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POWER SYSTEMS

CONFIDENTIAL COUNSEL ONLY

Enclosure (1) provides generic design differences between a typical C-E operating plant and B&W Three Mile Island Unit 2 (TMI). We are providing this material as background, knowing of your need to respond to regulatory inquiries. Since this information is generic and was gathered from services at Windsor, we recommend that you check the specific validity with respect to your unit.

Part a of Enclosure (1) provides a listing of design features of C-E operating plants which would tend to mitigate an event similar to the Loss of Feedwater Event which occurred at TMI. Such design features in conjunction with improved operating and emergency procedures will give confidence that an incident similar to the TMI event has a very low probability of occurrence at a C-E designed operating unit. C-E will continue to evaluate pertinent design features and promptly recommend any design modifications and/or procedural changes which enhance operating plant safety.

Part b of Enclosure (1) provides system descriptions which form the basis for the conclusions stated in Part a. Part c of Enclosure (1) provides a C-E engineering evaluation of the sequence of events which would transpire if an event similar to the TMI Loss of Feedwater incident were to occur on a C-E operating plant. Part c of Enclosure (1) is a generic description of expected C-E plant response to events similar to those experienced at TMI.

Combustion Engineering stands ready to assist you in responding to I.E. Bulletin 79-06. If you should require such assistance from C-E, please contact me.

Sincerely,

P. W. Kruse
Manager, Engineering Services

PWK:mtm

ENCLOSURE (1)

C-E/B&W

PRELIMINARY GENERIC PLANT COMPARISON

The information presented in this enclosure is based on preliminary information concerning the nuclear incident at Three Mile Island Nuclear Power Plant Number 2. The results stated in this enclosure are based on C-E engineering judgement of events which transpired during the accident. The information and material discussed and presented by C-E with respect to the Three Mile Island Nuclear Power Plant design has been obtained from the Preliminary and Final Safety Analysis reports and personal knowledge.

GENERIC DESIGN FEATURES WHICH DIFFER BETWEEN C-E AND TMI-2

DESIGN FEATURE	COMMENT
a) C-E plants have a low steam generator water level RPS trip. For a loss of feedwater, B&W plants trip on high primary system pressure. The B&W trip signal also lifts the pressurizer relief valves.	The combination of a low steam generator water trip and higher steam dump capacity prevent significant primary system pressure excursions on C-E plants.
b) C-E plants have 40% steam dump/turbine bypass capacity. B&W plants have 15% steam dump/turbine bypass capacity.	
C-E steam generators have a larger secondary water inventory than the TMI steam generators.	There is more time available to establish emergenc feedwater after loss of main feedwater in the C-E system and less likelihood of lifting a primary relief valve.
C-E core elevation is relatively lower in the NSSS layout than TMI core.	Only 20-25% of RCS fluid is needed to cover C-E core vs 40-45% of RCS fluid for TMI.
C-E steam generator elevation does not form a cold leg loop seal.	A significant fraction of the TMI RCS fluid, once there is void in the RCS, will collect in the steam generator/cold leg loop seal.
C-E steam generator elevation is higher than TMI steam generator.	The higher elevations result in a better potential to initiate and maintain natural circulation.
C-E U-tube steam generator design vs TMI once through straight tube (OTSG)design.	U-tube steam generator design allows reflux conden sation of RCS fluid in hot leg vs only cold leg return of condensed RCS fluid for OTSG design. U-tube steam generator also allows more water inve tory.
TMI steam generators have more primary side water inventory than the C-E steam generators.	There is more potential for water retention in the TMI SGs with consequent unavailability of the wate to cool the core.
The TMI design has a loop seal in the pressurizer surge line. The C-E design has no loop seal.	A surge line loop seal allows water or two-phase fluid retention in the pressurizer with the primar system voided. This will produce an indication th the reactor vessel is not voided based on a false pressurizer level indication. The C-E system without the surgeline loop seal is less susceptible to this false indication.

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	[illegible]	[illegible]
9.	The C-E pressurizer is at a higher elevations than the TMI pressurizer.	The increased elevation increases the likelihood of draining the pressurizer when the hot leg is voided
10.	The C-E HPSI shutoff head is lower than the TMI HPSI shutoff head.	The lower shutoff head will not result in lifting the pressurizer relief or safety vlaves if the HPSI pumps are left on.

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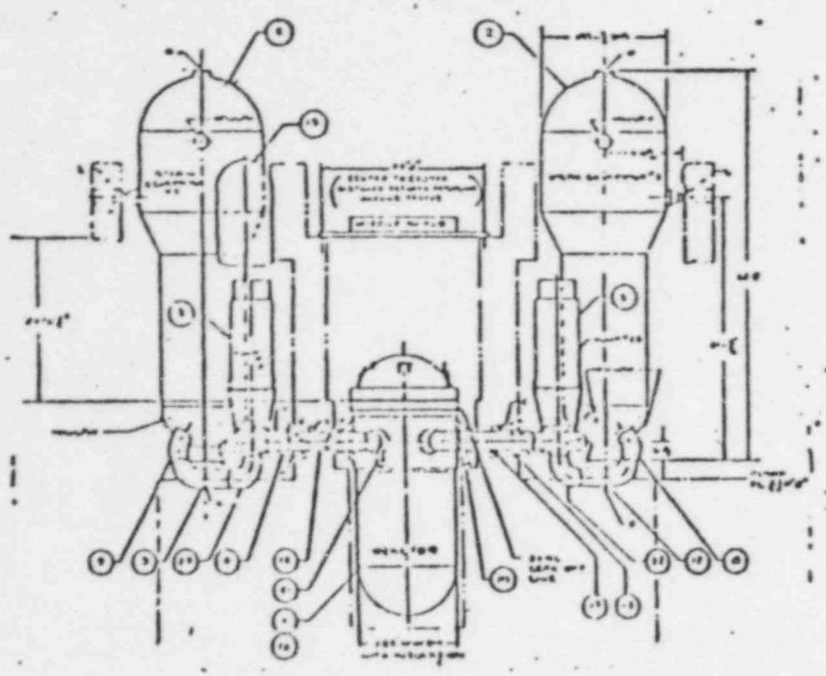
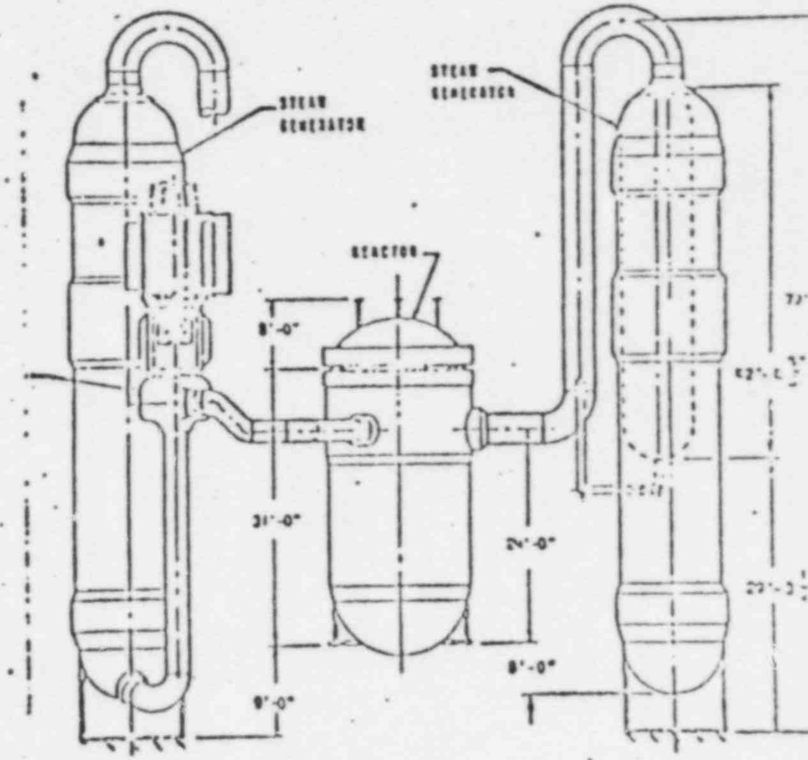
CE	B & W Three Mile Island
	

CE

Core Power	2570 MWt
Steam Pressure	[illegible] psig
Total Reactor Flow	359,576 gpm
Reactor Coolant Volume	
Reactor Vessel	3652 ft^3
Steam Generator	3053 ft^3
Pressurizer	1500 ft^3
Other	1513 ft^3
Total	9718 ft^3
Design Pressure	2485 psig
Operating Pressure	2235 psig
Design Temperature	650 °F
Operating Temperature	
Inlet	[illegible] °F
Outlet	[illegible] °F

B & W Three Mile Island

Core Power	2772 MWt
Steam Pressure	910 psig
Total Reactor Flow	[illegible] gpm
Reactor Coolant Volume	
Reactor Vessel	4010 ft^3
Steam Generator	2017 ft^3
Pressurizer	1500 ft^3
Other	[illegible] ft^3
Total	[illegible] ft^3
Design Pressure	2500 psig
Operating Pressure	[illegible] psig
Design Temperature	650°F
Operating Temperature	
Inlet	557°F
Outlet	[illegible]

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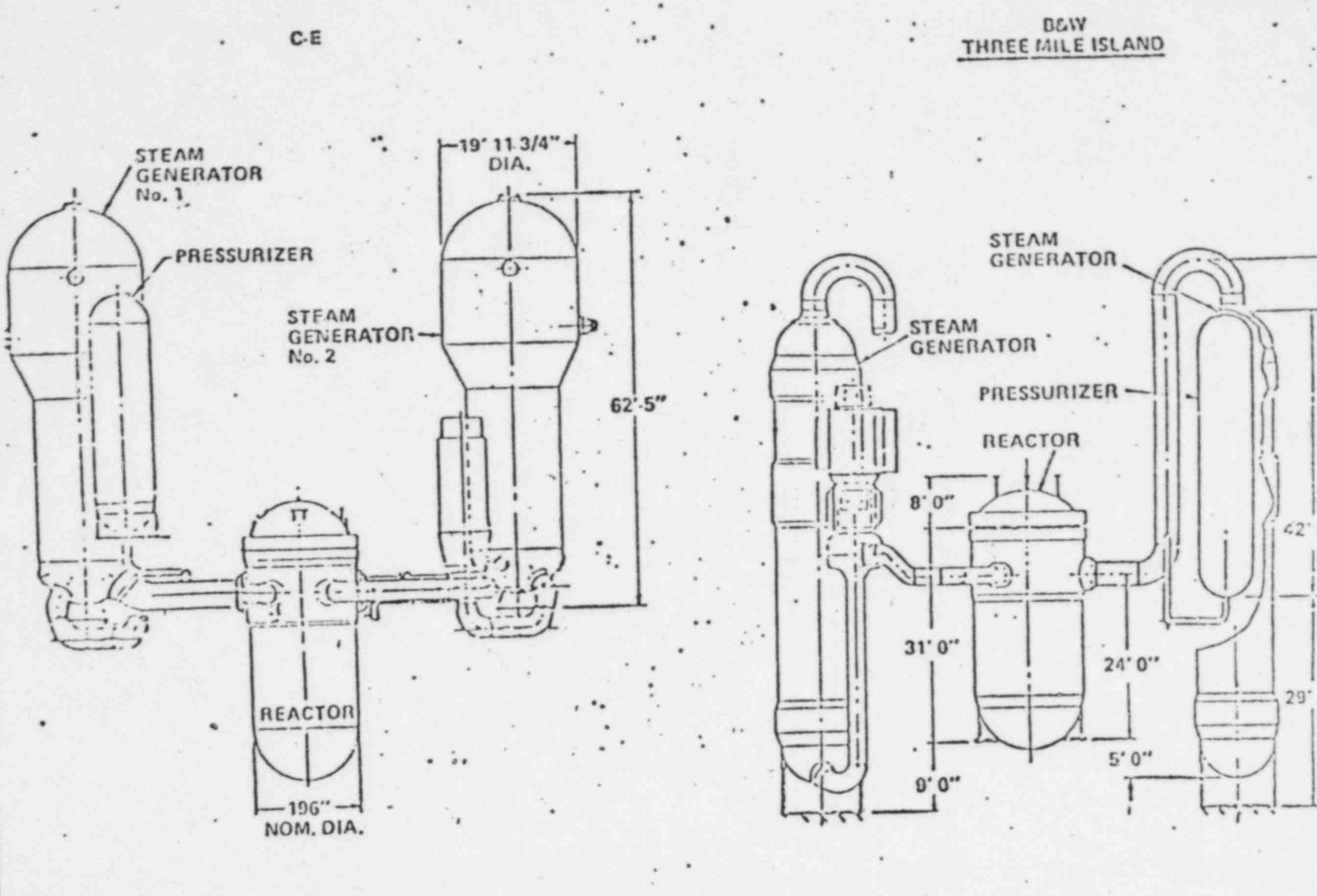

NOTE: DRAWN TO SAME RELATIVE SCALE

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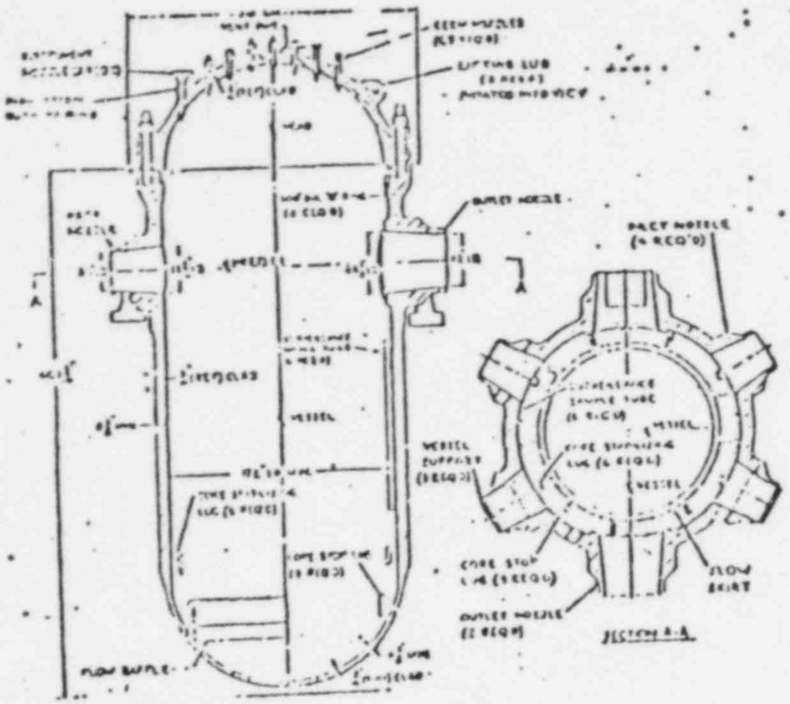
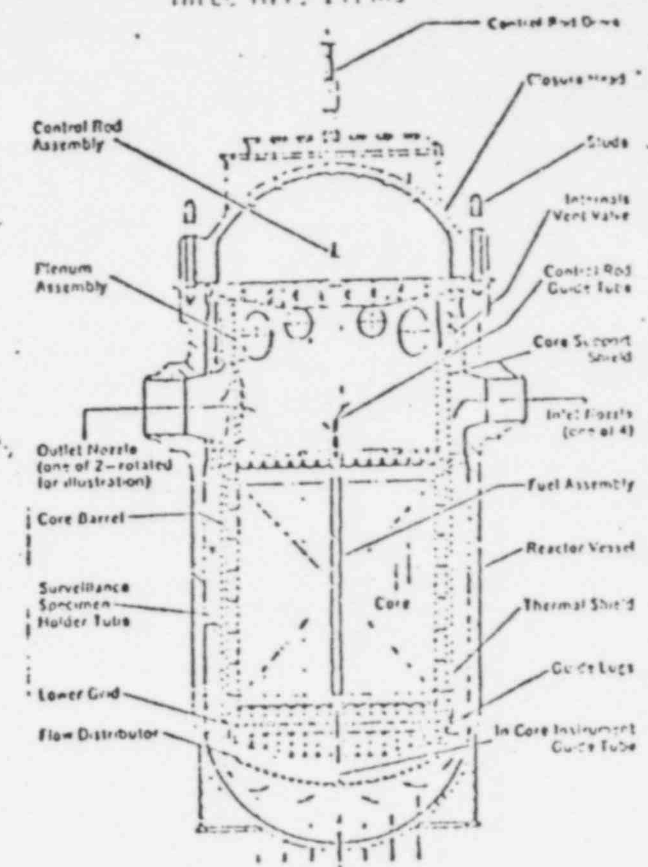


Reactor Vessel Volume Above Core

hot plenum from fuel plate to bottom of nozzle	810 $ft.^3$
CEA shrouds from fuel plate to bottom of nozzle	48 $ft.^3$
hot plenum nozzle region	314 $ft.^3$
CEA shrouds nozzle region	130 $ft.^3$
top of nozzle to UCS plate hot plenum	252 $ft.^3$
CEA shrouds and extension to head from top of nozzle bed	652 $ft.^3$
Total free volume above core	1728 $ft.^3$



Reactor Vessel Volume above Core

Upper plenum	775.9 $ft.^3$
Upper head	[illegible]
Total free volume above core	[illegible]

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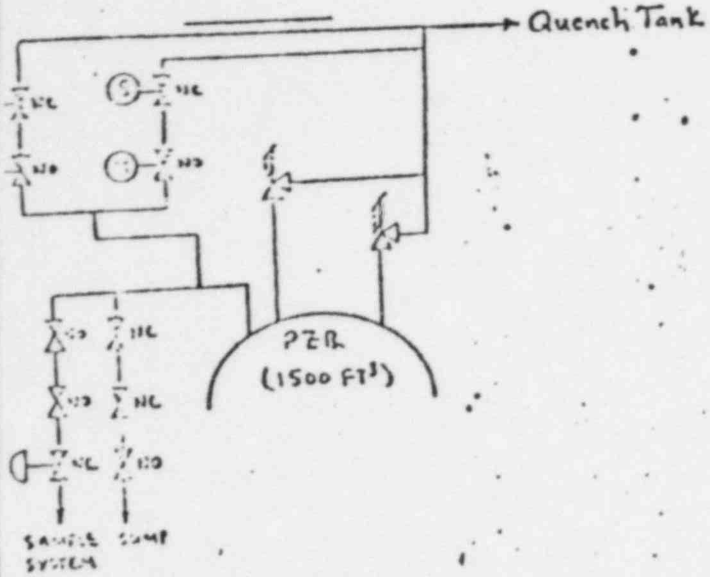


Quench Tank

Design: 100 psig/350 °F
Volume: 2[illegible] ft.³
Normal: 3 psig/120 °F/135 ft.³

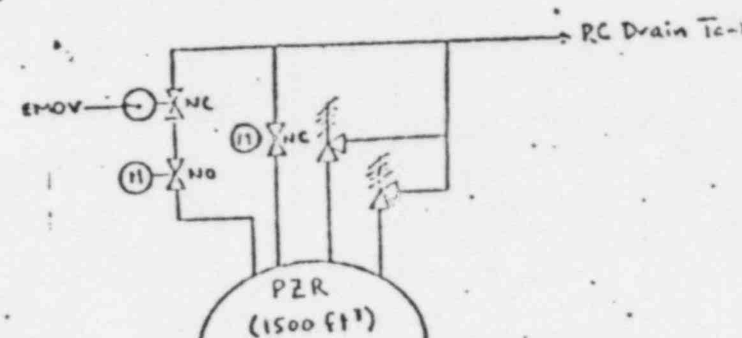
PORV

Number: 2
Capacity: 153,000 #/hr. (min)
Type: Solenoid Operated

Safeties

Number: 2
Capacity: 315,000 #/hr., 3% accumulation
Type: Spring loaded

B&W
Three Mile Island



Quench Tank (B & W: "RC Drain Tank")

Design 650 psig/300 °F
Volume 1000 ft.³
Normal:
Relief: 1 (a) 327,000 #/hr. @ 535 psi
1 (a) 342,000 #/hr. @ 550 psi

PORV

Number: 1
Capacity: ?
Type: Electromagnetic
Size: 2 1/2 inches

Safeties

Number: 2
Capacity: ?
Type: ?
Size: 3 inches

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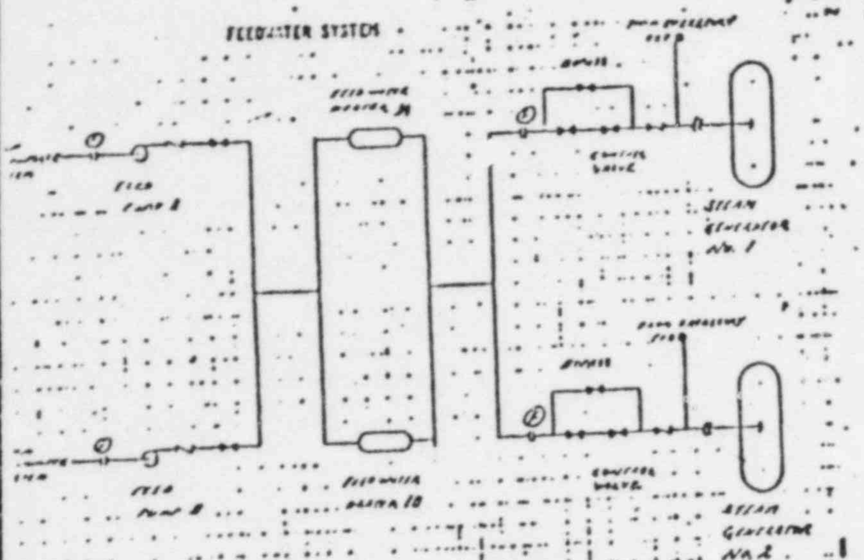
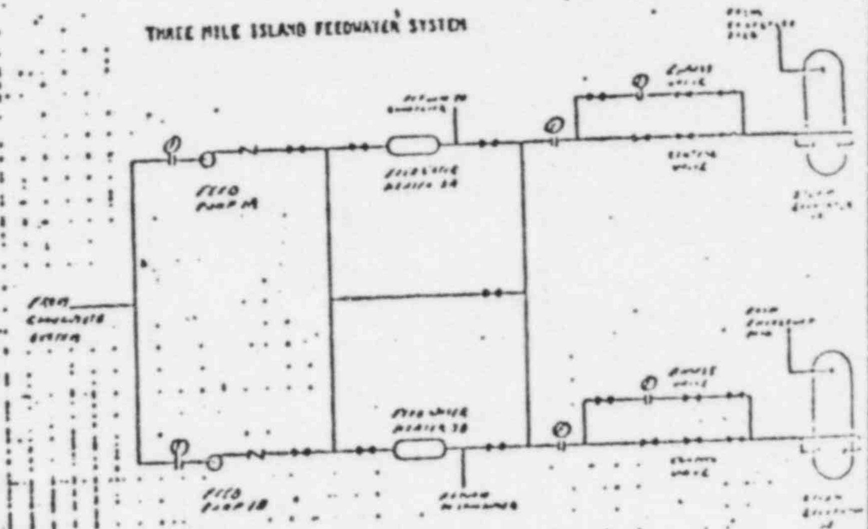
FEEDWATER SYSTEM



Steam Generator Pressure	920 psia
Plant Steam Rate	11.2 x 10^6 lbs/hr.
Feed Rate For Steam Generator	5.6 x 10^6 lbs/hr.
Feed Pump Rate	7.8 x 10^6 lbs/hr.
Feed Pump Rate As Percent of Full Plant Steam Rate	69%
Number of Feed Pumps	2
Steam Driven	

THREE MILE ISLAND FEEDWATER SYSTEM



Steam Generator Pressure	1000 psia
Plant Steam Rate	12 x 10^6 lbs/hr.
Feed Rate For Steam Generator	6 x 10^6 lbs/hr.
Feed Pump Rate	6.6 x 10^6 lbs/hr.
Feed Pump Rate as Percent of Full Plant Steam Rate	55%
Number of Feed Pumps	2
Steam Driven	

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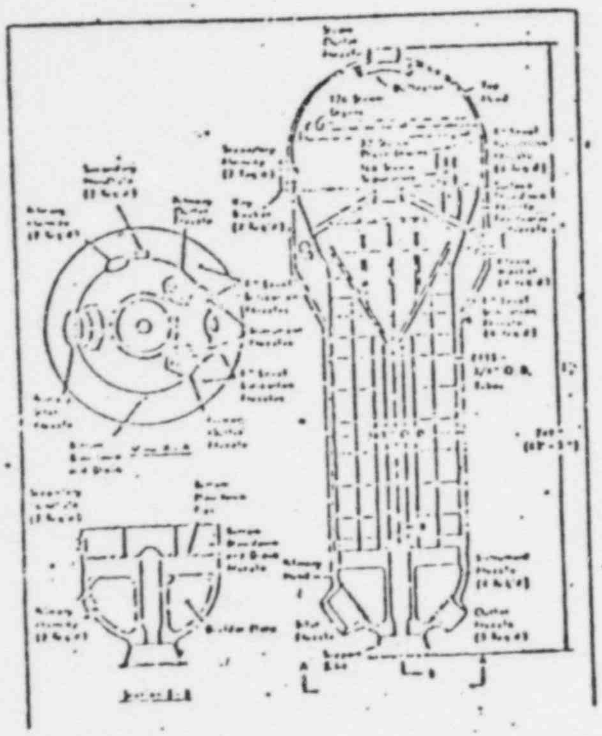

[illegible]	Value	Parameter	Value
[illegible]	[illegible]		
[illegible]	[illegible]		
[illegible]		[illegible]	[illegible]
[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]	[illegible]

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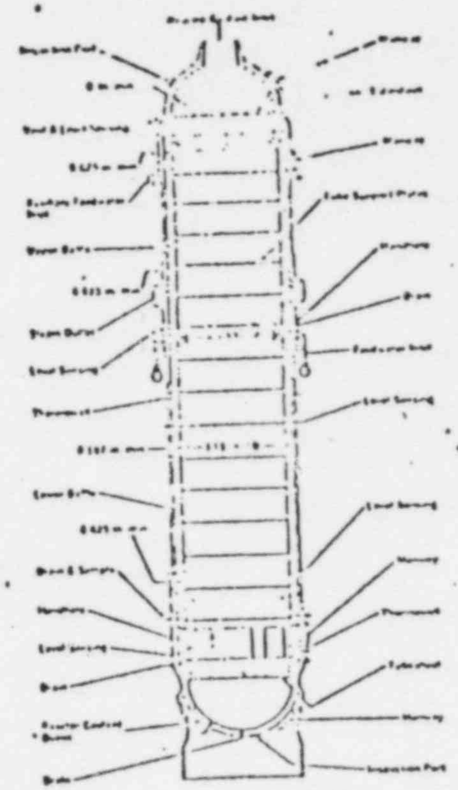
Three Mile Island



[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]		
[illegible]	[illegible]		
[illegible]		[illegible]	
[illegible]	[illegible]	[illegible]	[illegible]
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[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]	[illegible]
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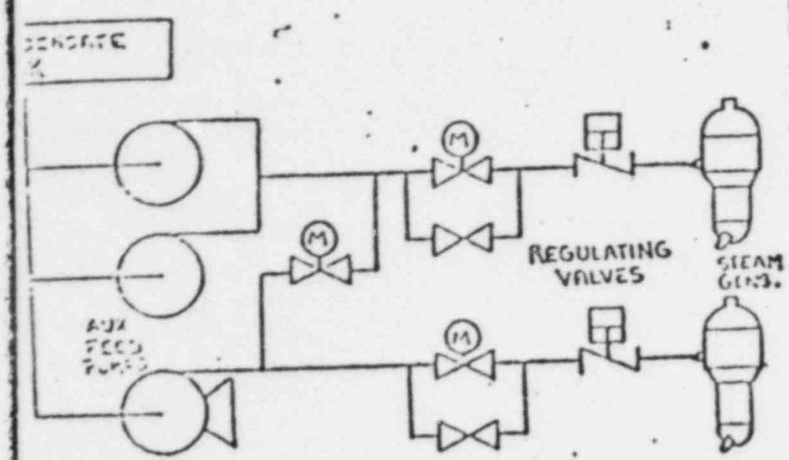
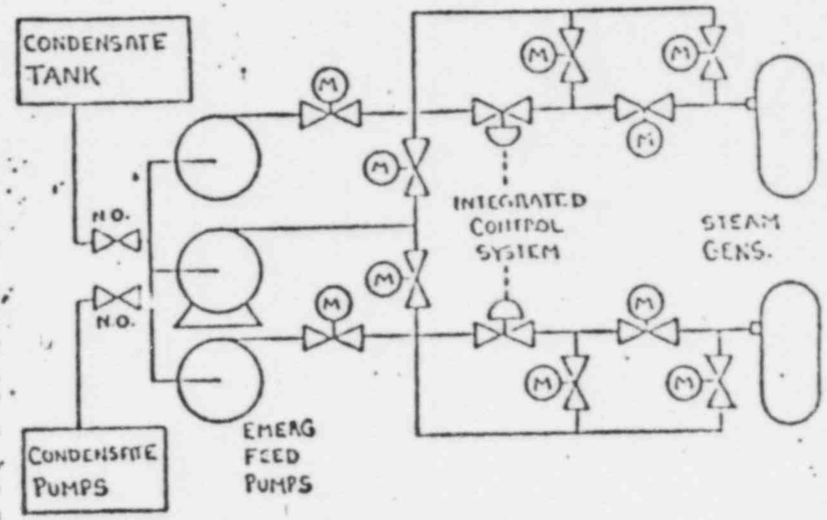
Three Mile Island





AUXILIARY FEEDWATER

ACTUATED BY:	MANUAL
CONTROLLED BY:	MANUAL
AUXILIARY FEED PUMPS:	1 TURBINE DRIVEN 2 MOTOR DRIVEN
CAPACITY:	TURBINE DRIVEN - 100% DECAY HEAT MOTOR DRIVEN - 50% DECAY HEAT (Each)
SOURCES OF SUCTION:	CONDENSATE STORAGE TANK
EMERGENCY POWER:	YES
SEISMIC CLASS I:	NOT SPECIFIED

EMERGENCY FEEDWATER

ACTUATED BY:	LOSS OF ALL RCP'S LOSS OF BOTH MAIN FEEDWATER PUMPS LOW FEED-STEAM DIFFERENTIAL PRESSURE
CONTROLLED BY:	INTEGRATED CONTROL SYSTEM (IF AVAILABLE) MANUAL
EMERGENCY FEED PUMPS:	1 TURBINE DRIVEN 2 MOTOR DRIVEN
CAPACITY:	TURBINE DRIVEN - 100% DECAY HEAT MOTOR DRIVEN - 50% DECAY HEAT (EACH)
SOURCES OF SUCTION:	CONDENSATE PUMPS (IF OPERATING) CONDENSATE STORAGE TANKS
EMERGENCY POWER:	YES
SEISMIC I:	YES

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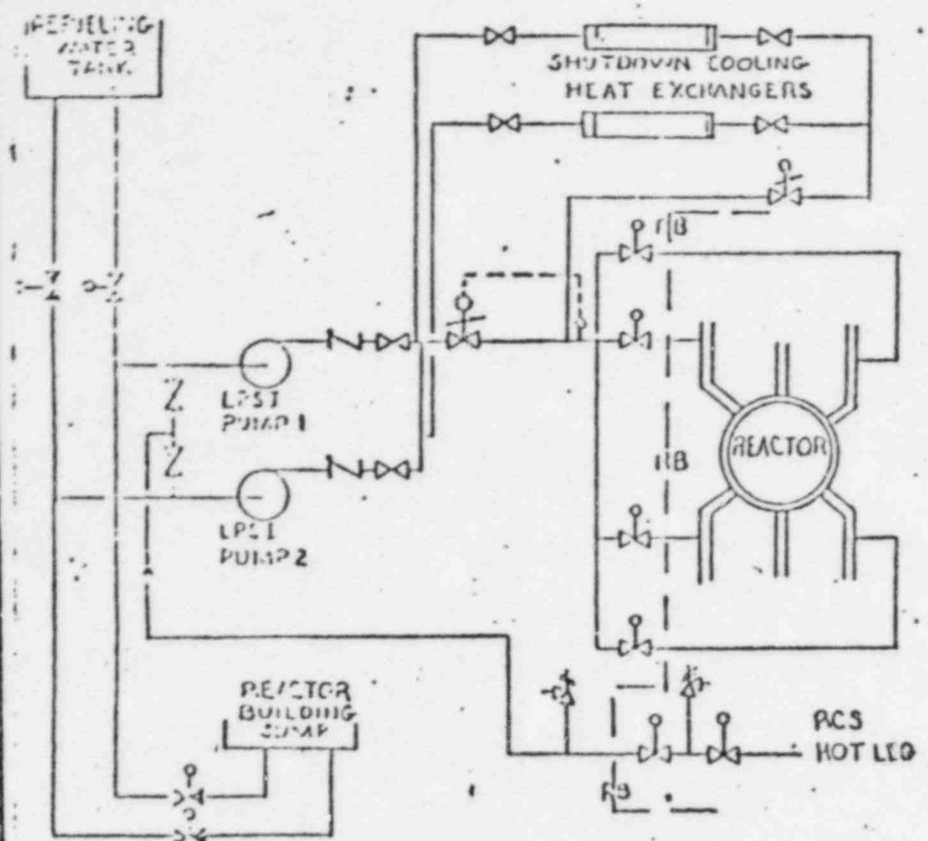


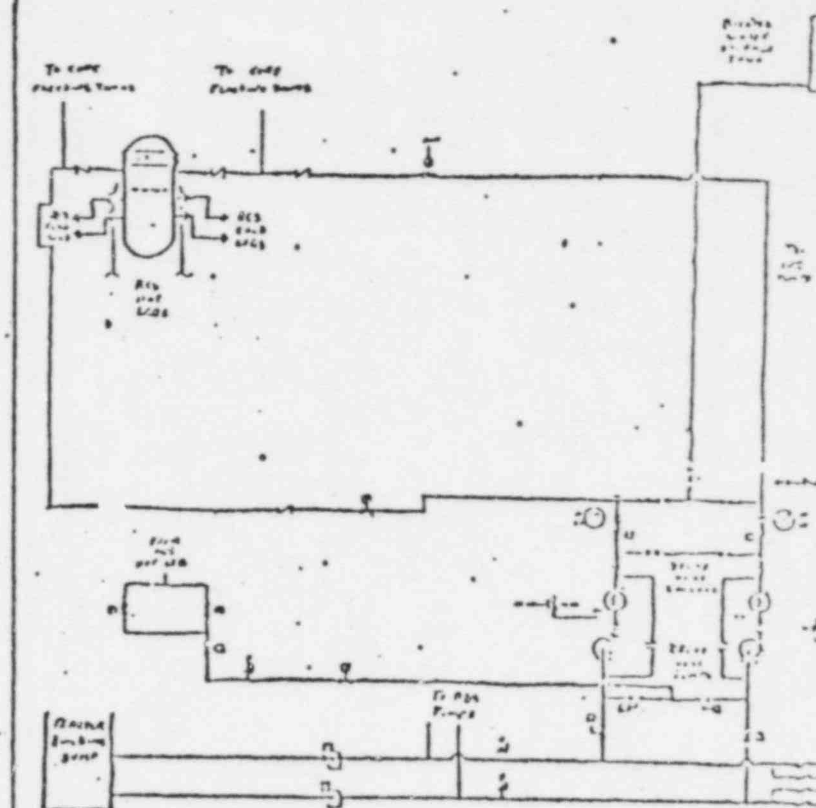
Shutdown Cooling Initiation Temperatures

Temperature, °F	300
Pressure, psig	250

Design Basis:

Cooldown RCS from 300 °F to 1[illegible] °F in 21.5 hours, starting 3 hours after shutdown.

B & W
Three Mile Island



Shutdown Cooling Initiation Temperatures

Temperature, °F	250
Pressure, psig	320

Design Basis:

Cooldown RCS from 250 °F to 140 °F in 16 hours, starting 6 hours after shutdown.

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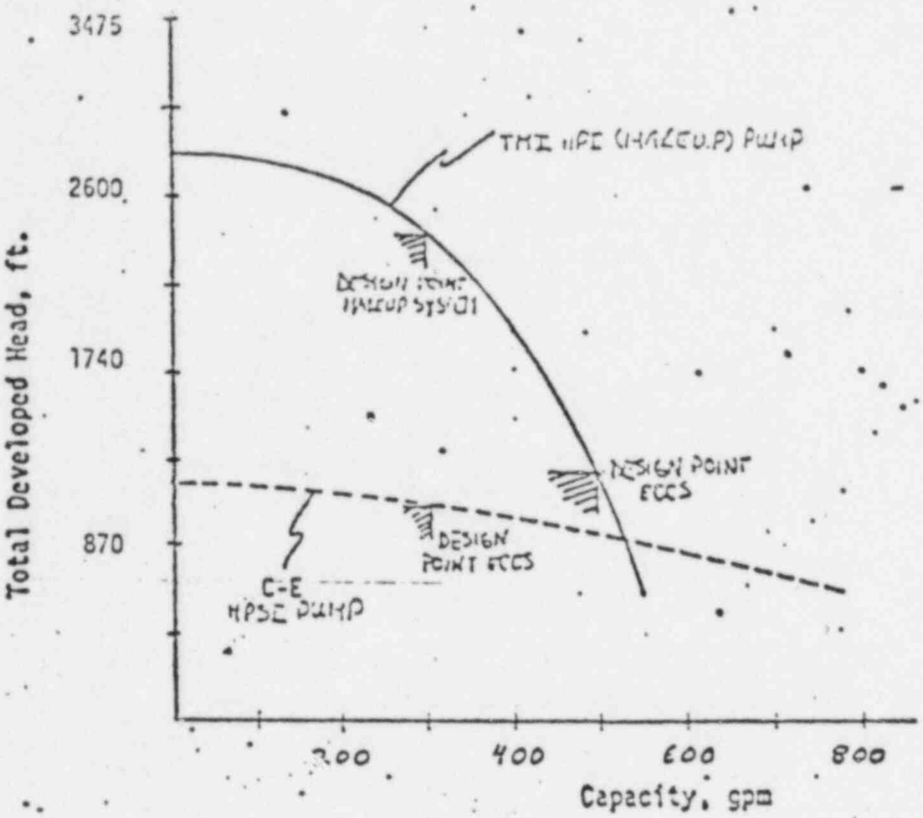
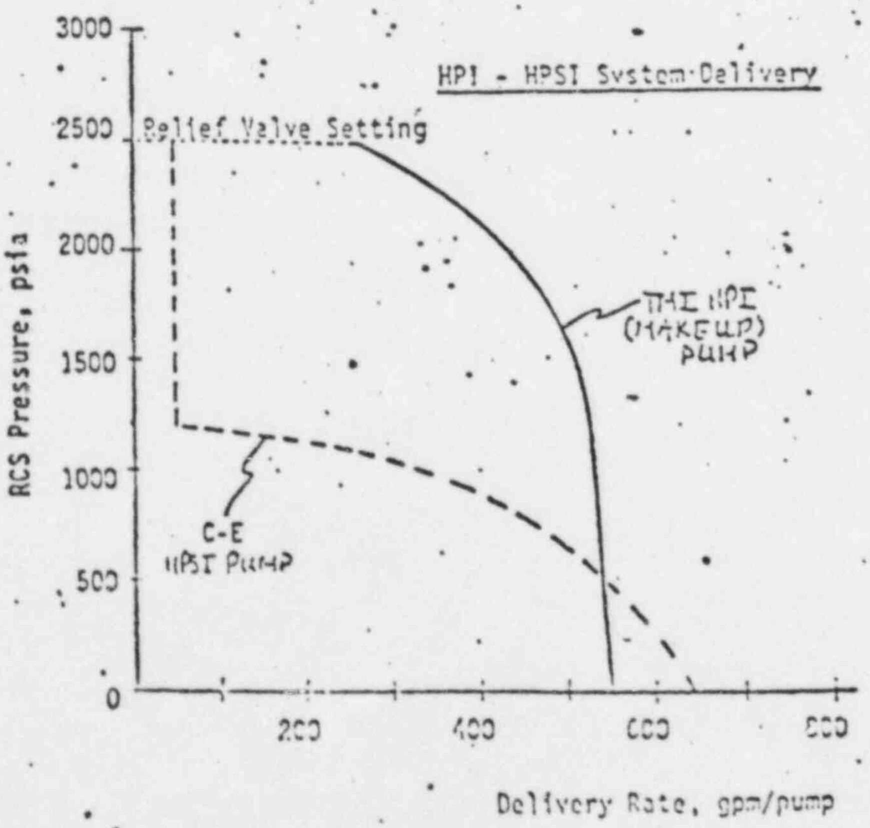
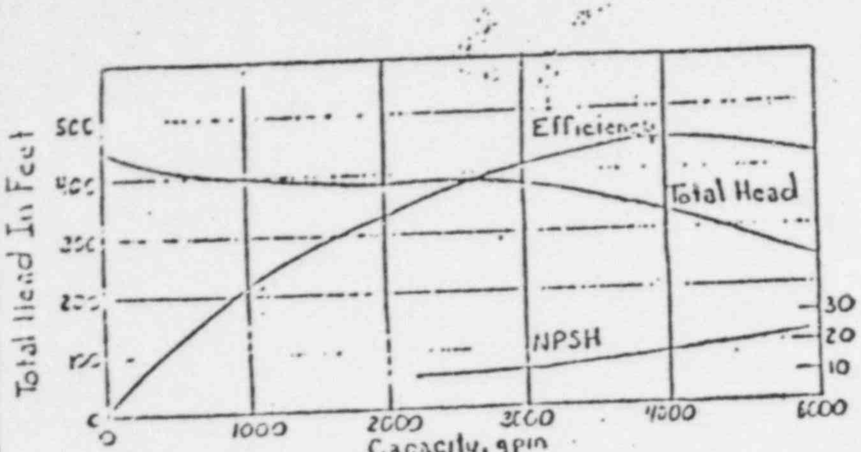


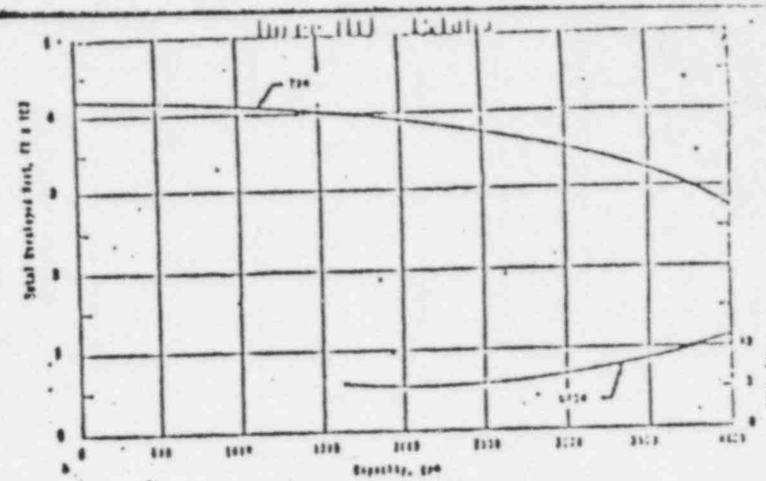
3475
2600
1740
870
Total Developed Head, ft.
THI HPI (MAKEUP) PUMP
DESIGN POINT MAKEUP SYSTEM
DESIGN POINT ECCS
DESIGN POINT ECCS
C-E HPSI PUMP
200
400
600
800
Capacity, gpm



HPI - HPSI System Delivery
3000
2500
2000
1500
1000
500
0
Relief Valve Setting
RCS Pressure, psia
TMI HPI (MAKEUP) PUMP
C-E HPSI PUMP
200
400
600
800
Delivery Rate, gpm/pump



Low Pressure Safety Injection Pump Pressure



Shutdown Cooling System Component Data

Low Pressure Safety Injection Pumps

Quantity	2
Type	Single stage, vertical centrifugal
Quality group class	B
Code	ASME III, Class 2
Design pressure	500 psig
Design temperature	400 °F
Design flow/head	3000 gpm/350 ft.
Maximum flow/head	4000 gpm/275 ft.
Materials	ASTM A351 CR CF8M
Motor horsepower	400 hp

Shutdown Cooling Heat Exchangers

Quantity	2
Type	Shell and tube, Horizontal, U-tube
Service transfer rate	256 Btu/h-°F-ft.^{2}
Heat transfer area/Hx.	5700 ft.^{2}
Tube side:	
Fluid	Reactor coolant
Design pressure	500 psig
Design temperature	400 °F
Material	Austenitic stainless steel
Shell side:	
Fluid	Component cooling water
Design pressure	150 psig
Design temperature	250 °F
Material	Carbon Steel
Design Cases	
Tube side temperature	300 °F
Shell side temperature	95 °F

Shutdown Cooling System Component Data

Decay Heat Removal Pumps

Quantity	2
Type	Single stage, centrifugal
Quality group class	Seismic I
Code	ASME III, Class 2
Design pressure	500 psig
Design temperature	350 °F
Design flow/head	3,000 gpm/350 ft.
Maximum flow/head	4,000 gpm/250 ft.
Materials	Stainless steel
Motor horsepower	350 hp.

Decay Heat Removal Coolers

Quantity	2
Type	Shell and tube
Service transfer rate	
Heat transfer area	6800 ft.^{2}
Tube side:	
Fluid	Reactor coolant
Design pressure	500 psig
Design temperature	250 °F
Material	Stainless steel 304
Shell side:	
Fluid	Decay Heat Removal Component Cooling Water
Design pressure	150 psig
Design temperature	250 °F
Material	Carbon Steel
Design Cases	
Tube side temperature	140 °F
Shell side temperature	95 °F
Tube side flow	3000 gpm
Shell side flow	[illegible]

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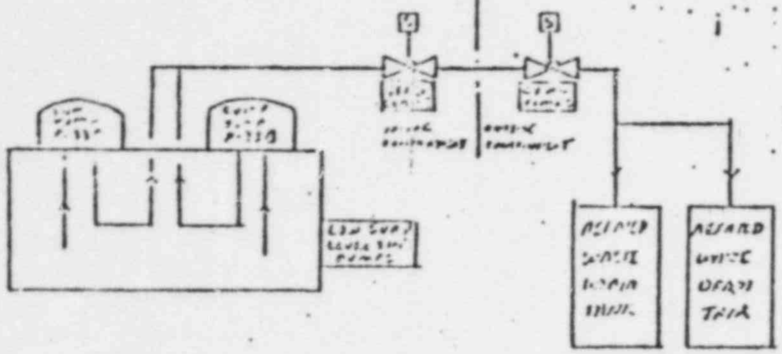
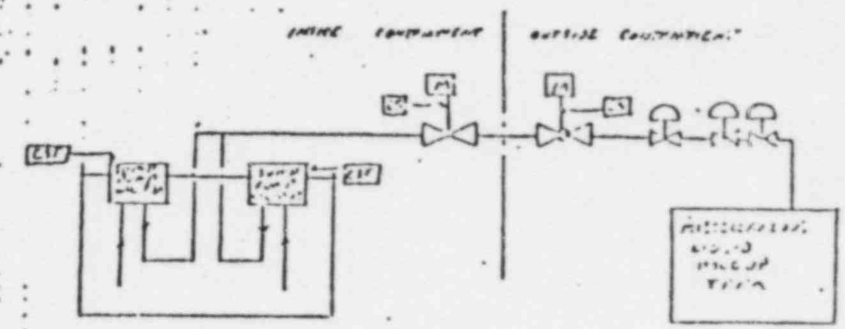
B & W
Three Mile Island





SUMP DEWATERING SYSTEM COMPONENTS

SUMP PUMPS	2. Manually Actuated, 100 gpm each Low Sump Level Interlock Stops Pumps
ISOLATION VALVES	2. ESFAS Actuated (One Closes on SIAS, One on CIAS). Solenoid Operated
AERATED WASTE DRAIN TANKS	2. 5000 gal each, ASME Section VIII, Seismic Category I, 15 psig Design Pressure 150°F Design Temperature ASTM A - 240 TP 304 SS

MISCELLANEOUS WASTE SYSTEM COMPONENTS

SUMP PUMPS	2. Receive ESF Signal, 200 gpm each
ISOLATION VALVES	2. Motor Operated, Receive ES Signal
MISCELLANEOUS LIQUID HOLDUP TANK	One, 2534 ft^3 (~18214 gal.) Seismic Category I ASME III - C Class 3 20 psig Design Pressure 150°F Design Temperature

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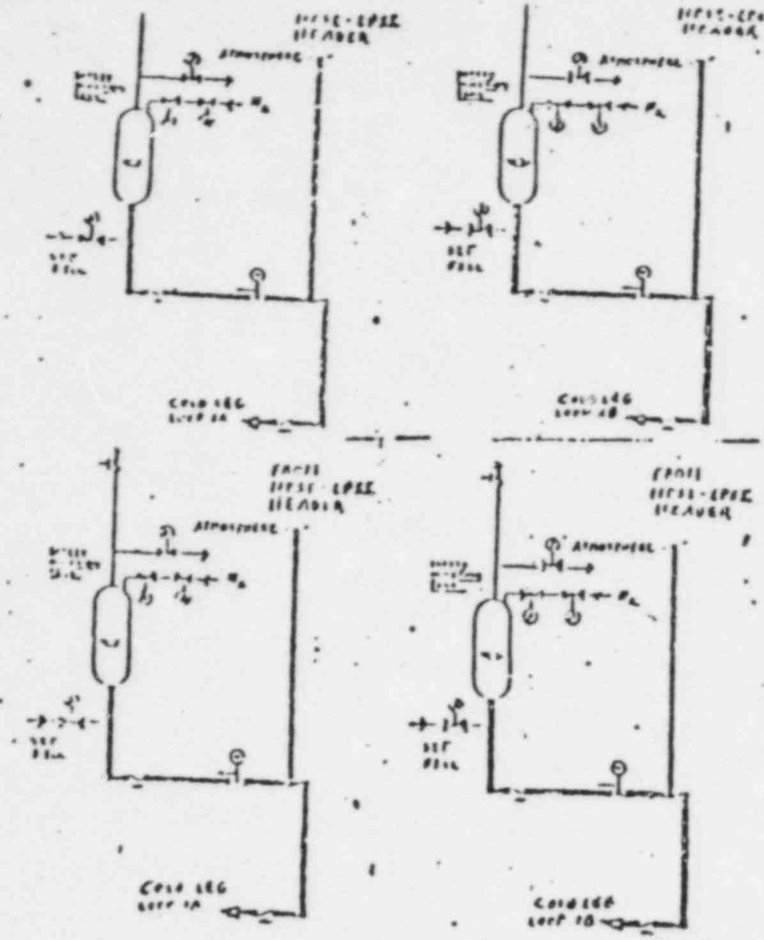
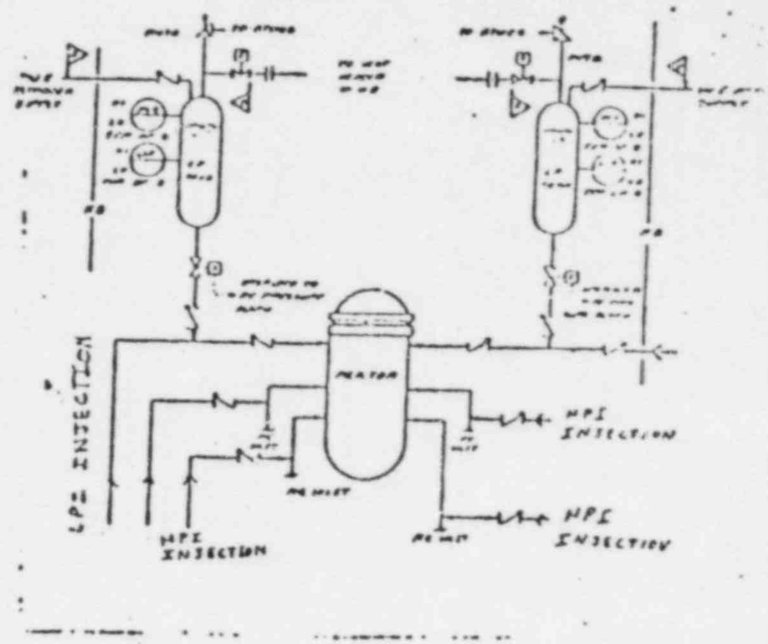
Three Mile Island





Safety Injection Tank Parameters

Number	4
Design pressure	250 psig
Design temperature	200 °F
Operating pressure	215 psig
Operating temperature	120 °F
Total volume	2019 ft.^{3}

Core Flooding Tank Parameters

Number	2
Design pressure	700 psig
Design temperature	300 °F
Operating pressure	600 psig
Operating temperature	110 °F
Total volume	1410 ft.^{3}

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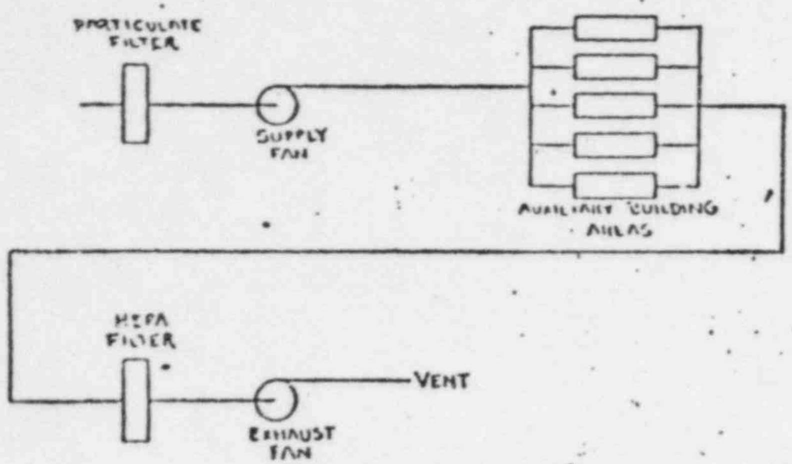
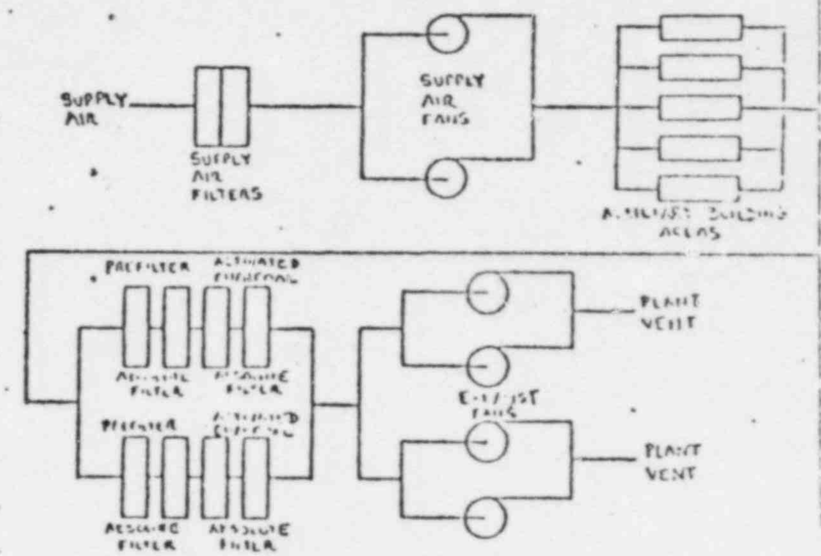


FILTERS

Parameter	Particulate	HEPA
Quantity	1	1
Type	Glass Fiber	High Efficiency Particulate Air
Efficiency (%)	20	99.97
Pressure Drop (in. W.G)	0.1	0.9
Capacity(CFM)	32,000	32,000

B & W
Three Mile Island



FILTERS

Parameter	Supply Air	Prefilter	Absolute Filter 1	Activated Charcoal	Absol[illegible]
Quantity	1	2	30/Assembly	90/Assembly	30/Asse[illegible]
Type	Automatically Advanced Rolled Media	Automatically Advanced Rolled Media	High Efficiency Particulate Air (HEPA)	Activated Charcoal	HEPA
Size	19'0"x11'8"	10'0"x10'4" 2" Media	24"x24" 11 1/2" Media	24"x60" 8" Media	24"x24" 11 1/2"
Differential Pressure (in. W.G.)	0.16	0.16	1.2	1.0	1.2
Capacity(CFM)	32500	32500	32500	32500	325[illegible]

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Part C

GENERIC RESPONSE OF AN OPERATING C-E PLANT TO
LOSS OF FEEDWATER OR A STUCK PRESSURIZER RELIEF/SAFETY VALVE

In the event of a complete loss of feedwater, the immediate response of a C-E NSSS would be a decrease in steam generator level. The loss of subcooled feedwater flow would result in a moderate reduction in steam generator heat removal capability and a slow rise in reactor coolant system (RCS) pressure and temperature. Within 15 to 40 seconds, a reactor trip signal will be produced by low steam generator level, resulting in a reactor trip and a turbine trip (for a plant with a manually initiated auxiliary feedwater system). After trip, the dump and bypass (if in the normal automatic mode) will regulate steam generator and RCS pressure and temperature to hot standby conditions and the power operated relief valve would not open.

The steam generator pressure will stabilize within a couple of minutes; decay heat will be removed through the dump and bypass. This condition could exist for 10 to 20 minutes before the heat transfer capability would start to decrease due to an excessively low steam generator level, thereby resulting in an increase in RCS temperature and pressure. If auxiliary feedwater is initiated within this time period, the power operated relief valve will not open.

For an event which does open the power operated relief or safety valve and which is followed by the unanticipated failure of a pressurizer relief or safety valve to reseat, the RCS will rapidly depressurize. If the relief valve does not reseat, the downstream isolation valve could be manually closed once the RCS pressure drops below the relief valve closing setpoint. If the valve is not, or cannot be isolated, the depressurization will continue. In this situation, the pressurizer surge tank rupture disk will fail, releasing fluid to the containment and the pressurizer bubble may be lost, resulting in liquid relief through the valve. The release of significant RCS fluid to the containment would result in a containment isolation signal.

Before the RCS pressure approaches the saturation pressure of the hot leg, a safety injection signal will occur, automatically starting the high pressure safety injection (HPSI) pumps. If the flow of the HPSI pump is sufficient to increase RCS pressure, there is no incentive for the operator to turn off these pumps to prevent overpressurization since the HPSI shutoff head is below the power operated relief valve setpoint.* If only minimum safeguards are available (i.e., one HPSI pump) due to loss of offsite power and a failure of a diesel or HPSI pump to start, the HPSI flow will be adequate to keep the core cooled. If the reactor coolant pumps are tripped, the core will be adequately cooled as long as the ECCS flow is maintained and the steam generators continue to remove decay heat. The layout of the RCS enhances core cooling because the core is essentially at the lowest elevation of the system, thereby, requiring less than 25% of the RCS inventory to keep the core covered.

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*This statement does not apply to Maine Yankee which has high shutoff head pumps.

Subsequent cooldown of the RCS must be regulated within rates specified by existing procedures, by manual control of auxiliary feedwater flow, and dump valve flow. The cooldown must not be so rapid as to also cause a depressurization below the saturation pressure of the hot leg.

Depressurization must be controlled within the appropriate existing pressure-temperature guidelines by balancing of the ECCS flow with the stuck pressurizer valve flow and other flows out of the RCS, as well as the shrinkage of the RCS as it cools down; again, to assure that the RCS pressure remains above the hot leg saturation pressure.