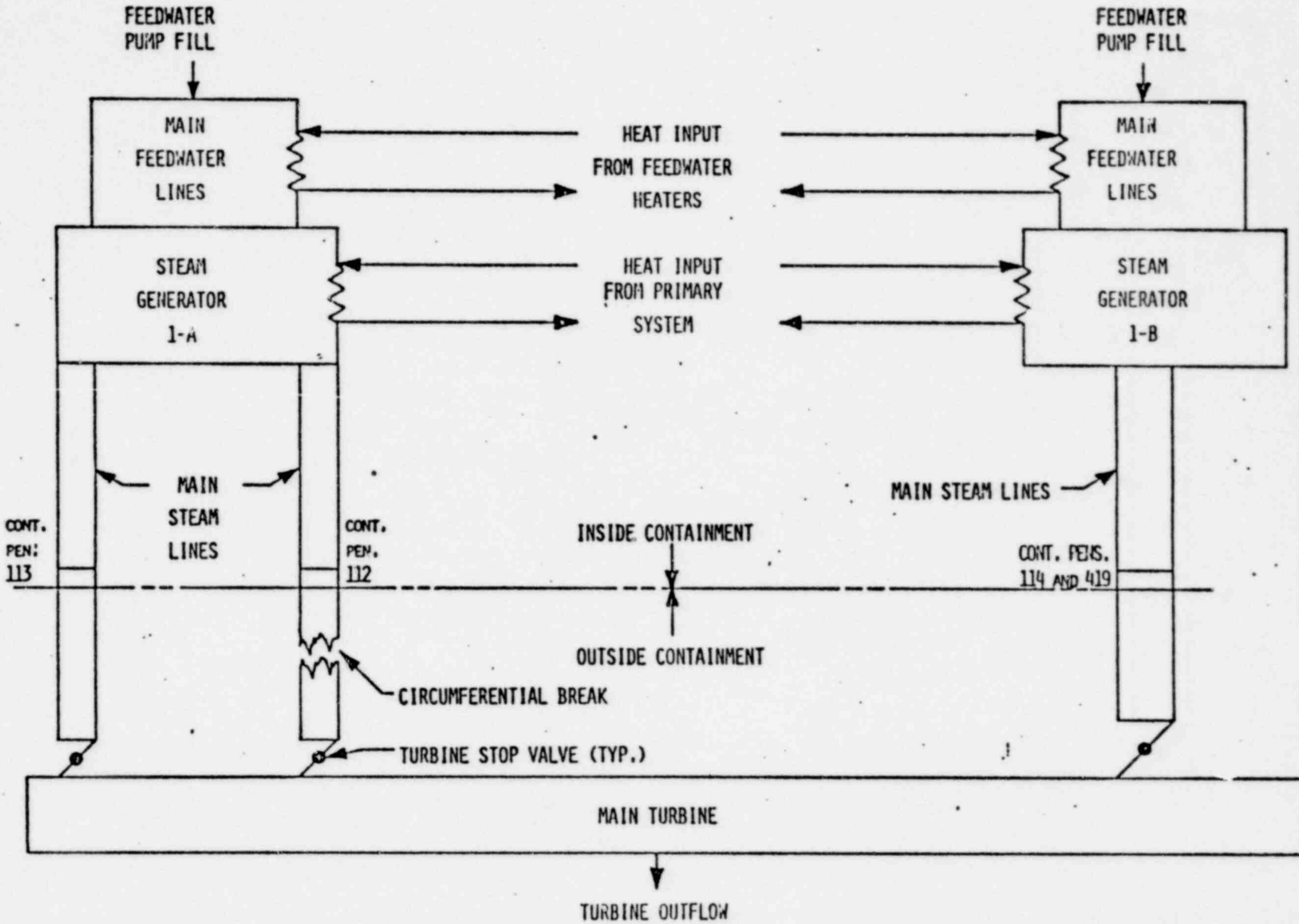

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MAIN STEAM LINE CIRCUMFERENTIAL BREAK
PR THRUST MODEL

FIGURE 2-1

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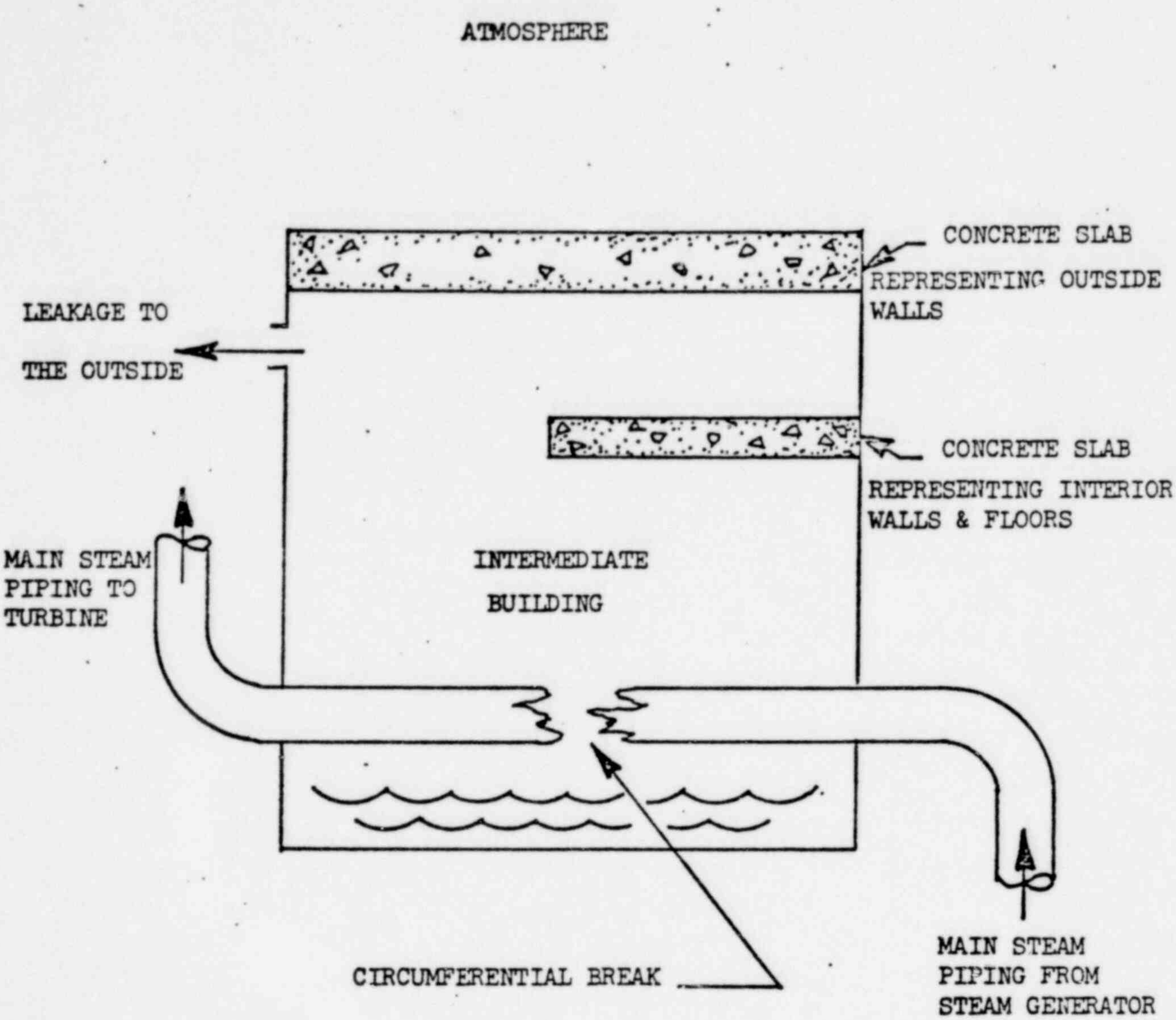

MAIN STEAM LINE BREAK
MODEL FOR CONTEMPT

FIGURE 2-2

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3.0 RESULTS AND DISCUSSION

3.1 PRTHRUST Analysis

Following a main steam line rupture, steam flows out of both ends of the broken line. The resulting depressurization of the affected steam generator is assumed to cause a reactor trip nine seconds after the break occurs. The turbine stop valves close 1.1 seconds after reactor trip, causing flow from the turbine side of the break to stop after the steam in the line between the break and stop valve has been exhausted, approximately 12 seconds after the break. Turbine side leak flow and enthalpy are shown on Figure 3-1.

Flow continues out of the steam generator side of the break, decaying with pressure, until the steam generator is exhausted. Normally, the feedwater control valve for the affected steam generator would close due to the steam line depressurization. The failure of the feedwater control valve to close is assumed to be the single active component failure for this accident. When this failure occurs, the steam generator will not exhaust as it is continually being fed by the feedwater system until the condenser hotwell is emptied 600 seconds after the break.

Initially, the fluid leaving the break is saturated steam. As the pressure decays, the enthalpy also lowers; the fluid, however, is still steam. Following the break, the water level swells in the steam generator, causing water to be introduced into the steam lines. When this water reaches the

break, fluid enthalpy drops and the flowing leaving the break is two phase. The fluid enthalpy then rises as the steam generator dries out and lowers again as the reactor heat decays. Steam generator side leak flow and enthalpy are shown in Figure 3-2.

3.2 CONTEMPT Analysis

Figures 3-3 and 3-4 show the temperature versus time for the single volume building environment. A maximum temperature of 323°F occurs at one second, decaying to a 212°F pure steam condition after about one minute. After blowdown ends at ten minutes, air drawn in from the outside results in gradual cooling of the building. The atmosphere returns to 100°F after about 40 minutes, although the liquid inside the building is still at about 120°F. The relative humidity drops initially from the 60% value due to the temperature increase caused by high energy steam addition and super-heat buildup. After about one minute, the relative humidity reaches 100% due to the presence of saturated steam. Figure 3-5 shows the relative humidity transient following the accident.

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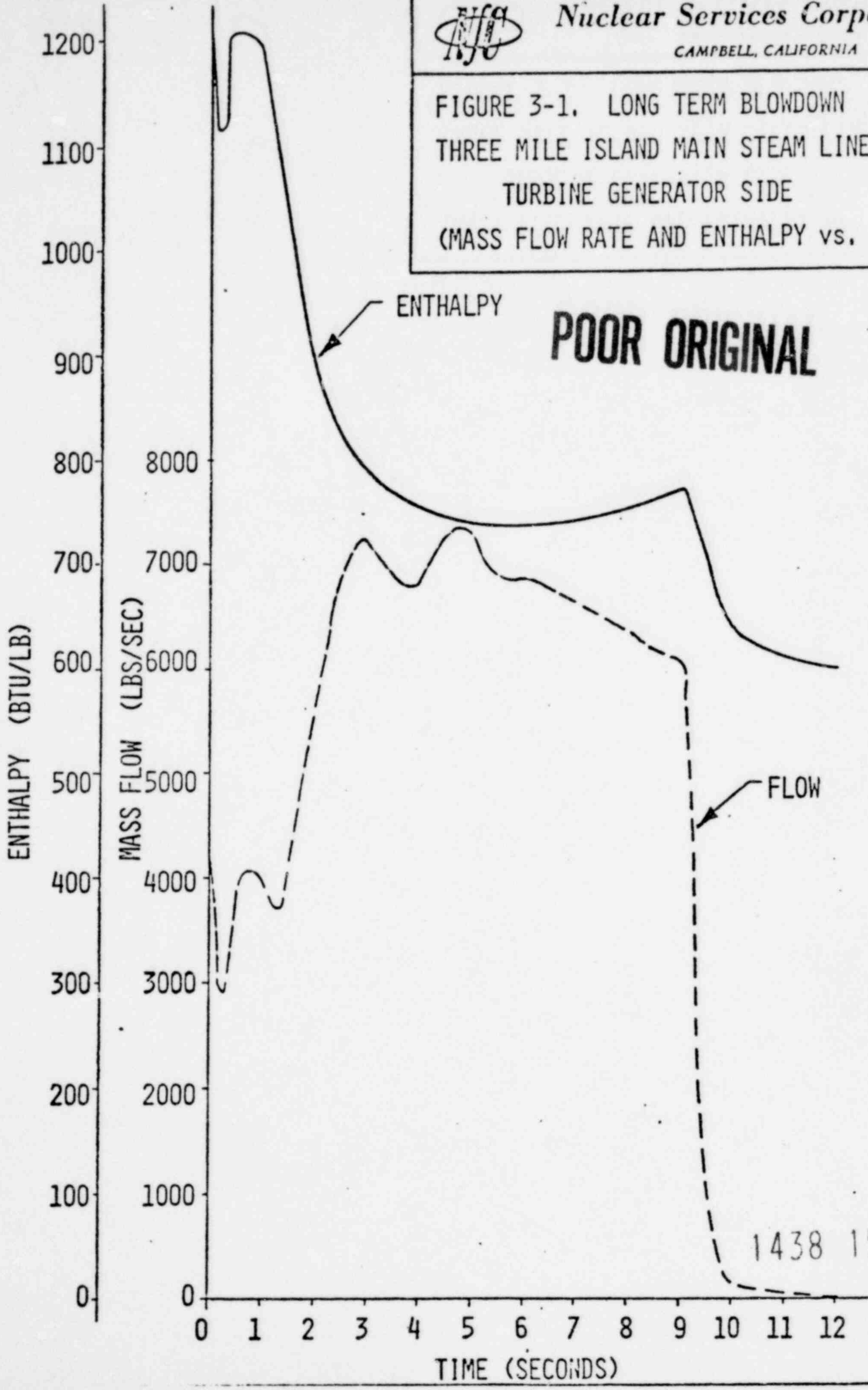
FIGURE 3-1. LONG TERM BLOWDOWN
THREE MILE ISLAND MAIN STEAM LINE
TURBINE GENERATOR SIDE
(MASS FLOW RATE AND ENTHALPY vs. TIME)



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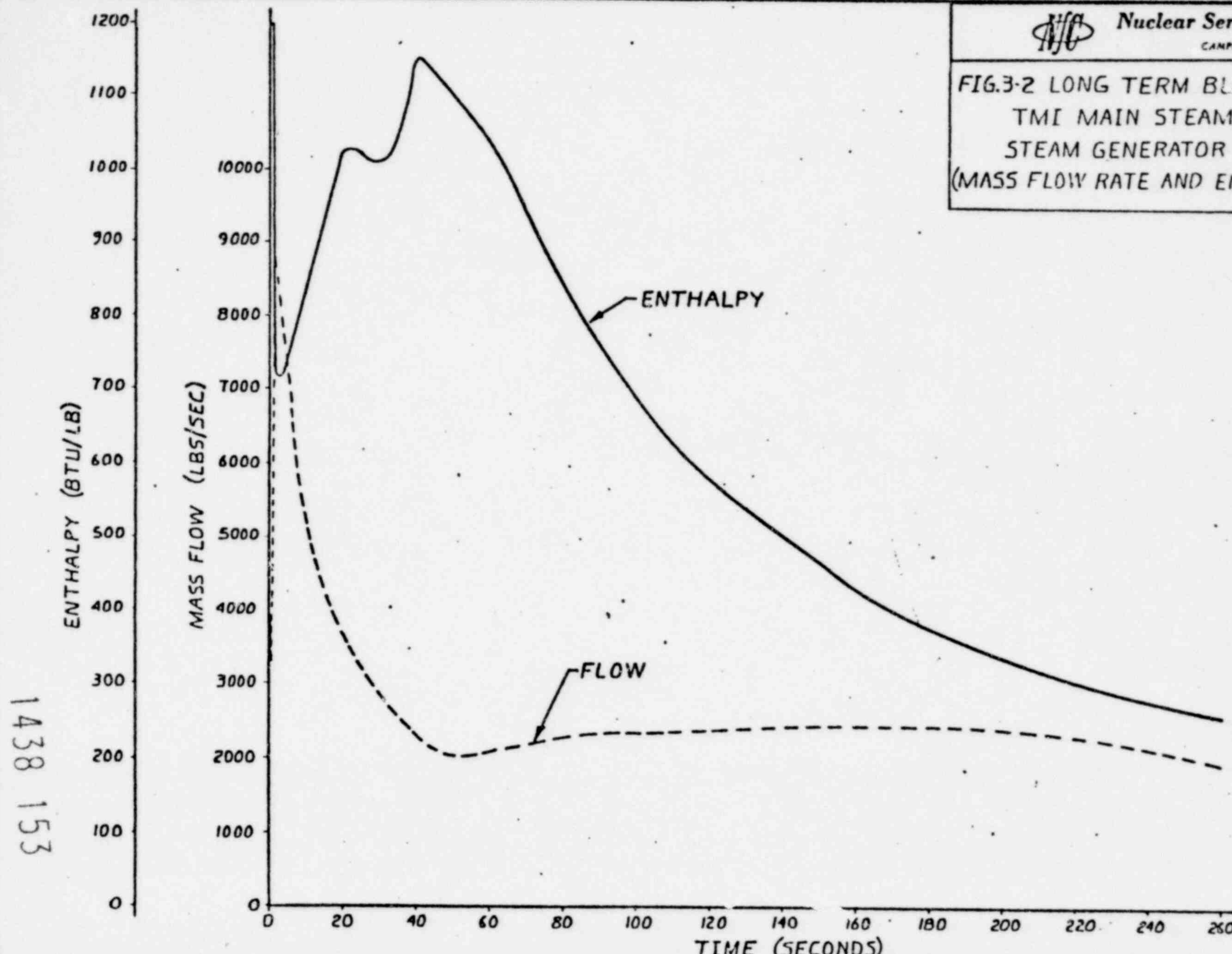

FIG.3-2 LONG TERM BLOWDOWN .
TMI MAIN STEAM LINE
STEAM GENERATOR SIDE
(MASS FLOW RATE AND ENTHALPY VS. TIME)

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FIG.3-3. TEMPERATURE VS. TIME
FOLLOWING MAIN STEAM LINE BREAK
THREE MILE ISLAND
INTERMEDIATE BUILDING
(SHORT TERM)

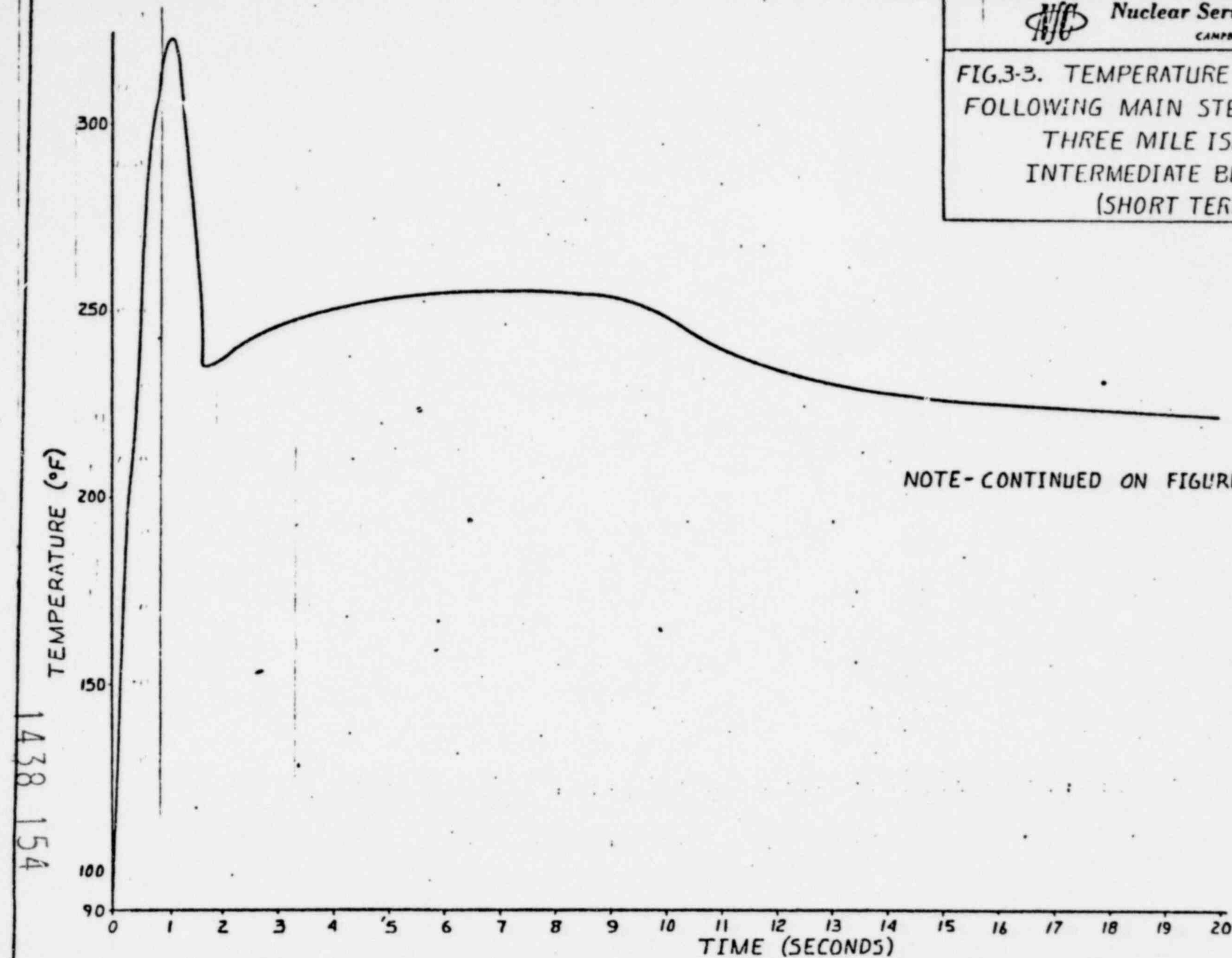

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FIG.3-4. TEMPERATURE VS. TIME (LONG TERM) FOR THREE MILE ISLAND INTERMEDIATE BUILDING FOLLOWING A STEAM LINE BREAK

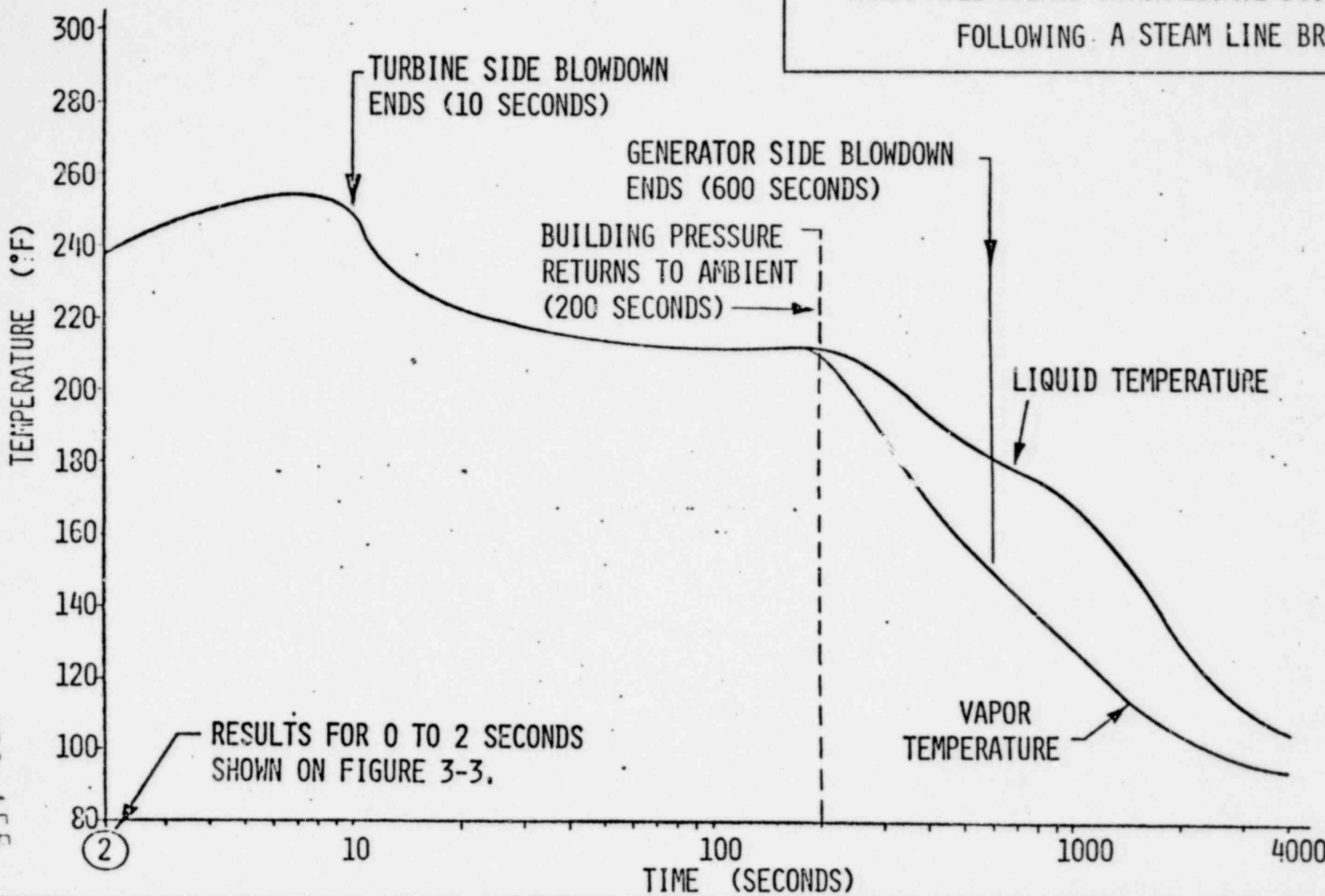

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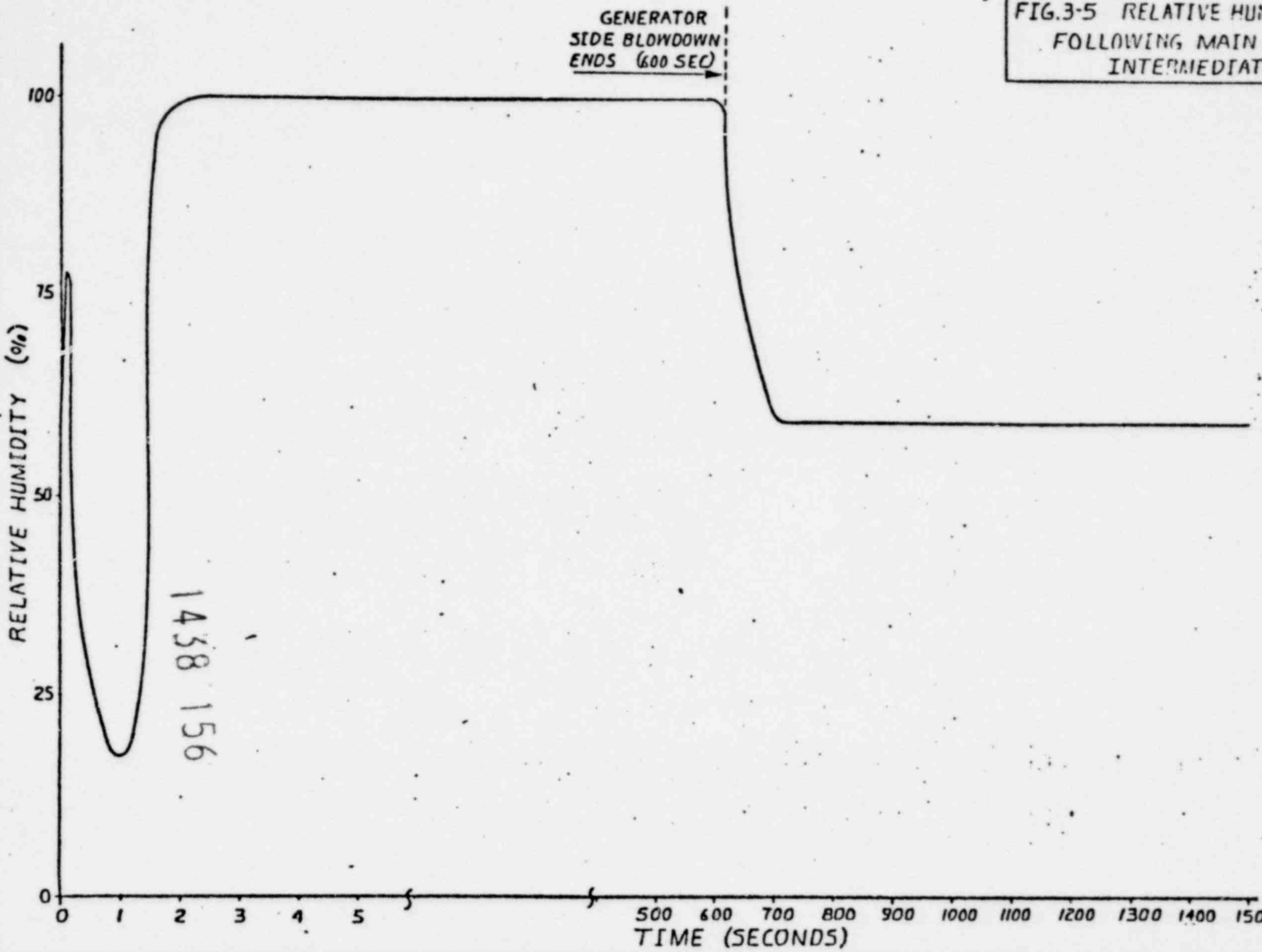
FIG. 3-5 RELATIVE HUMIDITY VS. TIME (SHORT TERM) FOLLOWING MAIN STEAM LINE BREAK T.M.I. INTERMEDIATE BUILDING



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4.0 REFERENCES

1. Nuclear Services Corporation, "PRTHRUST: Computer Code for Pipe Rupture Thrust Calculation," dated February 2, 1973.

2. Nuclear Services Corporation, "COMPRESSOR: Computer Code for Predicting the Temperature-Pressure History Within a Pressure-Suppression Containment Vessel in Responce to a Loss-of-Collant Accident.

3. Three Mile Island F.S.A.R.

4. Nuclear Services Report No. GIL-03-04 dated March 12, 1973 - Evaluation of effects of Pipe Rupture Outside Containment for Three Mile Island Nuclear Station - Unit 1.

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QUESTION

15. Provide your evaluation of anticipatory reactor trip parameters (feedwater pump turbine control oil rather than feed flow or other parameters). Include your evaluation of the need for a low steam generator level trip addressing various power levels. Discuss those transient scenarios that may not initiate anticipatory reactor trip for certain loss of feedwater/condensate events (rather than high pressure reactor trip).

RESPONSE

A response to this question will be provided by November 30, 1979 or as soon as possible thereafter.

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QUESTION

16. Your response to Question 33 is not complete. Provide or reference a schedule for submitting the necessary analyses, operator guidelines and revised emergency procedures required by Bulletin 05C. In addition, demonstrate that installation of a safety grade automatic pump trip is not practicable prior to restart.

RESPONSE

The Loss of Reactor Coolant Emergency includes instructions to trip all Reactor Coolant Pumps upon receipt of automatic actuation of High Pressure Injection (1600 psig). Guidance is being incorporated regarding restarting of the Reactor Coolant Pumps. These procedures are based on the Small Break Operating Guidelines dated November, 1979 and supporting analysis.

The analysis was submitted titled "Analysis Summary In Support of Inadequate Core Cooling Guidelines for a Loss of RCS Inventory", document No. 86-1105508-00. This document was sent to the NRC by Babcock and Wilcox and is intended to this requirement from NU REG 0578.

The modification to automatically trip the RCP's has only recently been initiated therefore the material requirements are not known. Without these details it is not possible to determine when the earliest possible date for installation might be. Preliminarily we believe that it can be accomplished by January 1981.

QUESTION

17. Your response to Question 34 does not provide sufficient information for our evaluation. Provide:

    A. Effects of liquid relief on the pressurizer electromatic relief valve following reactor trip no. 11. Include results of any inspections conducted of the valve and discharge piping. Address valve performance and operability since this reactor trip.

    B. Corrective action for main steam safety valves failure to reseat following reactor trip no. 12.

    C. Corrective action for improper operator control of feedwater prior to reactor trip no. 11.

RESPONSE

A. Reactor Coolant leak rate records indicate no change in measured leakage after reactor trip no. 11. As a result it was concluded that no apparent damage occurred to this valve as a result of the trip. In addition maintenance records on this valve indicate no repairs except for refueling interval routine maintenance have been performed on this valve.

B. Site records do not indicate that any corrective action for the main steam safety valves failure to reseat was taken as a result of reactor trip #12 nor has any further problem been experienced with the Unit #1 valves. In addition a review of the maintenance records indicate that no repairs of these valves have ever been performed.

C. No formal records of the action taken as a result of improper operator control of feedwater during reactor trip no. 11 exist. However normal practice is to provde a critique of the reactor trip with each shift. It, therefore, can be assumed that each of the shifts were instructed on the mistakes made during reactor trip #11.

QUESTION

18. Your response to Question 14 does not provide sufficient information for the staff to make an evaluation. Provide the calculations, assumptions and test data which justify the adequacy of 126 kw pressurizer heater capacity for maintaining natural circulation in the hot standby condition.

RESPONSE

The calculations, assumptions which justify the adequacy of 126 kw pressurizer heater capacity for maintaining natural circulation in the hot standby condition is stated in Section 2.1.1.3.1.2 of the Restart Report as modified by Amendment 3.

In addition, the data listed below is a summary of the heat loss through the insulation on the pressurizer which were taken during power escalation testing on May 20, 1974, at TMI. This indicates that the calculations were more conservative than the actual tests for heat loss through the insulation.

SUMMARY - HEAT LOSS THROUGH INSULATION ON PRESSURIZER

Area Considered	Sq. Ft. Area	Ambient Temp.	Average Surface Temp.	Heat Loss Through Insulation
Area below Platform at El. 312'-7"	97	93°	105°	855.5 BT
Area, El. 312'-7" to Bottom of Support Steel @ El. 324'-0", Excluding Htr. Area	318	93°	110°	4460.0 BT
Cylindrical Area (collar) Heater Bundle Insul.	43	93°	115°	823.0 BT
Elev. Circular Area, Outside of Heaters	20	93°	140°	1010.5 BT
Area, Hexagonal Part of Htr. Bundle Insul.	3	93°	200°	507.2 BT
Area at Support Steel, El. 324'-0" to 327'-0"	90	110°	207.5°	12723.8 BT
Area El. 327'-0" to 348'-0"	567	110°	152.5°	26748.2 BT
Area, El. 348'-0" to Bottom of Top Head	108	110°	151.6°	4784.9 BT
Vertical, Cylindrical Portion of Top Head	98	125°	152.4°	2550.9 BT
Top Surface of Head Insulation	76	125°	220°	10721.7 BT
TOTALS	1420			65185.7 BT
Add Convection Losses				1893.2
				2838.5
Total Heat Loss Through Insulation				69917.4 BT

NOTE:

1. Operating temperature of pressurizer is 645°F.

2. Average heat loss through insulation: $\frac{69,917.4}{1420}$ = 49.2 BTU/Ft^2/Hr.

3. Transco Reflective Insulation was 3" thick.

1438 161

QUESTION

19. As required by our October 30 letter, provide a description and schedule of the safety and relief valve test program. Verify that the program is applicable to TMI-1.

RESPONSE

As indicated in our response to Question 16 of Supplement 1, part 1, the subject relief and safety valve test program is to be conducted under the direction of EPRI/NSAC. The details of the test program have not yet been finalized. We expect that the details of the program will be available in early January 1980. We will describe the program details to the NRC at that time.

1438 162

QUESTION

20. Your response to Question 13 does not provide sufficient information for the staff to make an evaluation. Provide Appendix 2A to the Restart Report which contains calculations and test data for the PORV and safety valve flow indicators under various conditions. With respect to the seismic and environmental qualification, the staff requirements are provided in the Lessons Learned clarification letter of October 30, 1979, pages 7 and 8.

RESPONSE

The calculations and test data for the PORV and safety valve flow indicators under various conditions are not available. This information will be submitted at a later date under separate cover.

QUESTION

21. Bulletin 79-05A, Item 5.

Your response to Question 53 addresses safety related valve position checks. IE Bulletin 79-05A, Item 5, also requires a review of the positioning requirements on all safety related valves. State your intent to compare the valve positions noted in your procedures (normal and maintenance lineups, checklists, etc.) with the process flow diagrams to ensure the procedures establish the proper flow path.

RESPONSE

Positioning requirements for safety-related valves will be compared with process flow diagrams to ensure procedures establish the proper flow path. System Operating Procedures, surveillance test, and Special Operating procedures for safety-related systems all require review by the Plant Operations Review Committee and approval by the Unit Superintendent. One element of this review and approval process has been and continues to be the comparison of valve lineups with process flow diagrams to ensure applicable Technical Specification Limiting Conditions for Operation are observed. The Engineered Safeguards checklist and auxiliary operator's logsheets' valve positioning requirements have been compared by the Supervisor of Operations with the PORC/Unit Superintendent approved system operating procedure valve lineups and with the system process flow diagrams to verify that the specified positions established the required flow paths. Post-maintenance valve lineups of safety-related systems will be compared to system flow diagrams by two independent operator reviews of the switching orders prior to removal of safety tags to ensure proper restoration of the system to operability.

1438 164

QUESTION

22. Bulletin 79-05A, Item 5.

Although IE Bulletin 70-05A, Item 5, does not specifically address an independent (second operator) valve alignment verification (reference Question 54), it is our position that such a check is necessary to ensure proper positioning after necessary manipulation. State your intent to perform an independent valve alignment verification when returning the emergency feedwater system to operability after maintenance or surveillance testing. Also, submit your revised procedures reflecting this independent check.

RESPONSE

Met-Ed's acceptance of the NRC position on independent valve alignment verification is stated in memorandum GQL 1400, dated November 8, 1979 from J.G. Herbein to Richard Vollmer, NRC. This memo states, in part:

"At the completion of maintenance on ESAS and EFW Systems, an independent system checklist is required to return the system maintained to full operational status. This checklist will encompass a specific system valve and breaker lineup within the boundaries in which maintenance was conducted. Additionally, this checklist will require two independent review and signature verifications for correctness prior to its use in aligning the system in the field. Following this dual verification, the checklist will then be used in the field to return the system to full operational status.

Following surveillance tests or special operations on ESAS and EFW Systems, two independent valve and breaker lineups will be conducted within the boundary of the system affected by the tests or special operations to provide assurance that the system is returned to full operational status."

The requirements for independent (second operator) valve alignment verification for ESAS & EFW Systems are being incorporated into: 1) Administrative Procedure #1002, Rules for Protection of Men Working on Electrical & Mechanical Apparatus to require independent valve realignments upon completion of maintenance when safety tags are removed.

2) Administrative Procedure #1010, Technical Specification Surveillance Program to require independent valve realignments upon completion of surveillance tests.

3) Administrative Procedure #1001, Document Control, Appendix C, to require independent valve realignments upon completion of special operations on ESAS & EFW Systems governed by Special Operating Procedures.

As these procedure revisions have not completed the full process of review, approval and issue, they are not presently available for submission to the NRC. They will be submitted as soon as they have received the complete levels of approval required by Administrative Procedure #1001.

1438 166

QUESTION

23. Bulletin 79-05A, Item 7.

Your response to Question 58 (reference IE Bulletin 79-05A, Item 7) omits reference to operating procedure 1106-6, EFW System. State your intent to review this procedure and the others identified in your original response for proper positioning of the EFW system valves (i.e., verify valve lineups).

RESPONSE

Operating Procedure 1106-6 has been reviewed and procedural controls have been implemented to assure proper positioning of the EFW system valves. Additionally, the other procedures identified in our original response to Question 58 (Administrative Procedure 1012, Surveillance Procedures 1300-3F, 1300-3GA1B) have been reviewed and they also establish procedural controls necessary to assure proper valve positioning.

1438 167

QUESTION

24. Bulletin 79-05A, Item 9.

Your response to Question 59 (reference IE Bulletin 79-05A, Item 9) did not include operator guidance for resetting of containment isolation caused by a reactor trip or a spurious actuation signal. State your intent to provide this operator guidance.

RESPONSE

The design for containment isolation has been reviewed and it was determined that there is not automatic reopening of containment isolation valves upon resetting of the isolation signal. Operator guidance for resetting the containment isolation signal caused by reactor trip will be provided in the reactor trip procedure (EP 1202-4).

Guidance for responding to spurious actuation of containment isolation will be contained in OP 1105-3 "Safeguards Actuation System".

1438 168

QUESTION

25. Your submittal on the Lessons Learned requirements on Shift Supervisor Responsibilities (NUREG-0578), Section 2.2.1.a, as clarified in our letter of October 30, 1979) is not complete since only Position 2 is addressed. Provide the details of the management directive (Position 1), training program (Position 3), and review policy (Position 4) used to implement this item.

RESPONSE

Position 1 Management Directive

The management directive addressing the command responsibilities of Shift Supervisors in TMI-1 is included as Enclosure 1.

Position 3 Training Program

A five day decision analysis training program for Shift Supervisors which emphasized and reinforces their responsibility for safe operation and the Shift Supervisors management function for assuring operational safety will be conducted. The training program objectives are:

A. To emphasize the Command role of Shift Supervisors in applying the Command responsibilities to operating problems.

B. To assure an understanding of how to apply basic decision analysis techniques to operating problems.

Additionally, the operator accelerated requalification program includes specific training which will also emphasize and reinforce the Shift Supervisor's responsibility for safe operation and the management function to assure safety.

Examples of this are:

A. Use of Procedures

The shift supervisors' role in the use of procedures is defined and discussed.

B. Procedures Review

The shift supervisors, as well as the operators, are instructed in the content of emergency procedures.

C. Technical Specifications

The responsibility of the shift supervisor and shift personnel with respect to Technical Specifications is stressed.

D. Safety Analysis Work Shop

The team response to casualty situations is stressed.

E. Simulator Training

The team concept for casualty control was stressed. The shift supervisor was evaluated in his command role.

F. TMI Transient

Constructive criticism of operator action during the transient was stressed in this portion of their training.

Position 4 Review Policy

The administrative duties of the Shift Supervisor will be reviewed by appropriate members of the TMI Generation Group Staff in order to identify functions that detract from or are subordinate to the management responsibility for assuring safe operation of the plant.

The results of this review and recommendation will be documented by December 31. Appropriate recommendations will be approved by the Senior Vice President - Met-Ed, responsible for plant operations and will be implemented prior to 1 March, 1980.

1438 170

Met-Ed / GPU

Metropolitan Edison Company
Post Office Box 480
Middletown, Pennsylvania 17057
717 944-4041

November 28, 1979

TO: SHIFT SUPERVISORS
TMI NUCLEAR GENERATING STATION

SUBJECT: COMMAND RESPONSIBILITIES

Nuclear generating facilities have the potential to significantly impact the health and safety of the public. This potential impact places a special burden and responsibility on those who manage and command operations at the Three Mile Island Nuclear Station.

The first line of defense in portecting and assuring the health and safety of the public and the safety of personnel within the plant is the safe operation of all plant systems and components. You, as the Shift Supervisor, have the primary management responsibility until properly relieved, for the safe operation of the plant under all conditions occurring on your shift. Accordingly, you are directly charged with both the responsibility and the command authority over all shift operations, radiological controls and maintenance activities under normal and abnormal conditions. Both the supervisor coming on shift and the supervisor being relieved shall make certain they review, convey and understand plant status and on-going activities and that the activities are deemed to be in accordance with safety requirements.

Your responsibilities require you to constantly maintain the broadest perspective of operational conditions potentially affecting the general public, TMI personnel, and the safety of the plant. Maintenance of this broad perspective shall be your highest priority at all times when you are on duty. In this regard, in times of emergency, you should be sure never to become so involved in any single operation that you are preoccupied to the extent that you might not provide adequate direction when multiple operations are required in the Control Room. During accident situations you shall remain in the Control Room to manage and direct the activities of the Shift Foreman, Control Room Operators, Shift Engineer, other plant operators and required support personnel until properly relieved.

1438 171

Met-Ed / GPU

Metropolitan Edison Company
Post Office Box 480
Middletown, Pennsylvania 17057
717 944-4041

November 28, 1979

An essential element of protection of public health and safety is timely notification of State, NRC and Company officials in the event of an accident. There should be no reluctance on your part to initiate the notifications called for by the Emergency Plan if conditions indicate a potential threat to public health or safety even if more evaluation is necessary to confirm the existence of such a threat. Further, it is imperative that you provide the opportunity for guidance and direction from the line management to which you report by prompt notification to them of the existence of abnormal conditions.

Constant, vigilant recognition of your management role to maintain a command overview of the situation, to make decisions and to direct operations is the most important element in executing your responsibility to protect under all conditions, the health and safety of the public, the personnel on your shift, and the safe operation of plant systems and components under normal, off normal, and accident conditions.

J G Herbein
J. G. Herbein
Vice President

RC Arnold
R. C. Arnold
Senior Vice President

H Dieckamp
H. Dieckamp
President (Acting)

lda

1438 172

QUESTION

26. The Shift Technical Advisor discussion in NUREG-0578, Section 2.2.1.b (as clarified in our letters of September 13, 1979, and October 30, 1979) addressess accident assessment and operating experience assessment. Your submittal identifies the Shift Technical Engineer (STE) as providing the accident assessment function. State how you intend to accomplish the operating experience assessment function.

RESPONSE

Review of plant operating experience is a requirement that is placed on the Plant Operations Review Committee (PORC) by the TMI-1 Technical Specifications Section 6.5.1.6.f. To supplement the PORC review and to consider the experiences of other reactors a Plant Analysis Section has been established within the Technical Functions Division of the TMI Generation Group. This plant analysis section is responsible for reviewing all published LER's from other generating plants and determining the necessity for action at TMI-1. Although this group is located at the GPUSC home office, at least one of its members will be located at the TMI site to maintain close contact with the operations staff.

Information transmitted from the NRC through IE Bulletins, Notices, and Circulars relating to other reactors experiences is the responsibility of the TMI-1 on-site licensing group. This group is responsible for assuring that appropriate actions are completed on each IE transmittal. They assign responsibility depending on the issues, and then track satisfactory completion of any necessary action items.

QUESTION

27. Your submittal includes a proposed two year training program for the STE. This is not in agreement with the NUREG-0578 requirement for completion of STE training by January 1, 1981. State your intention to complete STE training in accordance with the NUREG-0578 requirement.

RESPONSE

The Shift Technical Advisor training requirements as described on page A-51 of NUREG-0578 are:

> "The shift technical advisor shall...have received specific training in the response and analysis of the plant for transients and accidents. The shift technical advisor shall also receive training in plant design and layout, including the capabilities of instrumentation and control in the control room."

The above training is included in parts I, II, III, and IV of the proposed training program and is scheduled for completion by January 1, 1981 in accordance with the requirements of Table B-1 in appendix B of NUREG 0578.

QUESTION

28. Your submittal indicates that shift maintenance turnover procedures will be developed in accordance with NUREG-0578, Section 2.2.1.c, Position 2. Provide your schedule for development and implementation of these procedures.

RESPONSE

The procedure has been written and reviewed by PORC and implementation is expected by January 31, 1980.

1438 175

QUESTION

29. Position 3 of Section 2.2.1.c in NUREG-0578 requires that a system be established to evaluate the effectiveness of the shift turnover procedure. Describe the system you intend to use.

RESPONSE

The effectiveness of the shift turnover procedure will be evaluated by the use of the following:

1. Requiring applicable department heads to periodically review/sign their departments' shift turnover logsheets.

2. The Operation Quality Assurance Department will periodically audit and review the effectiveness of the shift turnovers.

QUESTION

30. The October 30 clarification letter concerning the Lessons Learned requirements identified two new items under Emergency Power Supplies. The first item requires that the pressurizer heater be able to be switched to the emergency power supplies from the control room. The second item requires that the PORV and PORV Block Valve be powered from separate divisions. Provide either modified designs or justification for the acceptability of your existing designs.

RESPONSE

The requirement to provide redundant emergency power for the minimum number of pressurizer heaters required to maintain natural circulation conditions in the event of loss of offsite power does not demand the degree of ready and available access provided by a control room located manual changeover of heaters from normal offsite power to emergency onsite power. At Three Mile Island Nuclear Station Unit 1, this changeover can take place up to two hours after loss of offsite power, thereby allowing ample time to perform the changeover remote from the control room. During the situation for which this changeover is required, there will be no restricted access to the locations where the manual transfer is accomplished at the ES busses.

The manual transfer scheme described in Section 2.1.1.3.1 provides positive physical circuit separation which prevents the inadvertant connection of one safety bus to the redundant safety bus or to the non-safety bus. A changeover of heaters manually performed in the control room that requires remote operation of non-safety equipment does not provide as high a degree of assurance against jeopardizing the safety busses.

The power to the PORV block valve is ABT protected and power is supplied from two vital busses. The PORV is also powered from the same vital bus that the block valve ABT is normally slected to, however, the PORV does not require power to close but rather closes upon loss or removal of power. The objective of NUREG-0578 is, therefore fulfilled.

QUESTION

31. Provide full details of the GE model CR-2940 switch to be used to isolate the EFW manual control station from the ICS.

RESPONSE

Full details of the GE Model CR-2940 switch to be used to isolate the EFW manual control station from the ICS will be submitted to the NRC under separate cover.

QUESTION

32. Provide the electrical elementary drawings associated with the containment isolation modifications so that we may complete our review of this aspect of your design.

RESPONSE

The electrical elementary drawings associated with the containment isolation modifications will be submitted under separate cover during the first week in January, 1980.

1438 179

QUESTION

33. It appears from GAI Drawing SS-208-203 Rev. IA-1 that the manual and automatic initiation circuits are not arranged to prevent adverse interactions between them. If this is the case, provide justification for your design or modify it accordingly.

RESPONSE

The auto and manual initiating contracts are in parallel since they both must actuate the "close" circuit of the EFW Pump Motor Circuit Breaker. Otherwise the functions are diverse and independent.

1438 180

QUESTION

34. Question 12 has not been fully answered. A listing of all new loads and bus assignments is required to provide an independent evaluation of whether the emergency power trains have maintained their independence.

RESPONSE

Some of the new loads have not been assigned yet. When these loads have been assigned, theinformation will be supplied to the NRC.

QUESTION

35. Provide the basis for selecting 20% and 10% reactor power as the bypass setpoints for main turbine and main feedwater pump anticipatory reactor trip, respectively.

RESPONSE

A bypass of the turbine trip signal is automatically provided below 20% reactor power in order to provide for normal starting and shut-down of the main turbine. There is adequate steam bypass capacity to accommodate a turbine trip at 20% power without challenging the over-pressure trip function. A bypass of the FW Pump trip signal has been provided in the design with a tentative set point at 10% reactor power. Further investigation is being made to determine whether this latter bypass is required and what the set point should be. If it is determined that the bypass is not required it will be removed or the set point will be set to zero power.

QUESTION

36. Provide a detailed description of the backup capability provided for determining the position of the PORV and pressurizer safety valves beyond the differential pressure transmitters.

RESPONSE

The differential pressure transmitters will be the primary indication that either the PORV or the safety valves are open. In addition, the PORV position will be monitored by the B&W supplied accelerometers which have previously been described in the Restart Report. Another diverse indication that either the PORV or the safety valves are open is the temperature detectors on the discharge lines of these valves. We are currently completing analysis to document quantitatively how these detectors will respond at various valve openings and fluid conditions within the pressurizer. These calculations indicate that the temperature detectors will respond within a few seconds to the opening of a relief valve whereas response to the closing of a relief valve is several minutes. A plot of temperature detector response versus time for the various conditions analyzed will be prepared by January 15, 1980. This plot will be provided in the Restart Report and will also be used to train the plant operators the proper interpretation of the readings from the temperature detectors. Opening of the PORV or pressurizer safety valves will result in changes in the Reactor Coolant Drain Tank. The tank level is recorded in the control room and there is a high level alarm. There are also control room indicators for Reactor Coolant Drain Tank temperature and pressure.

1438 183

QUESTION

37. Provide detailed electrical drawings on the PORV and pressurizer safety valves position indication systems.

RESPONSE

The detailed electrical drawings for the PORV and pressurizer safety valve position indication systems were supplied to the NRC under separate cover. The drawings were:

ECM 057	D-601-44-002
D-601-17-006	D-610-15-006
D-601-16-006	D-662-17-001
D-601-42-001	D-662-16-001

QUESTION

38. With regard to a recent event at Oconee Unit 3 in which certain indications in the control room became unavailable, discuss the vulnerability of TMI-1 to a similar malfunction. Also, consider modifications which would reduce the potential for this type of event.

RESPONSE

The recent event at Oconee Unit 3 is being reviewed. If it is determined that TMI-1 could be subject to a similar malfunction we will proposed a suitable modification and the time frame in which it can be accomplished.

1438 185

QUESTION

39. Section 8 of the Restart Report indicates that additional transient and accident analyses will be performed using the RETRAN computer model. The connection of these RETRAN calculations to the previously reviewed B&W calculations (Restart Report, Section 8, Reference 2) is not clear. Explain the connection and provide a list of all calculations currently planned using RETRAN. For each analysis, indicate the reason for the calculation. The concern is that RETRAN may be replacing previously accepted analyses, or may be used to support a design modification to a safety related system of interest to the staff. If so, the staff would require supporting documentation to justify use of this computer model. For example, benchmark comparisons to data and accepted models similar to RETRAN would be necessary.

RESPONSE

Section 8 of the Restart Report draws conclusions about the acceptability of the TMI-1 design without making use of RETRAN analyses. The sources that were referenced to draw these conclusions include: the TMI-1 and TMI-2 FSAR; Reference 2 of Section 8 of the Restart Report; and, the response to Question 3, Supplement 1, Part 2 of the Restart Report. Based on these sources, we are able to conclude none of the plant modifications to TMI-1 result in a violation of safety criteria; hence, they do not result in an unacceptable risk to the public.

RETRAN/GPU-01 has not been used to come to conclusions about the acceptability of the TMI-1 plant design. However, the TMI-1 computer model is being used to perform a number of accident and transient analyses. These analyses will be used to expand the understanding of plant equipment performance during limiting and non-limiting accident and transient scenarios. RETRAN/GPU-01 has the capability of modeling plant control systems so that investigations can be made of systems such as emergency feedwater, turbine bypass and normal makeup during plant transients. The use of this detailed model provides plant specific information which is useful in operator training, evaluation of operating and emergency procedures, and design optimization. The use of RETRAN/GPU-01 does not replace the licensing basis analyses of the FSAR, or the abnormal Transient Operating Guidelines (ATOG) analyses being provided by B&W for the 177 FA Owner. Group. RETRAN/GPU-01 analyses would not be used to justify design modifications which fell outside the limits of previously accepted analyses. If RETRAN/GPU-01 results contradicted accepted analyses, then these discrepancies would be discussed and resolved with B&W. In this respect, our analyses can be considered a tool used to review proposed procedures and designs.

In summary, the RETRAN/GPU-01 model will be used to: extend our understanding of plant response to various transients; aid in preparing operator training programs; and be used as a design and procedure review tool.

Appendix 8A to the Restart Report provides some results of RETRAN/GPU-01 transient results. In the future, this Appendix will be expanded to include more modeling details as well as additional analyses. If practicable, the Appendix will eventually include all of the accident/transient cases listed in Table 8A-1.

1438 186

RETRAN/GPU-01 will not be used in support of licensing analyses used to support the initial restart of TMI-1. The model will be used in future licensing analyses. References 24, 25 and 26 of Section 8 provide benchmarks of TMI-1 and TMI-2 plant transients to RETRAN/GPU-01 predictions. In addition, RETRAN/GPU-01 similations of the TMI-2 accident can be compared to other computer code predictions and to plant data. These results will be available in the near future.

QUESTION

40. (1.7*)

Specify airborne activity levels for a clean area.

RESPONSE

A clean area is not defined in terms of airborne activity.

1438 187

QUESTION

41. (1.2.1)

Specify the ingestion EPZ.

RESPONSE

See Section 4.1.22.

1438 188

QUESTION

42. (2.1.3)

Provide a figure showing locations of schools, hospitals, in the EPZ.

RESPONSE

See Tables 1, 2 & 3.

1438 189

QUESTION

43. (Table 3)

Provide distance from site to hospitals.

RESPONSE

See Table 3.

•

QUESTION

44. (3)

Provide Tables similar to NUREG-0610 to indicate emergency classifications and their relationship to the participating authorities.

RESPONSE

See Tables 20, 21, 22 & 23.

1438 191

QUESTION

45. (4.1)

Define adverse meteorology.

RESPONSE

See Section 4.1.3

1438 192

QUESTION

46. (4.1)

Describe precautionary measures that may be taken in a General Emergency prior to a significant release.

RESPONSE

See Section 4.6.4.1 (2)

1438 193

QUESTION

47. (4.1(3))

Describe the classification system used by PEMA and the counties.

RESPONSE

See Section 4.4.2

QUESTION

48. (4.1(6))

Describe the methods of early warning of the public and the prompt initiation of protective actions within the EPZ.

RESPONSE

See Section 4.6.6

1438 195

QUESTION

49. (5.4)

Provide a specific cross reference to the information in State/ local plans requested in this section.

RESPONSE

See Section 4.5.2 and 4.5.3

QUESTION

50. (5.3)

Describe the manning of the offsite EDC by Local and State authorities.

RESPONSE

See Section 4.7.2.1

QUESTION

51. (5.3)

Provide the previously requested information for the following sections:

Regulatory Guide 1.101	Response to R.G. 1.101 See Emergency Plan Sections
5.2(1)	4.5.1.3
5.2.1(2)	4.5.1.3(1)
5.3.1(3)	4.6.5.4
5.3.1(4)	4.6.5.4
6.4(1)	4.6.4; 4.6.6; 4.6.6.1; 4.6.6.2
6.4(2)	4.6.4.1(1); 4.6.6; 4.6.6.1; 4.6.6.2
6.4.1	4.6.4.1(1)
6.4.3.2	4.6.4.3
6.5.1(2)	4.6.5.1
6.5.4(1),(2),(3)	4.6.5.4
7.2(1)	4.7.5
7.3.2	4.7.6.2
7.5(2)	4.6.5.4
8.1.2(7)	4.8.1.2
10(1)2	Figures 21, 22, and 23

Review Guideline No. 1	Response to Review Guideline No. 1 See Emergency Plan Sections
IA6	4.7.2.1
IB4	4.7.2.1
IB5	4.5.3.4
IIAB	4.4.1
IIA4	4.7.6.1(7)
IIB4	4.4.2
IIB5	4.6.6.1
IIB6	4.6.6.2
IIA7	4.6.6.2
IIIA1	4.7.6.1(7)
IVA1	Appendix C, Figures 20, 21, 22 and 23
VB2	4.8.1.2(5)
VB3	4.5.4

1438 198

QUESTION

52. Provide detailed design features of Fuel Handling Building environmental barrier.

RESPONSE

The following describes the TMI 1 and 2 Fuel Handling and Auxiliary Building supply and exhaust systems. The potential leakage paths between buildings or systems and the modifications designed to isolate the unit 1 refueling floor from the unit 1 Auxiliary Building and from the Control Access Building are discussed. These modifications include ventilation system changes and certain building layout changes. The major ventilation considerations are as follows:

A. The supply and exhaust systems for unit 1 are separate from those of unit 2. However, the unit 1 refueling floor air communicates directly with the unit 2 refueling floor air.

B. The supply systems of the Auxiliary and Fuel Handling Building (FHB) of TMI-1 are separate from each other. Both systems supply air to the building areas through duct distribution systems using outside air drawn from the air intake tunnel. Both supply fans are located in a common tunnel in close proximity to each other.

   None of the supply ducts of the Auxiliary Building are located in the FHB area. Thus, there is no potential for air leakage between Auxiliary and FH Building through c lets or through leaks in the Auxiliary Building supply duct system.

   The supply duct main for the FHB serves the general area at elevation 305'-0", the Spent Fuel Cooling Pump area at elevation 305'-0" and then serves the refueling floor at elevation 348'-0". The FHB refueling floor could communicate with the Auxiliary Building through the supply duct system because the general area and the Spent Fuel Cooling Pump area are open to the Auxiliary Building through an open doorway at elevation 305'-0".

C. The exhaust systems for Auxiliary and FH buildings of TMI-1 are separate in the specific buildings they serve but the FHB exhaust main becomes common with the auxiliary building exhaust main after leaving the FHB. The common main is directed to multiple filter plenums and fans that exhaust the mixed air.

The building modifications designed to isolate the TMI-1 refueling floor from the TMI-1 Auxiliary Building and from the Control Access Building include:

A. Two pairs of double doors at elevation 281'-0".

B. An enclosed passage at elevation 305'-0" with two main doors and one pair of equipment doors.

C. A wall at the east end of the truck bay at elevation 305'-0". The wall should be removable for large equipment access to the machine shop.

D. A security fence at the west end of the dock adjacent to the new enclosure at elevation 305'-0".

E. The stair tower between elevations 299'-2¼" & 211'-0" will be modified.

F. Pressure tight doors for the new fuel storage room at elevation 329'-0".

G. Pressure tight doors for the stair tower at elevation 331'-0".

H. An enclosure at elevator entrance with a pressure sealed door.

I. An enclosure for the ventilation duct chase in the northwest corner of the refueling floor with one pair of pressure tight doors.

The TMI-1 ventilation system modifications designed to prevent the leakage paths are given below:

A. Air leakage from the FHB through the supply duct, to the de-energized FHB supply fan to the Auxiliary Building are stopped by adding a leak tight damper in the discharge of the FHB supply fan.

B. Air leakage from the FHB through the supply duct, and the 48 x 2¼ branch duct, to the FHB general area at elevation 305'-0", and then to the Auxiliary Building will be stopped by blanking off this duct and providing an equivalent opening in the FHB supply duct. This would discharge the required 8000 cfm on the south side of the elevator shaft and this air would rise through the open stairwell to be exhausted at the refueling floor.

C. Air leakage following the same path as item B above but through a 12 x 12 branch duct in the spent fuel pool cooler area at elevation 305'-0" will be stopped by blanking off this duct. The 1000 cfm exhaust required by this area will then be supplied from the Auxiliary Building through the wall opening at elevation 305'-0".

D. Air leakage from the Auxiliary Building to the fuel building will be stopped by adding a leak tight damper in the exhaust duct main as it leaves the FHB but upstream of the connection with the Auxiliary Building main (60 x 50, elev. 348).

E. The leak tight dampers added to the FHB supply and exhaust ducts and the supply fan will function as follows:

1438 200

1. Automatically close on detection of differential pressure in the FHB with respect to the Auxiliary Building.

2. Automatically close on detection of high radiation on the refueling floor or in the FHB return air duct.

3. Open or close on manual command from control stations in the FHB and in the Control Room. The manual station would "over-ride" the above automatic signals.

The post modification system response following various assumed accidents and events is given below:

ACCIDENT/EVENT	RESPONSE
1. Differential pressure develops between the Auxiliary and FHB.	A differential between these two areas could develop as a result of the loss of either the fuel building or the auxiliary building supply fans or failure in the closed position of the dampers in the exhaust mains from either of the buildings. Leakage between the buildings in either direction would be stopped by adding the building and the ventilation system modifications.
2. High radiation in the FHB or in the FHB exhaust duct.	The ventilation system modification would automatically isolate the TMI-1 FHB from the Auxiliary Building. Supply air flow and exhaust air dampers would initially close and the FHB supply fan would stop. However, subsequently, these dampers could be opened and air flow through the fuel building could be re-established at the discretion of the plant operator. Also, the operator could limit the exhaust flow by opening only the exhaust damper. Operation of the Auxiliary Building supply and exhaust system would continue during the isolation phase.
3. High radiation in TMI-1 Auxiliary Building.	High radiation would be detected and alarmed locally and in the control room by area monitors and by a monitor in the exhaust duct from this area. The supply and exhaust system could continue operation to reduce the radiation level. If necessary, the refueling floor could be manually isolated.
4. High radiation in TMI-2 refueling floor.	The same response as noted in item 3 above would occur.

1438 201

In addition to the modifications described above, a ventilation system to mitigate the consequences of a postulated fuel handling accident in the FHB will be installed unless it can be demonstrated that such a system is unnecessary. This new system will meet the requirements of Regulatory Guide 1.52, Revision 2.

Preliminary calculations performed in accordance with Regulatory Guide 1.25 indicate that offsite doses will be a small fraction of 10CFR100 limits taking no credit for a ventilation system and assuming a ground level release. Other assumptions used are that: (1) 56 fuel pins are assumed to fail, (2) the DF for water in the fuel pool is 100, (3) core power is 2535 Mwt, (4) a radial peaking factor of 1.65 is assumed, and (5) a site X/Q of 6.8 E-4 is used.

Final results of the fuel handling accident analysis will be provided by December 7, 1979 and will form the basis for TMI-1 operation until the above mentioned ventilation system can be installed if it is necessary to install it.

QUESTION

53. For Solid Radwaste Systems, provide the following:

A. Description

B. Capacity

C. Process Control Program

D. On-Site Storage Facility

E. Expected amounts of Solid Wastes Per Year.

RESPONSE

A. A backfit to the TMI-1 Radwaste Solidification System is planned and consists of replacing the existing Protective Packaging, Inc. (PPI) System, which utilizes urea formaldehyde as the solidifying agent with one that uses a Dow Chemical Co. polymer as the binder.

The Dow waste solidification process will be used to solidify radwaste in the form of liquids, slurries, and spent ion-exchange resins. Proprietary technology is used to form stable water-resin emulsions which are then chemically cured to form hard, solid, monoliths. Liquid or slurry waste is stirred with a commercially available binder until a stable waste-binder emulsion is formed. The mixture is then cured by the addition of two chemicals. The final result is a dispersion of small spherical liquid particles (fine droplets) in a continuous matrix of cured binder.

An approved process control program will be incorporated into the system in accordance with Regulatory Guide 1.143.

The new system will be physically located in the TMI-1 Radwaste Solidification room where the PPI System is currently located.

Engineering for this backfit will begin in November 1979 and is scheduled for completion approximately April 1, 1980. Equipment installation and checkout should be completed by October 1, 1980.

B. Capacity of radwaste solidifcation system:

200 cwft/day	Solidified Waste
120 cwft/day	Raw Waste
(900 gallons/day)	

(Based on the operation of DOW System).

C. The Process Control Program will comply with Regulatory Guide 1.143.

1438 203

D. The solidified waste will be stored until shipment with the Epicore II wastes until a permanent waste storage building is available.

E. Anticipated amounts of solid radwaste produced per year:

5000 cwft	Solidified Evap. Bottoms
3000 cwft	Compacted Trash (Dry)
1000 cwft	Solidified Resin

(Based on a normal operating year with refueling outage).

1438 204

QUESTION

51. Provide information confirming that the TMI-2 decontamination and restoration operations will neither depend upon the TMI-1 radwaste systems nor will be interconnected with the TMI-1 radwaste system.

RESPONSE

Section 7.2 of the Restart Report provides information regarding the manner by which the radwaste systems for Units 1and 2 will be separated to prevent potential contamination of the unit. Upon completion of the currently planned activities, each unit will be completely separate and will be able to process all wastes associated with its own operation or recovery activities. All new processing equipment required to support the Unit 2 recovery operation will be located in Unit 2 facilities and all wastes generated will be handled in these facilities. No additional connection to Unit 1 radwaste facilities will be made, and hence no potential for cross connection to Unit 1 will be available.

1438 205

QUESTION

55. Provide the TMI-1 sump pumping and sampling procedure (SOP Z-33) for industrial waste treatment facilities.

RESPONSE

See attached procedure.

1438 206

Attachment to Question 55.

AP 1001
Figure 1001-8

Three Mile Island Nuclear Station
Special Operating Procedure

REV 3

SIDE 1

SOP NO. Z-33
(From SOP Log Index)

NOTE: Instructions and guidelines in AP 1001 must be followed when completing this form.

rev 2 - to 3
new cover sheet due to 90 days

Unit No. I & II

Date 9/16/79

1. Title WATER SUMP DISCHARGES TO IWTS & IWFS

2. Purpose (Include purpose of SOP) To ENSURE IWTS & IWFS Effluent meets release
This SOP SUPERCEDES Z-33 REV 1

CONTROLLED COPY
CONTROL ROOM
WORKING COPY

3. Attach procedure to this form written according to the following format.
   A. Limitations and Precautions
      1. Nuclear Safety - NA
      2. Environmental Safety - SEE CONTENT of PROCEDURE ATTACHED
      3. Personnel Safety - NA
      4. Equipment Protection - NA
   B. Prerequisites
   C. Procedure } SEE ATTACHED

4. Generated by [signature] Date 9/16/79

5. Duration of SOP - Shall be no longer than 90 days from the effective date of the SOP of (a) or (b) below — whichever occurs first.
   (a) SOP will be cancelled by incorporation into existing or new permanent procedure submitted by ______ ☐
   (b) SOP is not valid after 90 DAYS ☑
   (fill in circumstances which will result in SOP being cancelled)

6. (a) Is the procedure Nuclear Safety Related?
   If "yes", complete Nuclear Safety Evaluation. (Side 2 of this Form) . . . . Yes ☐ No ☑
   (b) Does the procedure affect Environmental Protection?
   If "yes", complete Environmental Evaluation. (Side 2 of this Form) . . . . Yes ☑ No ☐
   (c) Does the procedure affect radiation exposure to personnel? . . . . Yes ☑ No ☐
   NOTE: If all answers are "no", the change may be approved by the Shift Supervisor. If any questions are answered "yes", the change must be approved by the Unit Superintendent.

7. Review and Approval

Approved — Shift Supervisor [signature] 9/16/79
Date

Reviewed — List members of PORC contacted

PORC MEMBERS CONTACTED by TELECON

R. WARREN 9/16/79	ORLANDI 9/16/79
J. LAWTON 9/16/79	SUMMERS 9/16/79
W. MARSHALL 9/16/79	REED 9/16/79
T. MULLEAVY 9/16/79	RANDOLPH 9/16/79

UNIT II SUPERINTENDENT CONTACTED by TELECON OSSS 9/16/79

NRC Authorization for John T. Collins 9/16/79 Telecon 9/16/79

Approved — Unit Superintendent J. Logan 9/16/79 J. L. Seelinger 9/16/79
Date

Reviewed — Sup. Quality Control (If required) N/A N/A
Date

8. SOP is Cancelled ______ ______
Shift Supervisor/Shift Foreman Date

1438 207

"EVALUATION"

AP-1001
Figure 1001-8

Three Mile Island Nuclear Station
Nuclear Safety/Environmental Impact Evaluation

SIDE 2
SOP No. Z-33

1. Title ______

2. Nuclear Safety Evaluation

Does this SOP:

* (a) increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety? . . . . . . yes ☐ no ☐
* (b) create the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report? . . . . . . yes ☐ no ☐
* (c) reduce the margin of safety as defined in the basis for any technical specification? . . . . yes ☐ no ☐

Details of Evaluation (Explain why answers to above questions are "no". Attach additional pages if required.)

N/A

Evaluation By ______ Date ______

3. Environmental Impact Evaluation

Does this SOP:

* (a) possibly involve a significant environmental impact? . . . . . . yes ☐ no ☑
* (b) have a significant adverse effect on the environment? . . . . . . yes ☐ no ☑
* (c) involve a significant environmental matter or question not previously reviewed and evaluated by the N.R.C. . . . . . . yes ☐ no ☑

Details of Evaluation

This procedure will provide additional assurance that plant discharges meet both the NPDES and NRC limitations

Evaluation By [signature] Date 9/16/79

* NOTE: If these questions are "yes", the change must receive N.R.C. approval.

4. Unit Superintendent requests PORC review ☐ Check if YES.

5. Approval

J.L. Seelinger [signature]
Station Superintendent/Unit Superintendent

9/6/79
Date

1438 208

1.0 PURPOSE

This SOP ensures that all station sump discharges to the Industrial Waste Treatment System are monitored and sampled to ensure that 10 CFR 20 MPC Values are not exceeded.

2.0 LIMITS AND PRECAUTIONS

2.1 The following Sump Pump Breakers will be maintained open unless associated sump levels dictate pump operation. Prior to breaker closure and subsequent transfer of liquid to IWTS, a grab sample must be taken and an isotopic analysis performed to ensure 10 CFR 20 MPC Table 2, Column 2 Values are not exceeded at the Station discharge. In addition, permission to close sump pump breakers must be obtained from the Shift Supervisor. Caution Tags will be placed on each breaker referring to this SOP. Rev 2

	Sump	Sump Pump	Breaker Location
Unit I	Turbine Room Sump IWTS NOTE: SD-P-5 shall not be used since it bypasses the IWTS	 SD-P-2A SD-P-2B	 ICTPMCC Unit 1C IDTPMCC Unit 1E
Unit I	Auxiliary Boiler Blowdown Sump IWTS	SD-P-10A SD-P-10B	Local at pump Local at pump
Unit I	Powdex Sump IWFS	SD-P-1A SD-P-1B	1ATPMCC Unit 4D 1BTPMCC Unit 4D
Unit 2	Turbine Bldg. Sump IWTS	SD-[illegible] SD-[illegible]	2-31A Unit 3B 2-41A Unit 9C
Unit 2	Tendon Gallery Sump IWTS	SD-P-13A SD-P-13B	2-37 Unit HG3 2-47 Unit JH2
Unit 2	Control & Service Bldg. Sump IWTS	SD-P-9A SD-P-9B	2-37 Unit EG1 2-47 Unit GH2
Unit 2	Control Bldg. Area Sump IWTS	SD-P-3A SD-P-3B	2-31C Unit 4B 2-41C Unit 5C
Unit 2	Diesel A Sump IWTS	SD-P-10A SD-P-10B	2-11EC Unit 3FB 2-11EC Unit 3CB
Unit 2	Diesel B Sump IWTS	SD-P-10C SD-P-10D	2-21EC Unit 2EF 2-21EC Unit 2FF
Unit 2	Pretreatment Sludge Collection Sump IWFS	WT-P-16A WT-P-16B	2-41A Unit 5E 2-31A Unit 10E
Unit 1	Pretreatment Sump IWFS	WT-P-24A WT-P-24B	Pretreatment MCC Unit 2C Pretreatment MCC Unit 2D

Rev 2

NOTE: Controls for Unit 1 Pretreatment Dual Gravity Filter Backward Flow, Skimmers, and Sludge Collectors are not included in this procedure since it could cause undue interruption of Pretreatment System operation. These discharges are monitored at the IWTS Filtration System.

2.2 Immediately following sump pump-down open the associated breaker.

3.0 PREREQUISITES

3.1 One of the following sump levels is high and contents must be pumped to the IWTS or IWFS.

Unit 1 to IWTS

Turbine Room Sump
Auxiliary Boiler Blowdown Sump

Unit 1 to IWFS

Unit 1 Pretreatment Sump
Powdex Sump

Unit 2 to IWTS

Turbine Building Sump
Tendon Gallery Sump
Control & Service Bldg. Sump
Control Bldg. Area Sump
Diesel A Sump
Diesel B Sump

Unit 2 to IWFS

Unit 2 Pretreatment Sludge Collecti
Sump

Rev

3.2 The sump to be pumped down has had within the previous 24 hours an isotopic analysis performed on a sample of the contents and it is known not to contain concentrations of radionuclides in excess of 10 CFR 20 MPC Table 2 Column 2 limitations taking into account total plant effluent flow.

NOTE: I, the s [illegible] ndicates greater than MPC values of isotopes, the Shi.. [illegible] hall review the DF and fraction of MPC after dilution calculations to insure MPC is not exceeded at the plant discharge. Enclosures I & II. Sumps that must be pumped more frequently than every 24 hours shall be sampled at least every 24 hours.

3.3 Sump analysis results will be maintained by the Shift Foreman/Control Room Operator in the Water Sample Log Book in Unit I Control Room.

4.0 PROCEDURE

4.1 Ensure Shift Foreman or Control Room Operator has obtained results of sump contents Isotopic Analysis and sum of the ratios of radionuclides is less than 1.0 at the river. Use IWTS/IWFS discharge flow rate (150 gpm)and Effluent Flow Rate to determine dilution factor, Enclosures I & II.

NOTE: If sump samples indicate greater than MPC, the Shift Foreman must review dilution factor and fraction of MPC calculations to insure MPC is not exceeded at the plant discharge. This review must be done prior to pumping the sump.

NOTE: The total fraction of MPC (the fraction from this specific sump and the existing fraction of MPC discharge pt. 001 (plant discharge) must be less than one (1). See Z-51 Rev. 4 (line three of Liquid Release Form) and examples at ECS station.

4.2 Document less than MPC at the station discharge by completing Enclosure II.

4.3 Obtain permission from the Shift Supervisor to close the respective sump pump breakers. Record the person granting permission on Attachment I. Notify the NRC NRR and I&E and the Radwaste groups of intentions to pump sump and also record on Attachment I.

NOTE: Do not use SD-P-5 to pump the Unit I turbine building sump.

4.4 Close the sump pump breakers and allow the pumps to draw down the water level as low as possible.

4.5 Open the respective sump pump breaker.

4.6 Notify the Control Room (TMI-1) of start and stop times of sump pump-downs and record in both the Unit I CRO Log and ECS Log.

4.7 Attempt to identify and isolate the source and cause of all isotopic analysis high concentration indications.

1438 211

Discharge Start Date	Time	Sample Taken Date	Time	Station Discharge ($gpm \times 10^3$)	Sump Disch. (gpm)	DF	I^{131} ($\mu Ci/cc$)	I^{131} To River ($\mu Ci/cc$)	$\frac{I\text{-}131}{MPC}$		Combined Fraction of MPC	System Disch. yes/no
					150							

7438 212

SUMP PUMPING DATA SHEET

SUMP____

IWTS)
IWFS) SUMP LEVEL BEFORE_______________ TIME_______

DATE________

SOURCE OF WATER TO SUMP

PERMISSION GRANTED TO PUMP_______________________
Shift Supervisor

IWTS)
IWFS) SUMP LEVEL AFTER_______________ TIME__________

DATE__________

NRC NRR Notification____________________ ____________________ ____________________
Name Date Time

NRC I&E Notification____________________ ____________________ ____________________
Name Date Time

RADWASTE GROUP NOTIFICATION______________ ____________________ ____________________
Name Date Time

1438 213

ENCLOSURE I

I^{131} (μCi/cc) = Concentration of I^{131} Found In Sample

$$I^{131} \text{ To River } (\mu Ci/cc) = \frac{I^{131}\ (\mu Ci/cc)}{D.F.^*}$$

$$\frac{I^{131}}{MPC} = \frac{I^{131} \text{ To River } (\mu Ci/cc)}{MPC \text{ For } I^{131} \text{ In Water}} = \text{Fraction of MPC for } I^{131}$$

10 CFR 20, Table 2, Column 2

Combined Fraction of MPC: If > 1.0 do not discharge the sump to the IWTS(IWFS). — The sum of all the MPC fractions being discharged to the river.

$$^*D.F. = \frac{\text{Station Discharge}}{150 \text{ gpm}}$$ where station discharge equals station effluent.

NOTE: If any other isotopes other than I-131 are present in concentration > MDA the weighted MPC shall be calculated and verified to be < 1.0.

QUESTION

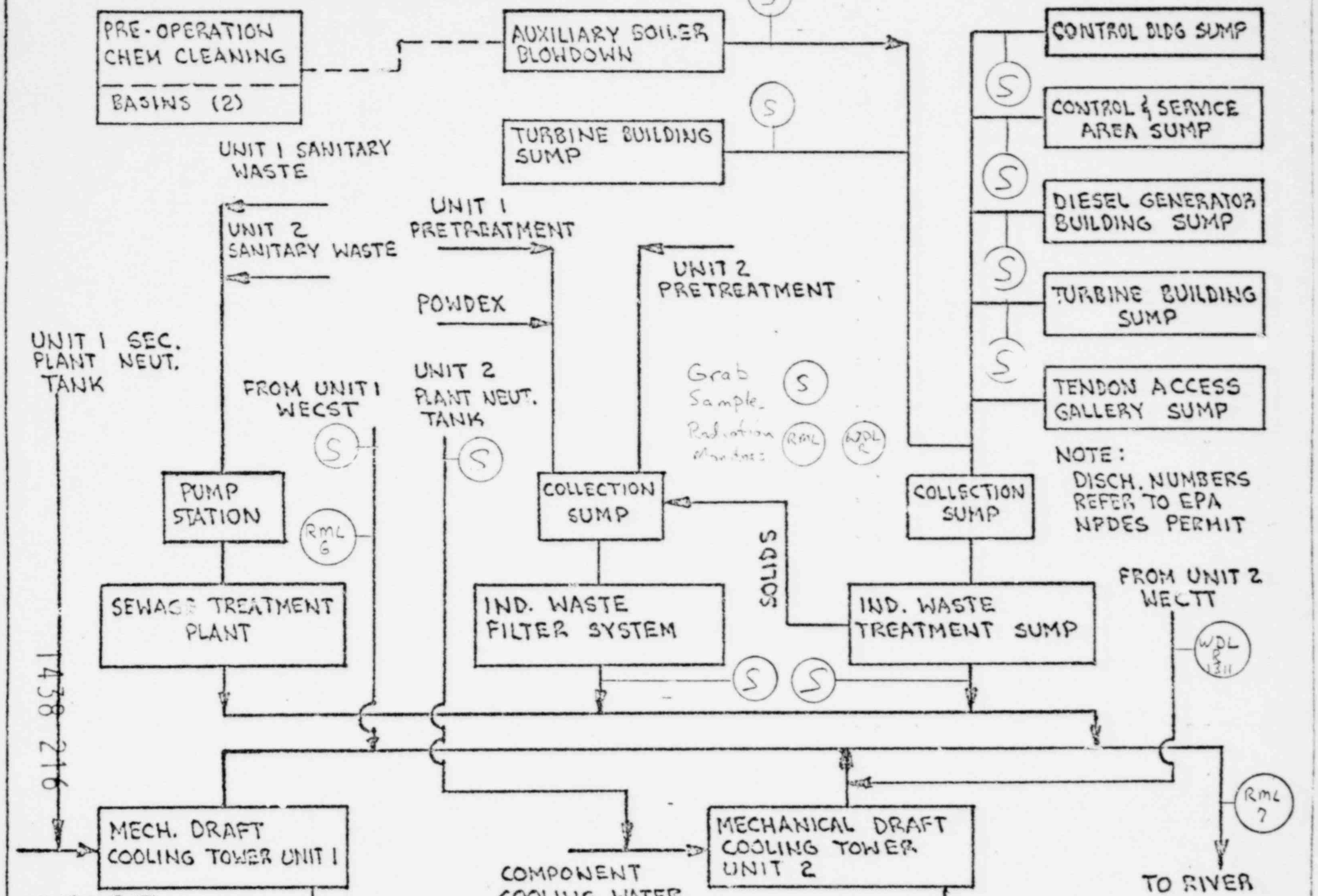
56. Provide revised Figure 7-1 "Station Discharge" indicating sample points, in-line radiation monitors, and applicable unit numbers on the equipment.

RESPONSE

See revised Figure 7-1.

1438 215

Attachment to Question 56.



TMI STATION DISCHARGES FIG ?-1

1438 216

QUESTION

57. Demonstrate conformance of the plant ventilation systems with NUREG-1.52 and NUREG-1.140 and provide leak-tight testing procedures for the ventilation systems.

RESPONSE

Met-Ed currently is working with the Operating Reactors Branch of the NRC in order to finalize Technical Specifications for the TMI-1 ventilation systems. The purpose of these new Specifications will be to invoke the testing requirements of Regulatory Guide 1.52 on these systems. In addition, before restart of TMI-1, the charcoal beds of the ventilation systems will be replaced with charcoal meeting the requirements of Regulatory Guide 1.52. After having accomplished these items the appropriate surveillance procedures will be available for NRC review. The affected ventilation systems are:

1. The Reactor Building Ventilation System (purge and exhaust).
2. The Auxiliary and Fuel Handling Building Ventilation System.
3. The Control Room Ventilation System.

There are no systems to which Regulatory Guide 1.140 applies since there are no normal filtered ventilation systems separate from the above systems.

1438 217

QUESTION

58. Does the Senior Vice-President - Met-Ed, Vice-President GPUSC have other responsibilities than TMI-1 and TMI-2? If so, describe the proportion of his time allocated to these other duties.

RESPONSE

The Senior Vice President - Met-Ed still has responsibility for the operation of fossil plants belonging to Met-Ed and construction activities associated with new fossil plants; however, he is currently devoting his time to TMI-2 Recovery and TMI-1 Restart Operations.

1438 218

QUESTION

59. Please clarify whether or not the position of Vice-President Nuclear Operations is a full-time onsite position.

RESPONSE

The Vice President - Met-Ed and Director of TMI-1 (formerly referred to as Vice President - Nuclear Operations) is assigned onsite on a full-time basis.

1438 219

QUESTION

60. Describe the specific responsibilities any functional block of the Station Organization shown in Figure 5.3-1 has for TMI-2; and if so, the time normally allocated to these responsibilities.

RESPONSE

The only areas in which TMI-2 will be serviced by the TMI-1 Station Organization will be in Training as indicated in Section 5.2.11.f.2 and for those common functions under the Manager Support Services and Logistics, i.e., a TMI-2 Security Force and personnel administration.

1438 220

QUESTION

61. Figure 5.2-1 appears not to be consistent with the description on page 5-2 with regard to Rad. Chem. and H.P. functions. Please clarify.

RESPONSE

Figure 5.2.1 has been revised to reflect a change in organization and is consistent with the description on page 5.2 and in Section 5.2.10

1438 221

QUESTION

62. Sections 5.2.12 and 5.2.13, Station Organization and Maintenance Description, are shown in Figure 5.3-1 as Station Support Organization. Please clarify this apparent inconsistency. In addition, clarify the distinction between preventive and corrective maintenance.

RESPONSE

The Maintenance Organizations for Unit 1 and 2 are entirely separate. Sections 5.2.12 and 5.2.13 have been revised to reflect the current reporting relationships.

Preventive Maintenance is defined as replacements, adjustments, major overhauls, inspections, and lubrications preplanned and scheduled on a cycle designated by engineering or by the maintenance departments in order to maintain equipment at normal efficiency.

Corrective Maintenance is defined as work required to restore equipment because of failure of components either through lack of preventive maintenance or through design or performance inadequacies.

1438 222

QUESTION

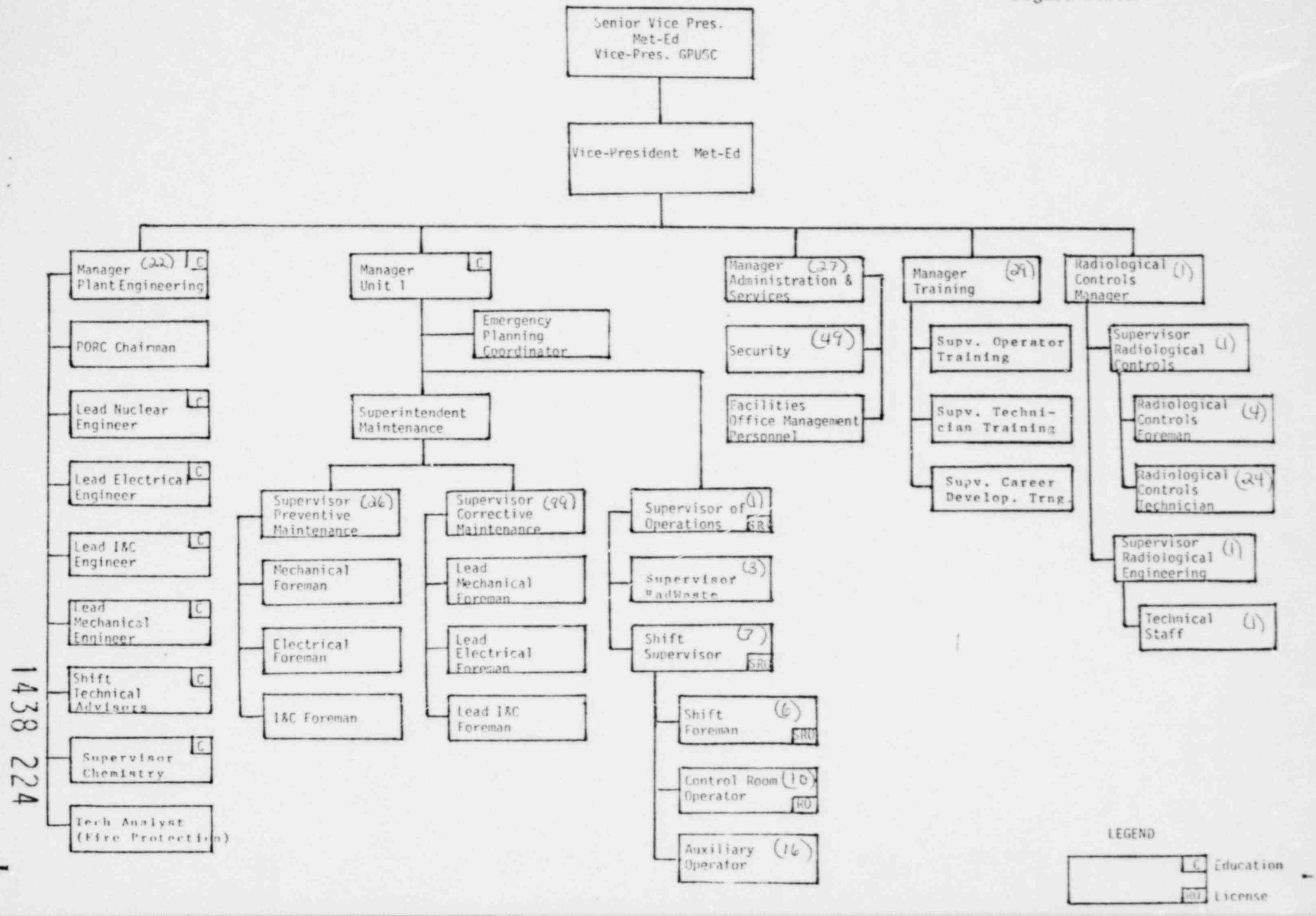
63. Expand Figure 5.2-1 to show details down to the technician level. Include qualification requirements for all functional blocks except those in the Manager Support Services and Logistics.

RESPONSE

See numbers in ( ) on attached figure.

1438 223

Figure 5.2.1



1438 224

QUESTION

64. Show the number of persons assigned to each of the functional blocks shown in the "revised" Figure 5.2-1.

RESPONSE

Within the functional blocks shown on the revised Figure 5.2-1, personnel are assigned to each group as indicated below:

1. There are 30 positions authorized to be filled by the Manager - Plant Engineering. Of these,30 are filled as follows:

   a. Manager
   b. PORC Chairman
   c. 4 Lead Engineers
   d. 6 Shift Technical Advisors
   e. Supervisor Chemistry
   f. 10 Engineers in various disciplines
   g. 7 Chemistry Foremen/Technicians

2. There are 222 positions authorized to be filled by the Manager - Unit 1. Of these, 167 are filled as follows:

   a. Manager
   b. Superintendent Maintenance
   c. Supervisor Preventative Maintenance
   d. Supervisor Corrective Maintenance
   e. 3 Maintenance Foremen (PM)
   f. 3 Lead Maintenance Foremen (CM)
   g. 83 Bargaining Unit Personnel in the various disciplines
   h. Supervisor of Operations
   i. Supervisor of Radwaste
   j. 4 Radwaste Engineer/Foremen
   k. 6 Shift Supervisors
   l. 6 Shift Foremen
   m. 19 Control Room Operators
   n. 36 Auxiliary Operators

3. There are 72 positions authorized to be filled by the Manager Administration and Services. Of these, all are filled with the exception of 4 individual positions in the personnel recruitment area.

1438 225

4. There are 18 positions authorized to be filled by the Manager Training, exclusive of administrative support. Of these, 11 are filled as follows:

   a. Manager
   b. Supervisor Operator Training
   c. 4 Licensed Operator Training Instructors
   d. 3 Unlicensed Operator Training Instructors
   e. 2 General Employee Training Instructors

5. There are 34 positions authorized to be filled by the Radiological Controls Manager. Of these, 15 are currently filled by Company employees and 15 are filled by contractor personnel. The position of Radiological Controls Manager, Supervisor Radiological Engineering, and two Technical Staff positions are vacant. The position of Radiological Controls Manager is being filled on an acting basis by the Vice President - Met-Ed.

QUESTION

65. Clarify the meaning of $\boxed{C}$ notation in Figure 5.2-1. If it is not a minimum requirement, delete it and any other reference such as "preferred" that is not a minimum.

RESPONSE

Where indicated in a functional block, "C" indicates that a college degree is required.

1438 227

QUESTION

66. Describe the function of each of the functional blocks reporting to the Director, Technical Support; Supervisor, Preventative Maintenance; and Superintendent, Radiological Controls and Chemistry.

RESPONSE

A. Reporting to the Manager Plant Engineering*

1. Chairman, Plant Operation Review Committee - Advises on all matters related to nuclear safety by reviewing all procedures and changes thereto, proposed changes and modifications to unit systems or equipment, and by performance of special reviews, investigations or analyses, and reports.

2. Lead Engineer I&C - Reports on all matters relating to instrument and control systems and components including operating, test, and maintenance procedure and review, to ensure and provide sound engineering evaluations, test procedures, and maintenance recommendations necessary for the safe, efficient operation of the unit.

   Included are such systems as Reactor Protection, Integrated Control, Non-Nuclear Instrumentation, Incore Monitoring, Loose Parts Monitoring, pneumatic Control Valves and Componenets, Turbine Electro-Hydraulic Control, Turbine Supervisory Instruments.

3. Lead Engineer Electrical - Reports on all matters relating to electrical systems and components, including operating, test, and maintenance procedure and review, to ensure and provide sound engineering evaluations, test procedures, and maintenance recommendations necessary for the safe, efficient operation of the unit.

   Included are such components and systems as Control rod drive, Pressurizer Heater Control Engineering Safeguards Actuation, Diesel Generators, Main and Auxiliary Transformers, Heat Trace, Main Generator Temperature Monitoring, Electrical Distribution, Grounding and Lighting Protection, Bettery and Battery chargers, Inverters and Vital Busses, Electrical systems for Security System, and Substation equipment.

*formerly known as the Director - Technical Support

4. Lead Engineer Mechanical - Reports on all matters relating to the mechanical engineering support of systems and components, including operating, test, and maintenance procedure and review, to ensure and provide sound engineering evaluations, test procedures and maintenance recommendations necessary for the safe, efficient operation of the unit.

   Included are such areas as Steam Generators, Reactor Coolant Pumps, Piping Hangers, supports, and snubbers, heat exchangers and coolers, Reactor Building integrates, containment isolation valve leak tightness, Emergency Diesel, ventilation systems, piping systems, valves, pumps, and filters.

5. Fire Protection - Reports on all matters relating to fire protection and associated systems and components, including operating, test, and maintenance procedure and review, to ensure and provide sound engineering evaluations, test procedures, and maintenance recommendations necessary for the safe, efficient operation of the unit.

   Included are such areas as the Fire Service System, Penetration Sealing, Ventilation fire dampers, fire doors and walls, fire extinquishers and CO_2 and Halon fire suppression systems.

6. Lead Engineer Nuclear - Reports on all matters relating to the operation of the nuclear reactor systems and components, including operating, test, and maintenance procedure and review, to ensure and provide sound engineering evaluations, test procedures, and maintenance recommendations necessary for the safe operation of the unit.

   Included are components such as nuclear fuel, Reactor vessel internals, and fuel handling equipment. Technical support will be provided by recommending such core related parameters as shutdown margin calculations, estimated critical rod positions and transient Xenon consideration.

7. Supervisor Chemistry - Accountable for laboratory chemical analysis, primary and secondary system chemistry control, water treatment, waste treatment, and radiochemistry to meet established regulations and to ensure reliable plant operations.

B. Reporting to the Superintendent of Maintenance

1. Supervisor of Preventative Maintenance - Plans, organizes, integrates, and directs the Preventative Maintenance program and the Technical Specification Surveillance program to ensure optimum equipment/systems utilization.

2. Supervisor of Corrective Maintenance - Directs the accomplishment of all work required to restore equipment because of failure of components or systems.

3. Lead Foremen reporting to the Supervisor of Corrective Maintenance direct the work of foremen and bargaining unit personnel in performing Corrective Maintenance in each of the three disciplines, electrical, mechanical, I&C.

4. Foremen reporting to the Supervisor of Preventative Maintenance direct the work of bargaining unit personnel in performing Preventative Maintenance and Technical Specification Surveillance in each of the three disciplines, electrical, mechanical, I&C.

1438 230

C. Reporting to the Radiological Controls Manager

1. Supervisor Radiological Controls - responsible for the coordination and performance of Radiological Control support and enforcement functions accomplished by Radiological Control Technicians including contract technicians assigned to the Radiological Control Technicians Group.

2. Supervisor Radiological Engineering - responsible for all matters involving the Radiological Control program design and the technical aspects related to implementation of the program support functions including ALARA, Bioassay, and Respiratory Protection.

3. Radiological Control Foreman reporting to the Supervisor of Radiological Controls direct the work of Radiological Control Technicians who provide required radiological support for station operators and maintenance activities and enforce Compliance with correct radiological control and work practices.

4. The Technical Staff reporting to the Supervisor of Radiological Engineering accomplished the design and technical aspects related to the implementation of the Radiological Control Program support functions including ALARA, Bioassay, and Respiratory Protection.

QUESTION

67. There appears to be no commitment that the Superintendent, Radiation C^ntrols and Chemistry needs any qualification requirements in chemistry and radiochemistry. Please clarify.

RESPONSE

The title for this position has been changed to "Radiological Controls Manager". The incumbent will meet the requirements of paragraph 4.4.4 ANSI/ANS-3.1 - 1978 and Reg. Guide 1.8 - 1978. The responsibility for chemistry will be assumed by a Supervisor of Chemistry who will meet the requirements of paragraph 4.4.3 of ANSI/ANS-3.1 - 1978.

1438 232

QUESTION

68. In Section 5.2.15 you state that the Aux. Operator performs the function of radiation protection monitor. The Emergency Plan Section 4 states there will be a rad/chem technician assigned to each shift. Please clarify.

RESPONSE

The Auxiliary Operator performs the function of "radiation protection monitor on his shift as required" as indicated in Section 5.2.15.b. This is not intended to be a primary responsibility since the function of radiation protection monitor is the responsibility of the Rad Chem Tech assigned to the shift.

1438 233

QUESTION

69. Describe the onsite fire protection organization including the reporting requirements for the Tech. Analyst SR-1.

RESPONSE

As indicated in the current revision to Figure 5.2.1, the Tech Analyst reports to the Manager Plant Engineering who is responsible for supervision of on-site fire protection.

1438 234

QUESTION

70. Describe the delegation of authority for overall station management in the event of unexpected contingencies of a temporary nature.

RESPONSE

The Vice President - Met-Ed has the responsibility for overall direction of day-to-day TMI-1 operations including (directing the management staff identified on Figure 5.2-1 Station Organization), and Technical Specification and Regulatory Requirement Compliance.

The Vice President - Met-Ed delegates the direct responsibility for operating the unit in a safe, reliable, and efficient manner to the Manager - Unit 1.

The Manager - Unit 1 delegates the responsibility for the day-to-day administration and direction of the Operations personnel and ensuring Compliance with the conditions of the plant operating license and technical specifications to the Supervisor of Operations.

The Shift Supervisor, who reports to the Supervisor of Operations, is the responsible management position on shift and as such shall manage and direct all activity in the physical plant and accepts full responsibility for all activity taking place on his shift.

In the event of unexpected contingencies of a temporary nature the Shift Supervisor is responsible to take whatever action is necessary to maintain the physical plant in a condition to protect and assure the health and safety of the public, the safety of personnel within the plant, and the safe operation of all plant systems and components.

If the unexpected contingency expands into an event of a more serious nature, the Shift Supervisor is charged with actuating the Emergency Plan which requires specific notification requirements and specific mobilization actions.

QUESTION

71. Provide greater detail on the personnel resumes already submitted (include relevent job assignments, educatio., and training, including dates for each). Include resumes of Lead Engineers reporting to the Director, Technical Support.

Response:

The following information amplifies biographical descriptions for incumbents listed in Sections 5.2.4, Shift Supervisor, and 5.2.5, Shift Foreman:

Incumbent A

Education:	High School Graduate - 1954
Military Service:	U.S. Army 1958-1959
Relevant Assignments:	Control Room Operator - October 1969 - October 1976
	Shift Foreman - October 1976 - July 1979
	Shift Supervisor July 1979 - Present

Incumbent B

Education:	High School Graduate - 1961
Relevant Assignments:	Auxiliary Operator - October 1969 - December 1972
	Control Operator - December 1972 - July 1975
	Shift Foreman - July 1975 - July 1979
	Shift Supervisor July 1979 - Present

Incumbent C

Education:	High School Graduate - 1959
Military Service:	U.S. Air Force - 1959-1963
Relevant Assignments:	Control Room Operator - October 1963 - August 1976
	Shift Foreman - August 1976 - April 1978
	Shift Supervisor April 1978 - Present

Incumbent D

Education:	High School Graduate - 1965
Military Service:	U.S. Navy - 1966-1971
Relevant Assignments/ Training:	U.S. Navy Basic Nuclear Power School (26 weeks) 1966-1967
	Nuclear Power Prototype School (26 weeks) - 1967
	Reactor Operator - U.S. Navy - USS Bainbridge 1969-1971
	Auxiliary Operator - February 1972-October 1973
	Shift Foreman - October 1973-May 1976
	Shift Supervisor - May 1976-Present

Incumbent E

Education:	High School Graduate - 1964
Military Service:	U.S. Air Force - 1964-1969
Relevant Assign-: ments	Auxiliary Operator - March 1969-July 1970

Control Room Operator -
July 1973-October 1975

Shift Foreman -
October 1975- October 1977

Shift Supervisor -
October 1977 - Present

Incumbent F

Education:	High School Graduate - 1963 Utah State University - 2 years
Military Service:	U.S. Navy - 1966-1973
Relevant Assignments Training:	U.S. Navy Basic Nuclear Power School (26 weeks) 1967
	Nuclear Power Prototype School - (26 weeks) - 1967
	Reactor Operator - USS Greenling - 1968-1973
	Auxiliary Operator - April 1973-August 1975
	Control Room Operator - August 1975-November 1976
	Shift Foreman - November 1976-March 1979
	Shift Supervisor - March 1979-Present

1438 238

SHIFT FOREMAN

Incumbent A

Education:	High School Graduate - 1963
Military Service:	U.S. Air Force - 1963-1968
Relevant Assign-: ments	Auxiliary Operator - July 1968 - June 1975
	Control Room Operator - June 1975 - July 1978
	Shift Foreman - July 1978 - Present

Incumbent B

Education:	High School Graduate - 1963
Military Service:	U.S. Navy - 1964-1971
Relevan Assign-: ments/Training	U.S. Navy Nuclear Power School - (26 weeks) - 1964-1965
	Nuclear Power Prototype School - (26 weeks) - 1965
	Mechanical Operator - USS Whale - 1968
	Engine Room Supervisor - USS Theodore Roosevelt - 1969-1971
	Auxiliary Operator - February 1971 - April 1975
	Control Room Operator - April 1975 - May 1978
	Shift Foreman - May 1978 - Present

Incumbent C

Education:	High School Graduate - 1966
Military Service:	U.S. Navy 1968-1974

1438 239

Relevant Assign-: ments/Training	U.S. Navy Nuclear Power School - (26 weeks) - 1968
	Nuclear Power Prototype School - (26 weeks) - 1968-1969
	USS Sea Devil - Electrical System Operator - 1969-1972
	USS Bates - Sound and Vibration Analysis - 1972-1974
	Auxiliary Operator - February 1974 - October 1976
	Control Room Operator - October 1976 - July 1978
	Shift Foreman - July 1978 - Present

<u>Incumbent D</u>

Education:	High School Graduate - 1967
Relevant Assign-: ments	Auxiliary Operator - January 1975 - October 1977
	Control Room Operator - October 1977 - August 1978
	Shift Foreman - August 1978 - Present

<u>Incumbent E</u>

Education:	High School Graduate - 1957
Relevant Assign-: ments	Auxiliary Operator - October 1969 - July 1970
	Control Room Operator - July 1970 - June 1977
	Shift Foreman - June 1977 - Present

<u>Incumbent F</u>

Education:	High School Graduate - 1962

Military Service:	U.S. Navy - 1963-1970
Relevant Assign-: ments/Training	U.S. Navy Nuclear Power School - (26 weeks) - 1965
	Nuclear Power Prototype School - (26 weeks) - 1965-1966
	Reactor Operator - USS George C. Marshall - October 1968 - March 1970
	Auxiliary Operator - June 1970 - February 1974
	Control Room Operator - February 1974 - February 1978
	Shift Foreman - March 1978 - Present

The following resumes are for Lead Engineers reporting to the Manager - Plant Engineering:

<u>I&C Incumbent</u>

Education:	University of Louisville BSEE - 1966 University of South Carolina MSEE - 1968
Relevant Assign-: ments	NSAEC Division of Naval Reactors (U.S. Navy - O3) - Nuclear Propulsion Engineer, I&C Section. Training - Bettis Reactor Engineering School; Reactor 6 months Design; 3 weeks - Prototype; 3 weeks - Shipyard - 1968-1973.
	Virginia Research, Inc. - Engineer 1973-1974
	TMI - Lead I&C Engineer Training - Several Inst. vendor courses 1 week B&W simulator 1974 - Present

<u>Nuclear Incumbent</u>

Education:	Rensselaer Polytechnic Institute BS Nuclear Engineering - 1976
Relevant Assign-: ments	Nuclear Engineering and Plant Performance Corporate Division Engineer I - Nuclear September 1976 - December 1976

Nuclear Fuels - Corporate Division
Engineer I - Nuclear
December 1976 - January 1979

TMI - I Nuclear Engineering
Engineer II - Nuclear
January 1979 - Present

Mechanical Incumbent

Education:	Pennsylvania State University BS Chemistry
Military Service:	U.S. Navy 1969-1974
Relevant Assign-: ments/Training	U.S. Naval Nuclear Power School and Prototype Training - 1969-1970 Qualified for Supervision of Operation and Maintenance of Naval Nuclear Propulsion Plant - 1970-1974 Two years experience as TMI-1 Operations Department Engineering during first two years commercial operation of Unit and first refueling outage. Three years experience as TMI-1 Lead Mechanical Engineer during commercial operation of Unit including three refueling outages.

Electrical Incumbent

Education:	Pennsylvania State University Dubois Campus Associate Degree - Electrical 1963-1965 Pennsylvania State University Capitol Campus Bachelor of Engineering Technology - Electrical 1968-1970
Relevant Assign-: ments	Summer Student - Engineering Assistant - TMI - June 1969 - September 1969 Project Engineer - TMI June 1970 - November 1973 Lead Engineer - TMI November 1973 - Present Holds Senior Reactor Operator License on TMI-I.

QUESTION

72. Does the Director, Technical Function have any responsibility for any plant other than TMI-1 and TMI-2? If so, describe the extent of those responsibilities and the time normally allocated to these responsibilities.

RESPONSE

The Director, Technical Functions is responsible for providing technical support to plants other than TMI-1 as requested by the operating companies. At this time the technical resources devoted to stations other than TMI are approximately 15% of the resources within the Technical Functions Group.

1438 243

QUESTION

73. Please clarify the description of the functions of the Director-TMI-2 recovery in the fifth paragraph of Section 5.3. It appears to be in error.

RESPONSE

The description of the functions of the Director, TMI-2 Recovery, in Section 5.3 is in error. The correct description is "The Director of TMI-2 Recovery reports to the Sr. Vice President, Met-Ed/Vice President, GPUSC, and is responsible forthe effective and safe conduct of plant activities including operations, maintenance and modifications."

1438 244

QUESTION

74. Provide the number of persons assigned to each of the functional blocks shown in Figure 5.3-1.

RESPONSE

Organizational Unit	Current Personnel	1980 Plant Personnel
Systems Engineering	30	51
Engineering & Design	53	74
TMI Engineering Management	8	12
TMI-2 Recovery Engineering	4	20
Quality Assurance	43	53

The foregoing addresses those functional blocks which address the engineering and quality assurance resources available to support the TMI station.

As described in Section 5.3.1, all staff, except as noted under Systems Engineering, includes only GPUSC permanent personnel. Certain Met-Ed personnel are permanently assigned to the Systems Engineering Department. Support from outside contractors is not included in the staffing totals, but is available on short notice to supplement the GPUSC staff as necessary. Such support continues to be obtained.

1438 245

QUESTION

75. Describe the qualification requirements in terms of education and experience backgrounds for each of the functional blocks shown in Figure 5.3-1 except for TMI-2 recovery. The breakdown should include each discipline you consider necessary to provide support for the operations staff.

RESPONSE

The functional blocks considered necessary to provide technical and quality assurance support for the operations staff are as described in the response to Question 74. The operational quality assurance plan establishes minimum qualifications for technical management personnel as having a minimum of a BS degree in Engineering or Science, and a minimum of five years relevant nuclear experience.

1438 246

QUESTION

76. Provide resumes of the managers and lead engineers for each of the functional blocks shown in Figure 5.3-1 (except TMI-2 Recovery and Director - Environment, Health and Safety) and summary information regarding educational and experience background on the staffing of each of the functional blocks shown in this figure.

RESPONSE

For the organizational units providing engineering and quality assurance support to the station organization, the management incumbent possess at least a BS degree in Engineering or Science, and the following experience.

Title	Years of Eng. Experience	Years of Nuc. Experience
Director, Technical Functions	26	24
Manager, Systems Engineering	22	22
Manager, Engineering & Design	20	14
Manager, TMI-2 Recovery Engineering	14	14
Manager Quality Assurance	18	14

Summary information on the staffing of these organizational units is described in the response to question 74.

1438 247

QUESTION

77. Describe the specific responsibility, if any, of the functional blocks shown in Figure 5.3-1, other than Director TMI-2 Recovery, has for other plants than TMI-1 and the proportion of time they will be assigned to these other plants.

RESPONSE

With reference to the engineering and quality assurance support to the station organization, the specific responsibility of organizational units for support of TMI-2 is as follows:

Systems Engineering

Provides support to TMI-2 which is currently approximately 5% of total resources.

Engineering & Design

Provides support to TMI-2 which is currently approximately 15% of total resources.

TMI Engineering Management

Provides support to TMI-2 which is currently approximately 25% of total resources.

TMI Recovery Engineering

Provides full-time support to TMI-2.

Quality Assurance

Provides support to TMI-2 which is currently approximately 25% of total resources.

QUESTION

78. Is it intended that the station support organization be permanently assigned to the site?

RESPONSE

For engineering and quality assurance support to the station organization, only the TMI-2 Recovery Engineering Department is totally located at the station. Elements of the other organizational units are assigned permanently to the station.

1438 249

QUESTION

79. Describe the regular assignment of all persons who will be assigned to the Emergency Organization shown in Figures 12 and 13 of Section 4.

RESPONSE

Figure 12:

Emergency Organization Assignment	Current Assignment
Emergency Director	Manager Unit 1 Technical Superintendent Unit 1 Supervisor Operations Unit 1
Communicator (The relief shift technical engineer will be utilized to fill this emergency position)	Shift Technical Engineer
Communications Assistants	Various administrative positions
Radiological Assessment Coordinator	Supervisor Radiological Engineering Radiological Controls Engineer Supervisor Rad Waste
Operations Support Center Coordinator	Operations Engineer Administrator-Technical Training-Nuclear Group Supervisor-Technical Training-Nuclear
Radiological Analysis Support Engineers	Group Supervisor-Tech. Training-Nuclear Administrator-Tech. Training-Nuc. Administrator-Tech. Training-Nuc.
Health Physics Coordinator	Radiological Control Foreman Radiological Control Foreman Radiological Control Foreman
Chemistry Coordinator	Chemistry Supervisor Chemistry Foreman
Security Coordinator	Shift Site Protection Sergeant
Site Security Force	Shift Security Force
Fire Brigade Team	Shift Operations Personnel
Monitors (Various) (Additional personnel may have to be called in)	Shift Operations & Rad-Con Personnel

1438 250

Figure 12:

Emergency Organization Assignment	Current Assignment
Technical Support Center Coordinator	Lead Mechanical Engineer Lead Electrical Engineer Lead I & C Engineer Manager Plant Engineering
Technical Support Engineers	Plant Engineering Staff
Operations Coordinator	Shift Supervisor
Operations Coordinator (The on-shift shift supervisor, the Supv. Operations or another Shift Supv. designated by the Emergency Director can fill this emergency position)	Supervisor Operations
Emergency Maintenance Coordinator	Shift Maintenance Foremen
Emergency Repair Team(s)	Shift Maintenance Technicians

Figure 13:

Emergency Organization Assignment	Current Assignment
Emergency Support Director	Senior Vice President Met-Ed Vice President Met-Ed
Assistant Emergency Support Director	Vice President Met-Ed Unit 1 Recovery Manager Director Reliability Engineering
Public Affairs	Manager TMI Projects Communications Administrator Public Information Coordinator Public Information
Emergency Support Communicator	Supervisor Training-Nuclear Representative-Safety Administrator-Safety
Group Leader Administrative Support	Manager Administration & Services Director-Personnel Administration Exempt Supervisor-Administration

Figure 13:

Emergency Organization Assignment	Current Assignment
Group Leader Security Support	Manager-GPUSC Security Supervisor-Security Manager Corporate Services
Group Leader Health Physics/ Chemistry Support	Radiological Controls Manager Manager Training Director Environmental Health and Safety
Environmental Assessment Coordinator	Porter/Gertz Consultants Engineer Assoc. Sr. II, S&L
Personnel Monitoring Coordinator	Dosimetry Supervisor Environmental Scientist II
Group Leader Technical Support	Director Technical Functions Manager Systems Engineering Control and Safety Analysis Mgr.
Group Leader Maintenance Support	Superintendent Maintenance Supervisor Corrective Maintenance

QUESTION

80. Describe the extent of mobilization of your Emergency Organization shown in Figures 12 and 13 for each class of emergency; i.e., (1) Unusual Events, (2) Alert, (3) Site Emergency, and (4) General Emergency.

RESPONSE

The extent of mobilization of the emergency organization is detailed in Section 4.6.1.3 of the Restart Report.

1438 253

QUESTION

81. The description of the responsibilities of the Manager, Support Services and Logistics, Section 4.5.1.1.d (page 4-50) appears to conflict with that described in Section 5. Please clarify.

RESPONSE

The Manager-Support Services and Logistics has been redesignated Manager-Administration and Services. The description of responsibilities have been revised in Sections 4.5.1.1.d and Section 5 and are now consistent.

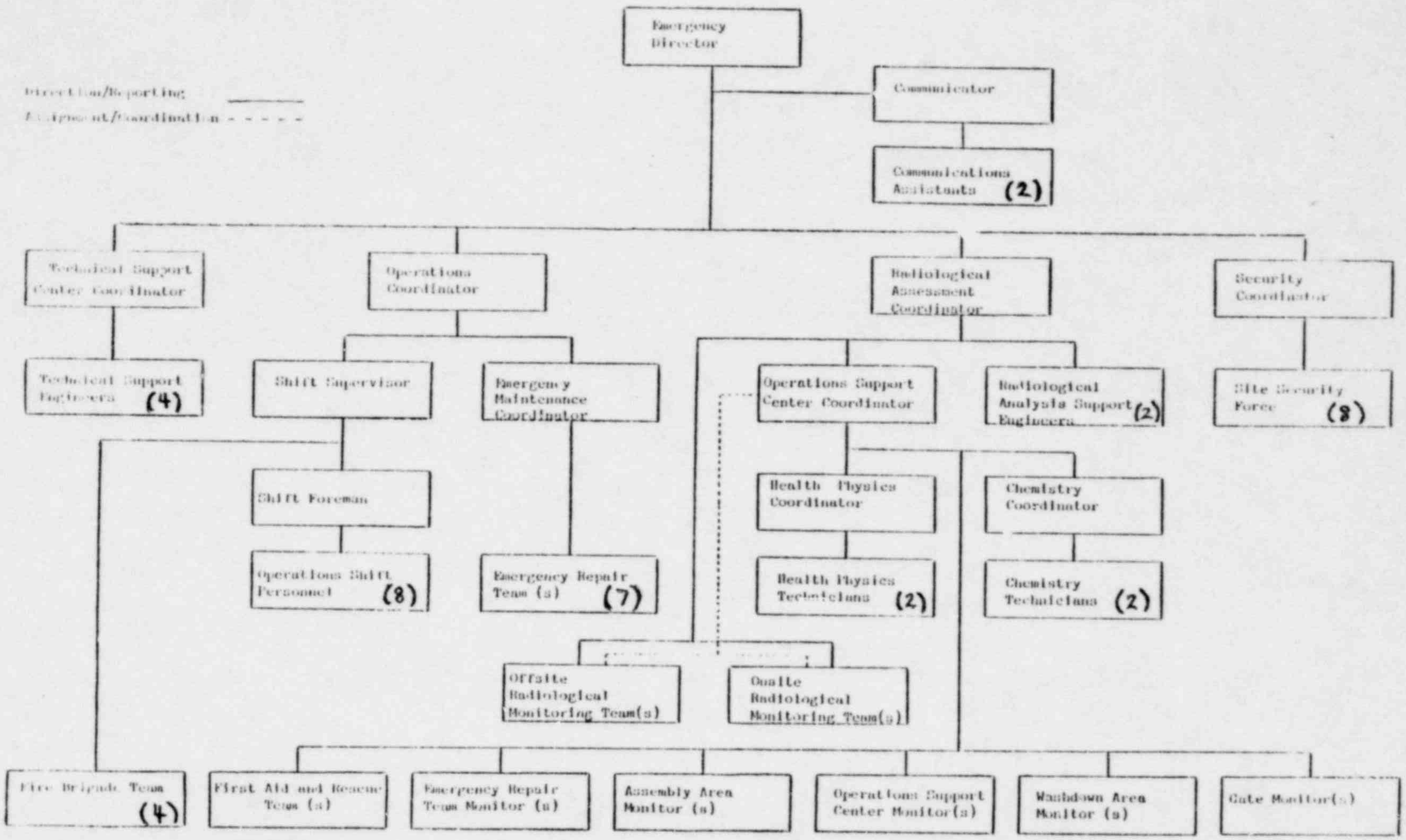
QUESTION

82. Describe the number of persons you plan to assign to each of the functional blocks shown in Figures 12 and 13.

RESPONSE

The duty section concept for staffing figure 12 is dicussed in the answer to Question 86. For the functional blocks where more than one individual is required, the annoted Figure 12 is attached. The number in ( ) indicates the average number of personnel expected to be assigned.

The staffing of Figure 13 will be accomplished by designating a primary and two alternate individuals to each of the key supervisory positions. For those functional blocks where more than one individual is required, the annotated Figure 13 is attached. The number in ( ) indicates the average number of personnel expected to be assigned.

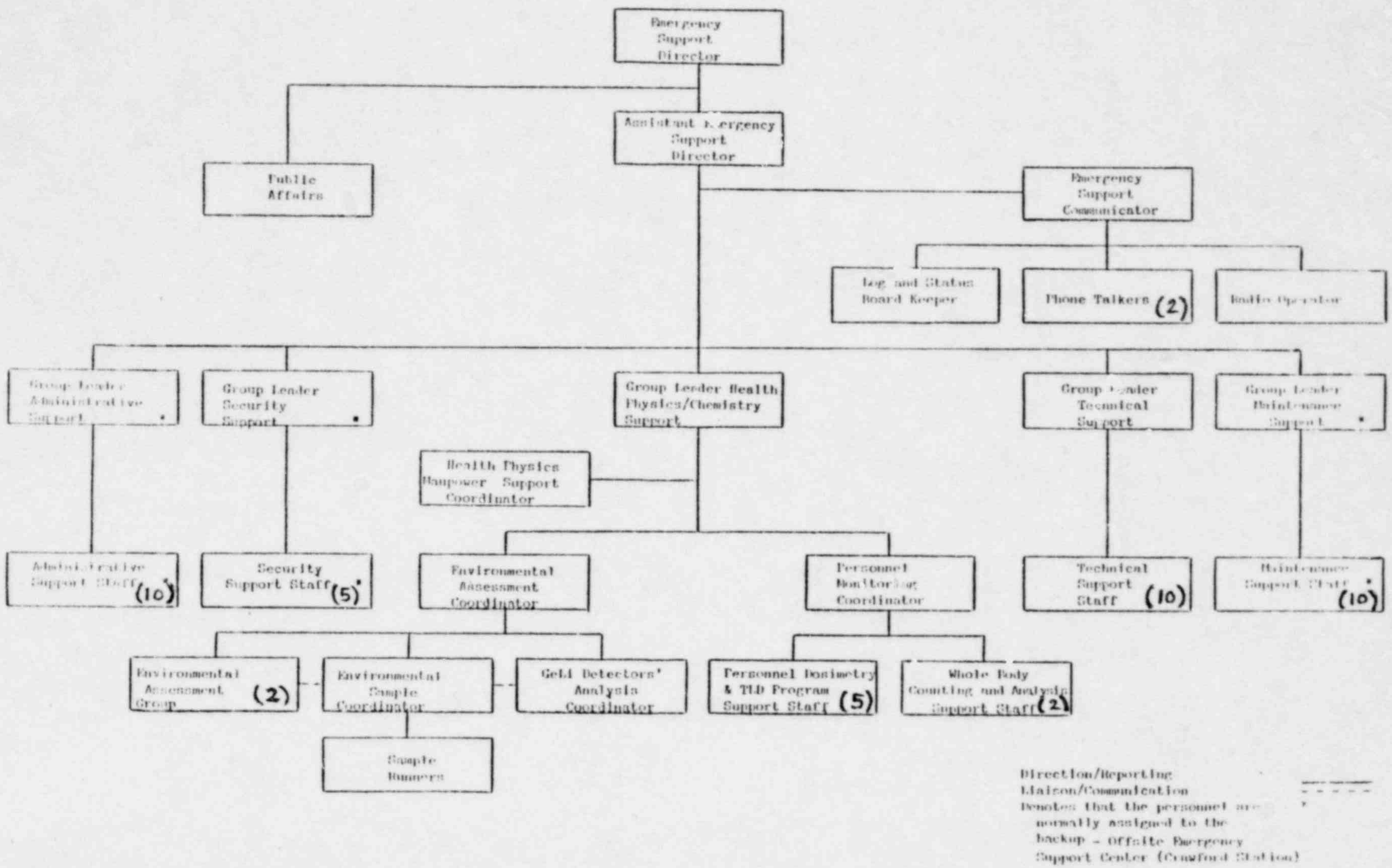


1438 256

Revision 0

ONSITE EMERGENCY ORGANIZATTON
Figure 12

October 1979



1438 257

OFFSITE EMERGENCY SUPPORT ORGANIZATION
Figure 13

QUESTION

83. Describe, for each of the functional blocks shown in Figures 12 and 13, the qualification requirements in terms of education background (discipline), experience, and training.

RESPONSE

Personnel assignments for the key positions in the emergency organizations of Figures 12 and 13 have been made such that these assignments are consistent with their normal position functions. Qualifications for current assignments are described in Section 5.

1438 258

QUESTION

84. Describe any interfaces between the Onsite Emergency Organization shown in Figure 12 and the Offsite Emergency Organization shown in Figure 13.

RESPONSE

Interfaces between the Onsite and Offsite Emergency Organizations are described in Sections 4.5.1.3 2.a, 2.b, 2.e; 4.5.1.4; and 4.5.1.4 1, 4. and 5.

1438 259

QUESTION

85. Describe the maximum time it will take from notification of those persons assigned to the emergency organizations (Figures 12 and 13) until they have been mobilized, and are performing their assigned duties.

RESPONSE

Table 8 in the TMI Unit I Emergency Plan (Section 4 of the Restart Report) describes the manpower and timing considerations for mobilizing the emergency organizations.

1438 260

QUESTION

86. Describe the Duty Sections referred to in Section 4.6.1.3 (pages 4-87 and 4-88).

RESPONSE

There will be three duty sections that will be on a "call out" rotation. The duty sections will be staffed by TMI station personnel and will be headed by a Duty Section Superintendent. The assignment of duty section personnel will be formal, procedurized, and controlled such that a full complement of personnel is always available in case of emergencies. In addition, each individual on the duty section will have a designated emergency organization position consistent with staffing Figure 12 in the TMI Unit 1 Emergency Plan. This specific designation will also appear in the duty section roster sheet.

1438 261

QUESTION

87. Describe the training to be received by other than the station organization in regard to their duties in the Emergency Organization. Note Section 4.8.11, page 4-134.

RESPONSE

Table 12 of the TMI Unit 1 Emergency Plan describes the scope and frequency of the training required for all TMI Generation Group personnel with specific emergency organization assignments. These personnel may be members of the station organization or the station support organization. Section 4.8.1.1 has been revised to reflect this.

1438 262

QUESTION

88. Describe in greater detail your plans for the long-term recovery organization. This information should include the following:

A. The maximum number of persons you plan to staff each functional block shown.

B. The minimum qualification requirements in terms of disciplines (education background) and experience.

C. The source of personnel to staff this organization.

D. The proposed assignment of management personnel for each functional block.

E. The time frame to activate this organization and have them functioning.

RESPONSE

A. The following numbers indicate the approximate range of staffing levels. It should be pointed out that actual staffing levels will depend on the type of accident and scope of the recovery effort.

Public/Government Affairs	- (6-12)
Administration & Logistics	- (30-100)
Task Management & Scheduling	- (6-10)
Technical Working Group	- (10-12)
Industry Advisory Group	- (30-60)
Technical Support Group	- (20-60)
TMI-1 Operations Group - normal station staff	
Waste Management Group	- (10-30)
Plant Modifications Group	- (10-30)

1438 263

B. | Title or Group | Qualifications |
|---|---|
| Recovery Operations Manager | Senior Vice President Met-Ed (Senior TMI Generation Group Officer) |
| Public/Government Affairs | Company Officer |
| Administration and Logistics | Company Officer |
| Industry Advisory Group | Industry technical experts as selected by the President-GPU |
| Technical Support Group | Qualifications consistent with providing the services described in Section 5.3. |
| TMI Unit 1 Operations Group | Qualifications described in Section 5.2 |
| Waste Management Group | Key people will be drawn from the Technical Functions Group |
| Plant Modifications Group | Consistent with the professional level required to meet design specifications, codes and standards |

C. Personnel to staff this organization will be drawn from the TMI Generation Group Station and Station Support Organization, consultants, other utilities, architect engineers, vendors, etc.

D. President GPU - incumbent
Recovery Operations Manager - Senior Vice President Met-Ed/Vice President GPUSC
Public/Government Affairs - Vice President Communications GPUSC
Administration & Logistics - Vice President Operations - Met-Ed
Task Management & Scheduling- Manager Management Services
Technical Support Group - Director - TMI-2
TMI-1 Operations Group - Vice President Met-Ed
Waste Management Group - Director TMI-2 Site Operations
Plant Modifications Group - Director Projects - GPUSC
Technical Working Group - The above group leaders plus representatives from NRC, the Architect Engineer, the NSSS Supplier and the Industry Advisory Group

E. Section 4.5.1.5 indicates that a deliberate shift to a long term recovery organization will occur if a complicated or long term recovery operation is anticipated. This shift will occur when conditions have stablized. The offsite emergency support organization in Figure 13 is structured to be compatible with the long term recovery organization. Once the decision is made to shift the organization into the long term recovery mode, the functional blocks in the Figure 13 organization can expand into the functional blocks in the long term organization (e.g. Public Affairs → Public/Government Affairs, Group Leader Technical Support → Technical Support Group).

The time frame to activate this organization, assuming that it is required, will vary based on the time it takes plant conditions to stablize. There is no need to rapidly organize into this structure since the Figure 13 organization contains all the essential elements of offsite support and can be staffed in a time frame described in Table 8.

QUESTION

89. Describe your provisions for keeping your emergency plans up to date, including assignments of personnel, their availability, and training programs, as necessary.

RESPONSE

Section 4.8 describes the methods for updating the emergency plan and the training programs. Section 4.5.1.3 describes the assignment of personnel to the emergency organization.

1438 266

QUESTION

90. Assure that your definition of "Safety Grade" in Section 1.4 of the Restart Report which identifies "redundancy" as a required feature, meets NRC single failure criteria including separability.

RESPONSE

The term "redundancy" is intended to mean single failure proof and separation in accordance with the TMI-1 FSAR separation criteria

1438 267

QUESTION

91. The following additional information is required concerning NUREG-0578 item 2.1.5.B - Hydrogen Recombiners:

A. You indicate that you have performed hydrogen generation rate calculation in accordance with Regulatory Guide 1.7 in order to verify that adequate time is available following an accident to install the second recombiner and still maintain the containment atmosphere hydrogen concentration with acceptable limits. Submit these calculations for our review. Indicate where the second hydrogen recombiner is to be stored prior to being used and verify that adequate procedures are available and training performed to assure that the second recombiner can be properly and expeditiously installed.

B. You indicate that you will perform an evaluation to demonstrate that potential leakage and discharge to the atmosphere of the Intermediate Building air used for recombiner cooling will not result in off-site dose releases in excess of 10 CFR 100 limits. Submit this evaluation for our review.

RESPONSE

A. We have stated that the design basis for the system is a LOCA with hydrogen generation rates calculated in accordance with Reg. Guide No. 1.7.

We have requested one of our consultants, Pickard, Lowe and Garrick, Inc., to evaluate and confirm that the recombiner which had been purchased for TMI Unit No. 2 is of sufficient capacity for TMI Unit No. 1. Note that the TMI-2 recombiner sizing is based on calculations in accordance with Reg. Guide No. 1.7 which indicate that the 3% by volume hydrogen concentration occurs approximately 250 hours after a LOCA. Since TMI-1 and 2 have approximately the same available free containment space and they both have the same size reactors, the time at which containment hydrogen concentrations reach the 3% level are approximately the same. Based on the 250 hours at which time the recombiners are required to oper e and the Standard Review Plan Section 6.2.5, Paragraph II.12e requirement, the stored recombiner must be made available to perform its function in a time period that is equal to or less than one-half of the 250 hours. Since the redundant recombiner will be stored in a seismic class I structure on the TMI site and all the connections for its use will be permanently installed, there seems to be no reason why the redundant recombiner cannot be made available to perform its function within 125 hours after a LOCA.

A copy of the calculations performed by our consultant will be submitted to you as soon as we are in its receipt, which is expected by November 30, 1979.

Adequate procedures for the maintenance and installation of the redundant recombiner will be made available to you as soon as they are completed. Training will be performed to assure that the second recombiner can be properly and expeditiously installed.

1438 268

B. The hydrogen recombiner system is an engineered safeguards welded piping system with the exception of the recombiner connection and flow element flanges. After installation of the redundant recombiner, the piping system shall be pneumatically tested to demonstrate that the leakage is within acceptable limits in order to limit the off-site releases to below the 10CFR100 allowables. Since any possible leakage from the closed piping system will be insignificant compared to the total allowable post LOCA containment leakage which yields releases below the 10CFR100 allowables, we conclude that exhaust of the recombiner cooling air which would include the recombiner piping leaks would also be below the 10CFR100 limits.

1438 269

QUESTION

92. The Met-Ed response to Question 17 (Supplement 1 to the TMI-1 Restart Report) identifies loss of natural circulation due to loss of heat sink as a condition to be analyzed for inadequate core cooling response. This analysis meets the requirements of NUREG-0578 but does not correspond to the analysis (DNB transient at power) proposed at the September 13, 1979 B&W Owner's Group meeting. Please identify those analyses of inadequate core cooling which will be performed independently, or in verification, of the generic analyses to be provided by B&W.

RESPONSE

The transient listed in our response to question 17 (loss of natural circulation due to loss of hint sink) was in error. This was intended to be consistent with the B&W transient to be analyzed (DNB transient at power). This listing has been corrected on the revised second page for the response to question 17, and a copy of this revision is attached.

The RETRAN analyses which will be conducted by GPUSC/Met-Ed are identified in Appendix A, Table 8A-1 of the Restart Report.

QUESTION

93. As requested by Question 18 (Supplement 1 to the TMI-1 Restart Report), please identify those emergency procedures which require the explicit use of (a) in-core thermocouples, (b) wide-range reactor outlet temperature measurement, (c) reactor coolant saturation temperature margin, and (c) other instrumentation identified in the response to Question 17 of the Supplement. Identify each instrument class separately.

RESPONSE

The following procedures have or are being revised to provide specific guidance on inadequate core cooling which require the use of:

a) In-core Thermocouples

EP 1202-6	Loss of Reactor Coolant/RC Pressure
EP 1202-2/2A	Station Blackout and Station Blackout With Loss of Both Diesels
EP 1202-39	Inadequate Core Cooling

b) Wide Range Reactor Outlet Temperature

EP 1202-6	Loss of Reactor Coolant/RC Pressure
EP 1202-26A	Loss of Steam Generator Feed to Both OTSG's
EP 1202-39	Inadequate Core Cooling

Although specific reference is not made to the wide range reactor outlet temperature in all procedure the assumption that reference to the wide range indication will be made if narrow range is exceeded.

c) Reactor Coolant Saturation Meter

EP 1202-4	Reactor Trip
EP 1202-6	Loss of Reactor Coolant/RC Pressure
EP 1202-26A	Loss of Steam Generator Feed to Both OTSG's
EP 1202-2/2A	Station Blackout and Station Blackout With Loss of Both Diesels
EP 1202-39	Inadequate Core Cooling

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OP 1102-16 Natural Circulation

EP 1202-14 Loss of RC Flow/RC Pump Trip

d) Other existing Instrumentation

EP 1202-4 Reactor Trip

EP 1202-6 Loss of Reactor Coolant/RC Pressure

EP 1202-14 Loss of RC Flow/RC Pump Trip

EP 1202-26A Loss of Steam Generator Feed to Both OTSG's

EP 1202-2/2A Station Blackout and Station Blackout With Loss of Both Diesels

EP 1202-39 Inadequate Core Cooling

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QUESTION

94. As requested by Question 20 (Supplement 1 to the TMI-1 Restart Report), please complete the description of the proposed subcooling meter. Include:

A. Overall display uncertainty

B. Display qualification

C. Information from page 2 of Table 20-1 (omitted from Submittal).

If the device utilizes an analog steam table approximation as stated, describe the process in sufficient detail to verify the applicability of the method. Include range and accuracy of the process.

RESPONSE

See revised table 20-1 of the response to Question 20 in Supplement 1, Part 1.

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TABLE 20-1

Information on the Subcooling Meter
Plant Name: Three Mile Island, Unit No. 1
Vendor: Foxboro (Calculator)

Reference for Information: ______________________

Display

Information Display	Tsat Margin, Psat Margin
Display Type	Digital
Continuous or on Demand	Tsat margin continuous, Psat margin on demand
Single or Redundant Display	Single (Selectable)
Location of Display	Control Room Console
Alarms	Low Tsat margin (set point later)
Overall uncertainty	(Later)
Range of Display	0-1000°F, 0-1000 psi
Qualifications	(Not yet specified)

Calculator

Type	Analog
If process computor is used specify availability	Not applicable
Single or redundant calculators	Redundant
Selection Logic	Highest T.
Qualifications	(Seismic, Environ, IEEE 279
Calculation Technique	Steam Table Approximation

For computing saturation temperature or pressure, the steam table saturation curve will be synthesized with 0.5% accuracy by means of a "Signal Characterizer" module. This is a solid state function generator which has the capability of simulating the characteristic curve of a process by means of a number of straight line segments. Up to eight segments may be used. The slope and intersection of each segment are individually adjustable.

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Input

Temperature	RTD's
Temperature	4 sensors, T hot
Range of Temperature sensors	120^{o} - $920^{o}F$
Uncertainty* of temperature sensors (^{o}F at 1)	(Later)
Qualifications (seismic, environmental, IEEE 323)	Control Grade (Short Term)
Pressure (Specify instrument used)	Foxboro E11GH
Pressure (number of sensors and locations)	Two, PZR Press.
Range of Pressure sensors	0-2500 PSIG
Uncertainty* of pressure sensors (PSI at 1)	(Later)
Qualifications (seismic, environmental, IEEE323)	Seismic, Environmental

Backup Capability

Availability of Temp & Press	Individual press. & temp. indic. available in Control Room
Availability of Steam Tables etc.	Steam tables available on control panel and in computer
Training of operators	In accordance with operator retraining program
Procedures	Procedures will be generated for use of instruments
Other	Psat & Tsat will be independently computed by plant computer.

*Uncertainties must address conditions of forced flow and natural circulation.

QUESTION

95. Paragraph 2.1.3.b of NUREG-0578 requires a description of further measures and supporting analyses that will yield more direct indication of low reactor coolant level and inadequate core cooling such as reactor vessel water level instrumentation. Section 2.1.1.6 of the Restart Report does not address further measures (to be implemented by January 1, 1981), nor does it address the question of reactor vessel water level instrumentation. Provide a conceptual description of what additional measures will be taken to detect inadequate core cooling. Provide an implementation schedule for these changes.

RESPONSE

Babcock & Wilcox is currently evaluating the need for further measures that will yield more direct indication of low reactor coolant level and inadequate core cooling. This evaluation will cover all the inadequate core cooling evaluation cases and is scheduled for completion on December 14, 1979. Babcock & Wilcox has scheduled to provide recommendations for additional instrumentation (fi any) by February 1, 1980. It is our intent to rely on this generic B&W evaluation. The conceptual description of additional measures to detect inadequate core cooling and an implementation schedule for any required changes will not be available until February, 1980, at the earliest.

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