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SUBJECT:
RESPONSE TO NRC REQUEST DURING SYSTEMATIC EVALUATION REVIEW OF JUNE 26 - 28, 1978 . FORWARDING PLAN AND EVALUATION DRAWINGS OF THE REACTOR BLDG SUMP. W/ATT PAGES RE SECTIONS 6 & 11 OF THE FINAL SAFETY ANALYSIS REPT.

PLANT NAME: THREE MILE ISLAND - UNIT 1

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July 7, 1978
GQL 1154

Director of Nuclear Reactor Regulations
Attn: R. W. Reid, Chief
Operating Reactors Branch No. 4
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Si:

Three Mile Island Nuclear Station, Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289

In response to a request by Mr. J. J. Watt during the Systematic Evaluation Program Review Team visit to TMI-1 on June 26 - 28, 1978, enclosed please find plan and elevation drawings of the Reactor Building Sump. Also enclosed are copies of pages from Section 6 of the Final Safety Analysis Report with information on RB sump level following a LOCA and NPSH data for the DHR pumps, and a page from Section 11 of the FSAR concerning the RB sump grating.

It would be appreciated if a copy of the final report be forwarded to Metropolitan Edison.

Sincerely,

J G Herbein

J. G. Herbein
Vice President-Generation

JGH:WSS:tas

Enclosures

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A001
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1/1
REDUCE
DRWG

d. Opening the emergency cooling coil isolation valve on the outlet side of the coil. Inlet valves are normally open for leak monitoring purposes.

e. Closing the normal cooling coil isolation valve.

6.4 ENGINEERED SAFEGUARDS LEAKAGE AND RADIATION CONSIDERATIONS

6.4.1 INTRODUCTION

The use of normally operating equipment for engineered safeguards functions and the location of some of this equipment outside the reactor building requires that consideration be given to direct radiation levels from the fluids circulating in these systems and the leakage from these systems after fission products have accumulated.

The shielding for components of the engineered safeguards is designed to meet the following objectives in the event of a maximum hypothetical accident:

a. To provide protection for personnel to perform all operations necessary for mitigation of the consequences of the accident.

b. To provide sufficient accessibility in all areas around the station to permit safe continued operation of the required equipment.

6.4.2 SUMMARY OF POST-ACCIDENT RECIRCULATION

Following a postulated reactor coolant system rupture, flow is initiated in the makeup and purification and decay heat removal systems from the borated water storage tank to the reactor vessel. Flow is also initiated by the reactor building spray system to the building spray headers. When the borated water storage tank inventory is exhausted, recirculation from the reactor building sump is initiated by the operator for both the decay heat removal flow and the reactor building sprays. The post-accident recirculation flow paths include all piping and equipment external to the reactor building as shown on Figures 6-2 and 6-3 up to the valves leading to the borated water storage tank.

The NPSH available to the decay heat and reactor building spray pumps during the post LOCA recirculation phase has been calculated based on:

a. "issued for construction" piping drawings,

b. pipe and fitting losses calculated using the information in Crane Technical Paper No. 410,

c. total flow in a single string (i.e., consisting of one decay heat and one reactor building spray pump served by a single sump suction line) is 4500 gpm or 3000 gpm to the decay heat pump and 1500 gpm to the reactor building spray pump,

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d. sump water temperatures and reactor building pressures given in Table 6-11,

e. water level in the reactor building sump is 284 ft 6 in.,

f. R.B. spray pump shaft center line at elevation 262 ft 3 in.,

g. decay heat pump shaft center line at elevation 263 ft 7 in.,

The NPSH's available to each pump, calculated from the above data, are given in Table 6-11. The minimum calculated NPSH's available to each pump, assuming a saturated sump, are compared below with the NPSH's required.

Item	Pump: Decay Heat	Pump: R. B. Spray
Flow Rate, gpm	3,000	1,500
Maximum Head Loss in Suction Piping, ft H_2O	7.5	7.9
NPSH, ft H_2O		
Available	13.4	14.4
Required	10.5	13.8

The required NPSH's indicated above reflect measured test values. The certified performance test data for R.B. spray pump BS-P1A are shown in Figure 6-8A. In addition, the pump performance will be verified prior to fuel loading (see page 13A-5 for Decay Heat Removal System Functional Test and page 13A-7 for Reactor Building Spray System Functional Test).

In the event that the available NPSH is considered insufficient while operating on recirculation from the reactor building sump, the decay heat flow rate can be reduced from the control room by throttling the valves DH-V4A and DH-V4B and/or the reactor building spray flow rate can be reduced from the control room by throttling valves BS-V1A and BS-V1B. There is "jog" close valve control in the control room for these four valves. If finer control or long term flow throttling of the decay heat flow rate is desired, manually operated globe valves DH-V19A and DH-V19B (normally wide open) may be utilized to achieve this. Extension handles on these two valves permit their operation from the 305 ft floor elevation in the auxiliary building.

Table 6-11
Reactor Building Conditions for the Design Accident With Two Reactor Building Air Coolers and 1500 GPM Spray

Period	Time, seconds	Reactor Building Total Pressure, psig	Reactor Building Steam Pressure, psia	Vapor Temp, F	Sump Temp, F	DH Pump NPSH Available, ft	BS Pump NPSH Available, ft	DH Pump NPSH Available Saturated Sump,[a] ft	BS Pump NPSH Available Saturated Sump,[a] ft
Initial (Before Accident)	0	0	0	110	--	--	--	--	--
Beginning of Recirculation	4,200	3.9	2.7	138	210	24.1	25.1	13.4	14.4
Maximum Recirculation Sump Temperature	8,400	5.4	3.9	152	218	22.1	23.0	13.4	14.4
Maximum Recirculation Vapor Temperature	11,800	5.7	4.1	154	213	26.4	27.4	13.4	14.4

[a] Reactor building total pressure assumed to be equal to the saturation pressure of the sump water (boiling sump assumption).

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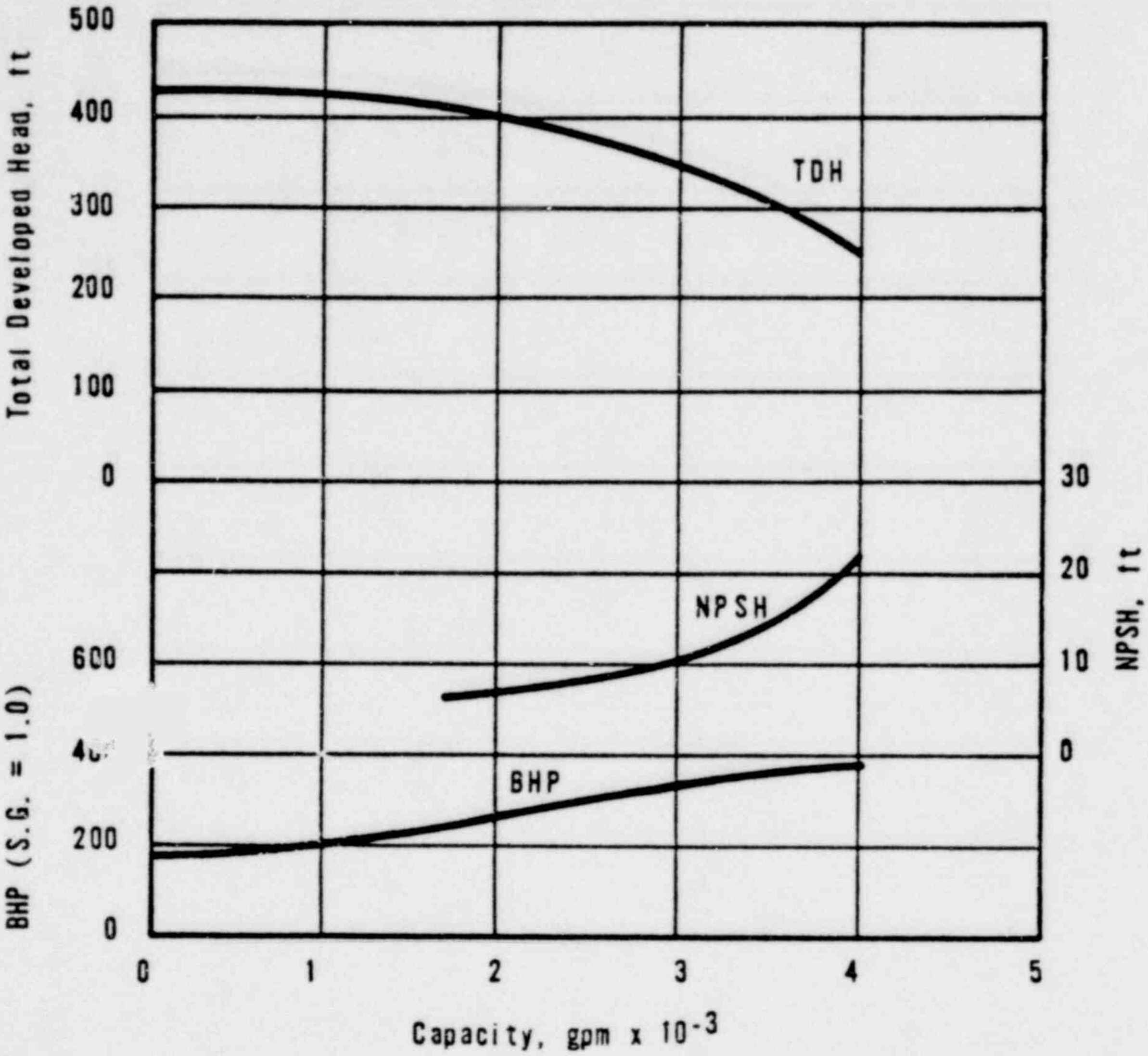

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DECAY HEAT REMOVAL PUMP CHARACTERISTICS
THREE MILE ISLAND NUCLEAR STATION UNIT 1

FIGURE 6-6

(AM. 25 4-10-72)

Table 11-4
Disposal System Component Data

Sheet [illegible]

SUMPS

NAME/FUNCTION	TYPE	CAPACITY, FT^3 Full (to 1' from top)	SUMP DEPTH	SUMP PUMPS	SUMP PUMP CAPACITY GPM	SUMP PUMP CAPACITY Head, Ft	COMMENTS
Reactor Building/collects water from floor and misc. equip. drains within reactor building.	S.S. lined concrete pit	1,170	7'-6"	None Gravity drain via 6" line	-	-	Grating box keeps particulates out of outlet lines. Drains to Auxiliary Building Sump
Auxiliary Building/collects water from floor & misc. equip. drains within Auxiliary and Fuel Handling Buildings	S.S. lined concrete pit	1,380	7'-10"	WDL-P-5A WDL-P-5B	150 150	75 75	Auxiliary Bldg. Sump pumps are S.S. construction and CLASS I Seismic Design. Sump pump out prevents flooding engineered safeguards components.
Spent Fuel Pit Room/collects water from floor and misc. equipment drains in Fuel Handling Building	S.S. lined concrete pit	204	6'-8"	WDL-P-2A WDL-P-2B	20 20	35 35	
Heat Exchanger Vault/collects ground water seepage and water from misc. floor and equipment drains within HX vault	S.S. lined concrete pit	120	6'-0"	WDL-P-3A WDL-P-3B	20 20	35 35	Not considered part of liquid waste system. Water normally collected is that from groundwater seepage, the river or the closed cooling system. Remote potential for contamination.
Borated Water Tank Tunnel/collects ground water seepage into borated water tank tunnel	Concrete pit lined with protective coating	42	3'-0"	WDL-P-4A WDL-P-4B	20 20	35 35	Not considered part of liquid waste system. Remote potential for contamination.
Tendon Access Gallery/collects ground water seepage into tendon access gallery	Concrete pit lined with protective coating	27	2'-6"	WDL-P-1A WDL-P-1B	20 20	60 60	Not considered part of liquid waste system. Remote potential for contamination.

TOTAL LIQUID CAPACITY IN LIQUID WASTE SYSTEM SUMPS - 2,754 ft^3

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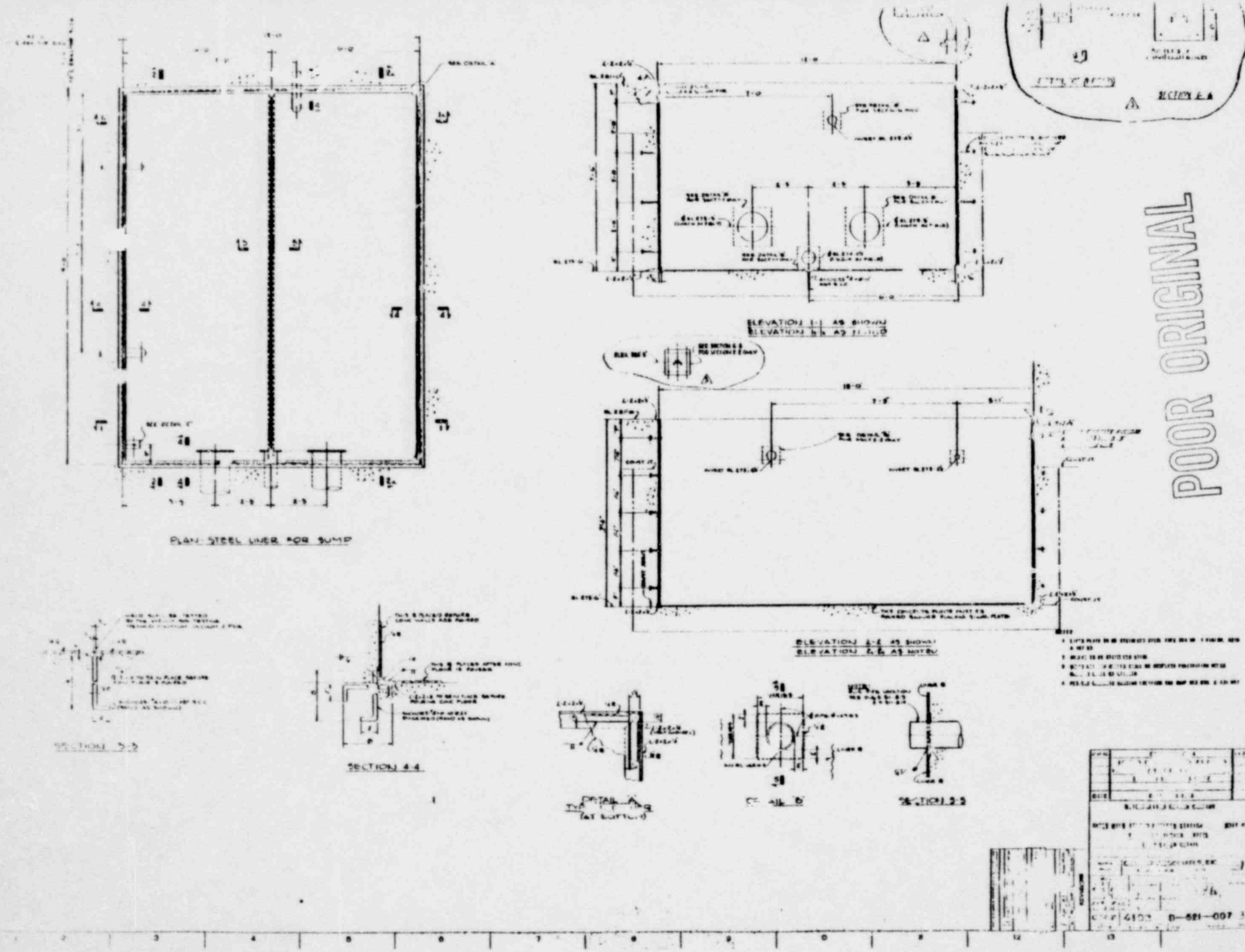

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