NORTHEAST UTILITIES
The Connecticut Light And Power Company
Western Massachusetts Electric Company
Holyoke Water Power Company
Northeast Utilities Service Company
Northeast Nuclear Energy Company

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Re: 10CFR50.73(a)(2)(iv)
November 2, 1990
MP-90-1169

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Reference: Facility Operating License No. DPR-21
Docket No. 50-245
Licensee Event Report 90-016-00

Gentlemen:

This letter forwards Licensee Event Report 90-016-00 required to be submitted within thirty (30) days pursuant to 10CFR50.73(a)(2)(iv).

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

Stephen E. Scace
Director, Millstone Station

SES/CW:mo

Attachment: LER 90-016-00

cc: T. T. Martin, Region I Administrator
W. J. Raymond, Senior Resident Inspector, Millstone Unit Nos. 1, 2 and 3
M. Boyle, NRC Project Manager, Millstone Unit No. 1

IE22
1/1

NRC Form 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

Estimated burden per response to comply with this information collection request: 50.0 hrs. Forward comments regarding burden estimate to the Records and Reports Management Branch (p-530), U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to the Paperwork Reduction Project (3150-0104), Office of Management and Budget, Washington, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Millstone Nuclear Power Station Unit 1	0 5 0 0 0 2 4 5	1 OF 1 0

TITLE (4)
Manual Reactor Trip Due to Loss of Cooling

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	
1 0	0 4	9 0	9 0	- 0 1 6	- 0 0	1 1	0 2	9 0		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9)		THIS REPORT IS BEING SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
N		20.402(b)	20.402(c)	X 50.73(a)(2)(iv)	73.71(b)
POWER LEVEL (10)	1 0 0	20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
		20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
		20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
		20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
		20.405(a)(1)(iv)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
Charles Wargo, Ext. 4374	AREA CODE 2 0 3 4 4 7 - 1 7 9 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
X YES (If yes, complete EXPECTED SUBMISSION DATE)	NO		0 5	3 1	9 1

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

On October 4, 1990 at 1849, while reducing power during storm conditions, a manual reactor trip from 45% power (530 degrees F, 1030 psi) was initiated because of degraded conditions in the Service Water and Circulating Water supplies. High winds and heavy seas caused a seaweed buildup on the intake structure traveling screens, which exceeded the Screen Wash system removal capability. Three of five traveling water screens were collapsed by the high differential pressure. Traveling screen failure resulted in Service Water Self Cleaning Strainer fouling. A manual scram was initiated when low service water pressures were noted combined with increasing containment temperature, pressure and decreasing condenser vacuum. The containment temperature/pressure increase was the direct consequence of degraded Reactor Building Closed Cooling Water heat exchanger performance. The Service Water system was restored via the Self Cleaning Strainer Bypass Valve. A cooldown to cold shutdown was accomplished with the remaining intact Traveling Screens, Circulating Water Pumps and Service Water Pumps. A subsequent reactor protection trip signal was received on low reactor water level. Reactor water level was immediately restored. All safety systems functioned as designed.

1. Description of Event

On October 4, 1990 at 1849, while reducing power during degraded weather conditions, a manual reactor trip from 45% (530 degrees F, 1030 psi) power was initiated when Control Operators noted low service water pressures combined with increasing containment temperatures, pressures and decreasing condenser vacuum. High winds and heavy seas had been building through the afternoon of October 4. At 1330 Off Normal Procedure 514 "Natural Occurrences", was entered when winds exceeded 30 mph. The Service Water Strainer was placed in the continuous blowdown mode. All screens were selected for continuous slow rotation. The following chronology describes the sequence of events. Refer to the attached figures for additional information.

1800 Alarms indicated one traveling water screen had greater than a 10 inch differential pressure. Four to five feet of debris had collected in front of the screen. Operators commenced manually cleaning 'E' traveling water screen, however differential pressure was increasing on other screens.

1810 Maintenance personnel were contacted to remove the Screen Wash Sluiceway. The sluiceway provides a pathway for the live return of fish and crustaceans to the sea after being sprayed clear of the traveling water screens.

1825 Plant Equipment Operators attempted to manually clean 'A' traveling water screen. All screens were stopped by securing the Screen Wash Pumps to accomplish this task.

1835 The Control Operator noted all screen differential pressures are off scale high (greater than 60"). A rapid power reduction was initiated. Operators at the Intake Structure were instructed to restart all screens.

1836 Condenser Vacuum alarmed at 27.3 inches of Hg.

1837 Service Water pressure was 9.6 psig.

Note: This data point was taken from Computer logs after the event. It is provided for general interest and to support the analysis provided later in the discussion.

1838 In response to off scale high traveling screen differential pressure the Control Operator tripped 'A' and 'D' Circulating Water Pumps. Associated screens 'A' and 'D' began to operate. ('B','C','E' traveling screens were breached by the accumulated debris and associated differential pressure and failed to rotate.)

1839 The Service Water Strainer high differential pressure alarm was received in the Control Room. (Debris passed through the failed screens reached the Self Cleaning Strainer.)

1839-1847 Control Operators stopped and restarted Circulating Water Pumps in an effort to reduce differential pressure and seaweed load on traveling screens.

1841 Condenser vacuum was 25.5 inches of Hg.
Service Water pressure was 1.6 psig.
Reactor Power was 70%

Note: These data points were taken from Computer logs after the event. It is provided for general interest and to support the analysis provided later in the discussion.

1845 Reactor was power 50% and decreasing.
Circulating pump and traveling screen conditions were not improving,'B','C','D' Traveling screen motor breakers were tripped open on a fault condition. Circulating Water pump seal water pressures and flows were low.

1. Description of Event (Continued)

1847 A high drywell pressure alarm was received. The Control Operator began venting the drywell through the Standby Gas Treatment System.

1848 Control Operators noted Service Water pressure at 7 psig. 'C' Service Water pump was started to improve header pressure.

1849 A manual reactor scram was initiated.
Drywell Pressure - 1.42 psig increasing
(1.1 psig normal)

Drywell Bulk Temperature - 142 degrees F increasing
(130 degrees F normal)

Reactor Building Closed Cooling Water
Heat Exchanger discharge - 110 degrees F increasing
(72 degrees F normal)

Note: These data points were taken from Computer logs after the event. It is provided for general interest and to support the analysis provided later in the discussion.

1850 Group 2 and Group 3 containment isolations actuated at 8 inches Narrow Range Yarway Reactor Water Level signal (normal level response following void collapse).

1854 Control Operator reset the Reactor Scram.

1902 Drywell pressure peaked at 1.48 psig.

Note: This data point was taken from Computer logs after the event. It is provided for general interest and to support the analysis provided later in the discussion.

1907 Reactor Water Low level alarm was received in the Control Room. (20 inches Narrow Range Yarway reactor water level)

1909 PEO restored Service Water pressure by opening the Service Water Strainer Bypass Valve (1-SW-19).

1909 Reactor Protection System trip signal was received on Low Reactor Water Level.
Group 2 and Group 3 containment isolations actuated at 8 inches Narrow Range Yarway Reactor Water level.
Control Operator restored level by starting a Reactor Feed Pump.

1911 Control Operator reset the Reactor Scram.

1912 Drywell Bulk Temperature peaked at 155 degrees F.
Senior Control Operator entered EOP 580 and verified that required actions had been taken.

A cooldown was initiated with the intact 'A' and 'D' Traveling Screens and associated Circulating Water Pumps using the Main Condenser. Service Water remained available during the period through the Strainer Bypass Valve (1-SW-19). The plant achieved Cold Shutdown on October 5, 1990 at 1835 hrs.

II. Cause of Event

The root cause of this event was that the Traveling Screen fouling rate exceeded the Screen Wash System Cleaning capability during severely degraded weather conditions. On shore winds, high seas and an incoming tide combined to create a condition where the debris removal equipment (traveling screens, screen wash system, debris sluiceways) was unable to handle the debris volume.

Contributing Causes: Plant Equipment Operators did not coordinate cleaning of the 'A' traveling screen with the Control Room. All screens were stopped by securing the Screen Wash Pumps to accomplish this task. Therefore, Control Room Operators were not confident of the off scale screen differential pressure indication. Control Operators elected to trip only 'A' and 'D' Circulating Water Pumps.

The Root Cause for the second Reactor Protection System Trip (Low Reactor Water Level) was that establishing a valve lineup which maintained the Feedwater Coolant Injection System operable impacted timely manually controlled injection of feedwater when level was less than 30 inches on the Narrow Range GEMAC instruments.

Contributing Cause: The published low water level scram setpoint is 8 inches Narrow Range Yarway Reactor Water level, however the actual low water level scram setpoint can conservatively set as high as 12 inches indicated Narrow Range Yarway Reactor Water level. The operator actions were influenced by his belief that the low water level RPS trip would occur at 8 inches.

III. Analysis of Event

At 1800 on October 4, 1990 the Traveling Screen fouling rate exceeded the Screen Wash System Cleaning Rate during severely degraded weather conditions. On shore winds, high seas and an incoming tide combined to create a condition where the debris removal equipment (traveling screens, screen wash system, debris sluiceways) were unable to handle the debris volume. Plant Equipment Operators engaged in manual cleaning of the traveling water screens elected to stop all screens to provide access to the interior of the screen housings for manual debris removal. At this time the Control Operator had not been informed of the new screen status. As debris continued to collect the traveling screen differential pressure rapidly increased. Control room personnel observed all five traveling screen differential pressure instruments off scale high. A rapid decrease in reactor power was initiated. It is believed that the traveling water screens were greater than 75% fouled at this time. Control Room personnel questioned the off scale differential pressure and did not elect to trip all Circulating Water Pumps per Operating Procedure 323. 'A' and 'D' Circulating Water Pumps were stopped at 1838 hrs, relieving the strain on the associated screens. 'B' and 'C' Circulating Water Pumps continued to pump, lowering level in the bays and increasing differential pressure on 'B' and 'C' screen as well as the interconnected 'E' screen. The level decrease cavitated the operating Service Water Pumps. 'B','C' and 'E' screen baskets and support structure yielded to the strain. 'C' Circulating Water Pump was stopped and started and the 'B','C' and 'E' bays of the Intake Structure reflooded, ending the cavitation but, introducing debris to the Service Water pump suctions. Seaweed (Genus *Codium*) then fouled the Service Water Self Cleaning Strainer pores and was not expelled by the strainer blowdown.

III. Analysis of Event (Continued)

(*Codium* is not native to North America, introduced around 1955 it is expanding its domain in this area. It experiences high productivity during the late summer months. *Codium* is buoyant, free floating in the water column, when not attached to its holdfast. The cross section of this marine plant closely matched the pore diameter of the Service Water Strainer, plugging a high percentage of the strainer basket. Divers participating in the repair of damaged equipment also reported large masses of seaweed on the sea bottom local to the facility.)

The low Service Water Pressure reduced Reactor Building Closed Cooling Water Heat Exchanger capacity. These heat exchangers are located at the 42'6" elevation of the Reactor Building. Drywell Temperature and Pressure began to increase and a high Drywell pressure alarm was received in the Control Room. The drywell was vented through the Standby Gas Treatment System. During the evaluation of the containment pressure increase, Service Water pressure was observed to be 7 psig in the Control Room. An additional service water pump was started to increase header pressure. A Manual Scram was initiated at 1849 hrs. Service Water pressure was restored by throttling open the Service Water Strainer Bypass Valve.

If all circulating water pumps had been tripped on increasing differential pressure, traveling screens 'B','C' and 'E' may not have been breached. Service Water Strainer fouling and the associated containment temperature/pressure increases would not have occurred, however a Turbine trip and associated Reactor Scram would still have occurred. The decision to trip Circulating Water Pumps was delayed because the Control Room personnel were unaware that all traveling water screens had been stopped for cleaning. This complicated the decision while the legitimacy of the simultaneous off scale differential pressure indications were considered. Delays in tripping 'B' and 'C' Circulating Water pumps eventually threatened the operation of the Service Water system. However, Operations and Maintenance personnel were successful in maintaining the Service Water system functional by cleaning heat exchangers and throttling the Service Water Strainer Bypass Valve.

This event had the potential to result in degradation of the Service Water and Emergency Service Water to a condition where available flowrates from those systems could have been ineffective. The Service Water System provides cooling to the Turbine and Reactor Building Closed Cooling Water Heat Exchangers and to the Diesel Generator Heat Exchangers. The Emergency Service Water System provides a long term source of cooling water to remove heat from the Suppression Pool during both LOCA and non-LOCA conditions (refer to the attached system diagrams). Millstone One Off Normal Procedures 524D "Loss of Service Water", and 525G "Degraded Fire in the Intake Structure", provide operator guidance for more severe events of this type. In this case, operation of the Service Water Strainer Bypass Valve maintained service water header pressure and flow to heat exchangers during the event such that adequate cooling remained available.

A loss of Service Water & Emergency Service Water is an event considered to be outside the design basis for Millstone One. Postulating the concurrent occurrence of a LOCA in addition to the loss of Service Water/Emergency Service Water event is not required from a design basis perspective.

Both the Gas Turbine Emergency Generator and Isolation Condenser remained unaffected by the event, and were available at all times to provide decay heat removal capability and emergency power. Therefore, the safety significance of this event is minimal.

A second Reactor Protection System (RPS) trip on low reactor water level was received at 1909 hrs on October 4, 1990. This Emergency Safety Feature actuation was not separately reported on October 4, 1990 as it was considered part of the overall event. It is discussed here in relation to this event as this RPS actuation was an integral part of the recovery from the transient.

III. Analysis of Event (Continued)

Following a scram the level control system responds to reduced level from void collapse, and the Feedwater Regulation Valves open. This coincides with decreased steam production post-scram. The ensuing overshoot of the level control system must be managed by the Control Operator. Control Operators are trained to sequentially secure Reactor Feed Pumps to terminate the increasing level trend. From this point on, pressure control and decay heat removal via the Main Turbine Bypass Valves and Main Condenser lowered reactor water level while make up was limited to 50 gpm from the Control Rod Drive cooling water flow. The Control Operator upon receiving the low level alarm proceeded to line up the Feedwater System, closing one Feedwater Regulation Isolation Valve and selecting the other Feedwater Regulation Valve to manually control the rate of injection per the Operating Procedure 316 "Feedwater System". While waiting for the Feedwater Regulation Isolation Valve to close, the RPS trip signal was received. The Technical Specification Low Reactor Water Level RPS trip is 8 inches. This setpoint can be conservatively set as high as 12 inches indicated. As left setpoints for this RPS trip would have resulted in a scram at 10 inches reactor water level. Instrument drift between calibrations is on the order of one-half inch and no greater than 1 inch during typical calibration intervals. A Reactor Feed pump was started and level was immediately restored and the RPS trip was reset. The safety significance of this event is minimal in that the Control Operator was standing by in control of reactor water level when the RPS trip was received and all Emergency Core Cooling systems were available to respond to a more severe level transient.

This event is reported in accordance with 10CFR50.73(a)(2)(iv), any event or condition that results in manual or automatic actuation of an Engineered Safety Feature. Immediate notifications were performed in accordance with 10CFR50.72(b)(2)(ii).

IV. Corrective Action

The Circulating Water Pump trip logic was modified to reinstate a 30 inch screen differential pressure trip. Operating Procedure 323 "Circulating Water System", was revised to instruct Control Operators to verify Circulating Water Pump trips at 30 inch screen differential pressure.

Off Normal Procedure 514A, "Natural Occurrences" will be revised to terminate Screen Wash debris/fish return capability and begin intercepting debris cleaned from the Traveling Screens at sustained winds of greater than or equal to 30 mph. This is to minimize the debris load in front of the Intake Structure during degraded weather conditions.

Additional guidance has been communicated to the Operations staff should winds approach 30 mph., including:

a) Request an updated weather report.

b) Place service water strainer in manual blowdown.

c) Increase watch frequency at the intake structure.

d) Remove sluiceway and put trash baskets in place.

e) Make arrangements for Maintenance support of trash basket and heat exchanger cleaning.

f) Make arrangements for additional Plant Equipment Operator (PEO) support.

g) As directed in OP 514A - Natural Occurrences, if loss of Circulating Water Pumps is imminent, quickly lower power to minimum speed of Recirculation Pumps.

A design review has been commissioned to evaluate Traveling Screen performance in severe weather with respect to debris removal methods and equipment.

IV. Corrective Action (Continued)

The Operations Manager will review plant operating philosophy with the Operations Department personnel on the following subjects. This activity is in progress and will be completed by 12/15/90.

a) The operators use and belief of instrumentation with appropriate confirmation.

b) The operators use of effective communication during normal and abnormal conditions with emphasis on informing the Control Room when changing equipment status.

c) The importance of conservative decision making. This review will focus on strengthening existing philosophy and assuring the Operations staff that this philosophy is supported by Station Management.

The Operator Training Department has been requested to evaluate this event and provide an assessment from a training perspective.

A review of past Millstone One design changes will be conducted to ensure that any protective trip functions previously removed have no significant impact on plant safety or reliability.

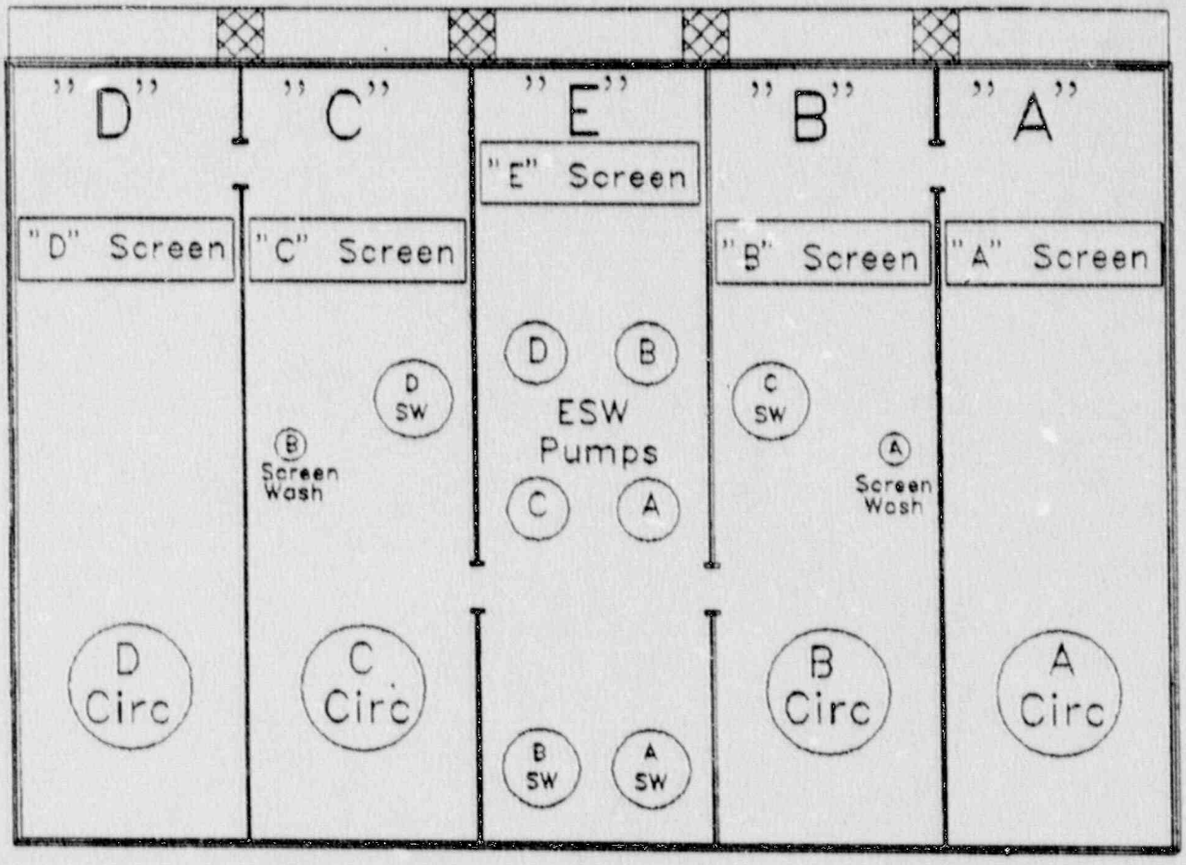
A review of Operating Procedure 316 "Feedwater System" will be conducted. This review will evaluate reactor water level response post-scram and provide the operator actions required to control level more efficiently.

V. Additional Information

The following attachment and information is being provided to identify the system and components affected by the event.

EIIS Codes

Systems	Components
Circulating Water System - KE	Pumps - P
	Traveling Water Screens - SCN
	Condenser - COND
Emergency/Service Water Systems - BI	Service Water Strainer - STR



Intake Structure

Figure 1

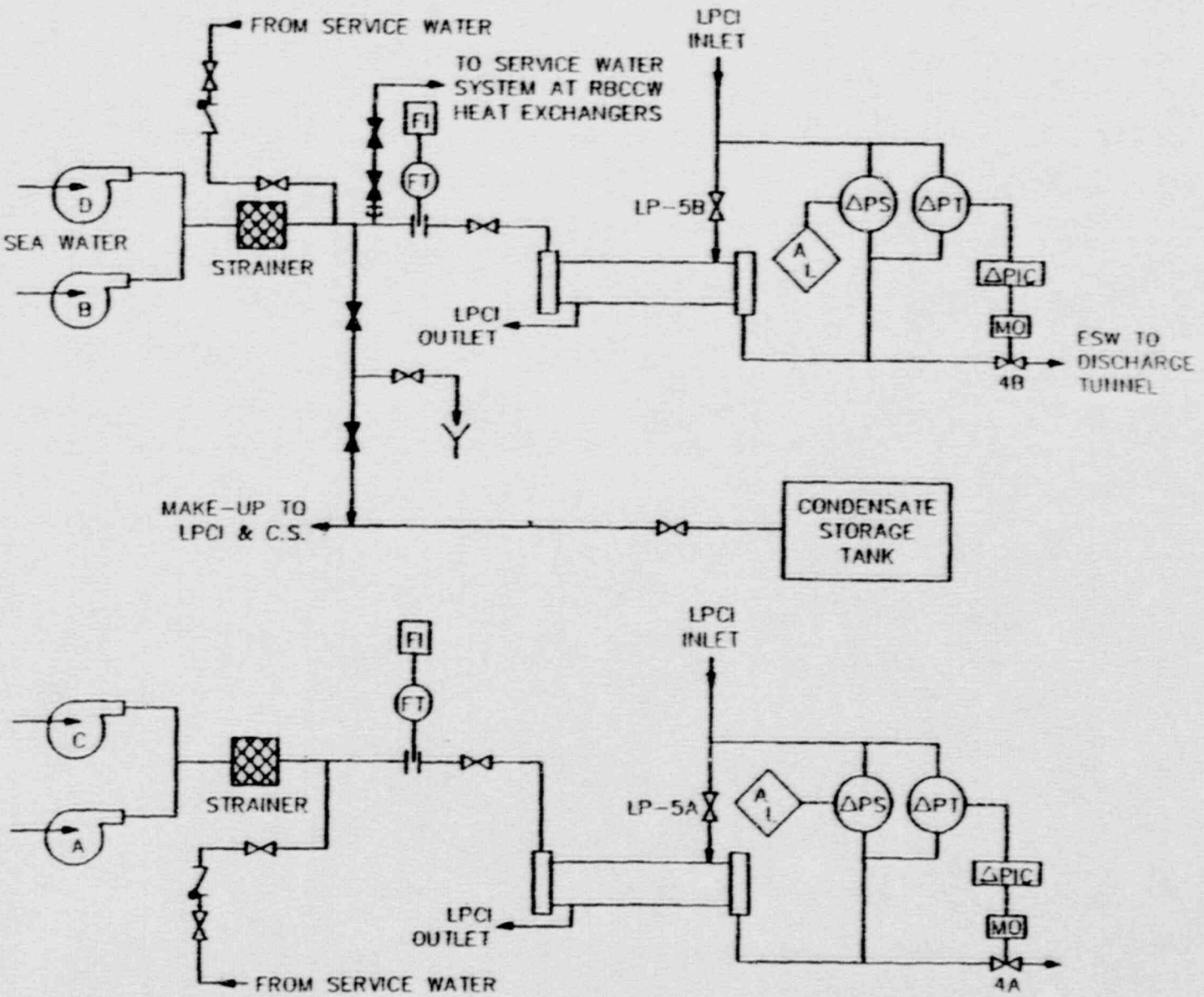


FIGURE 2

EMERGENCY SERVICE WATER SYSTEM

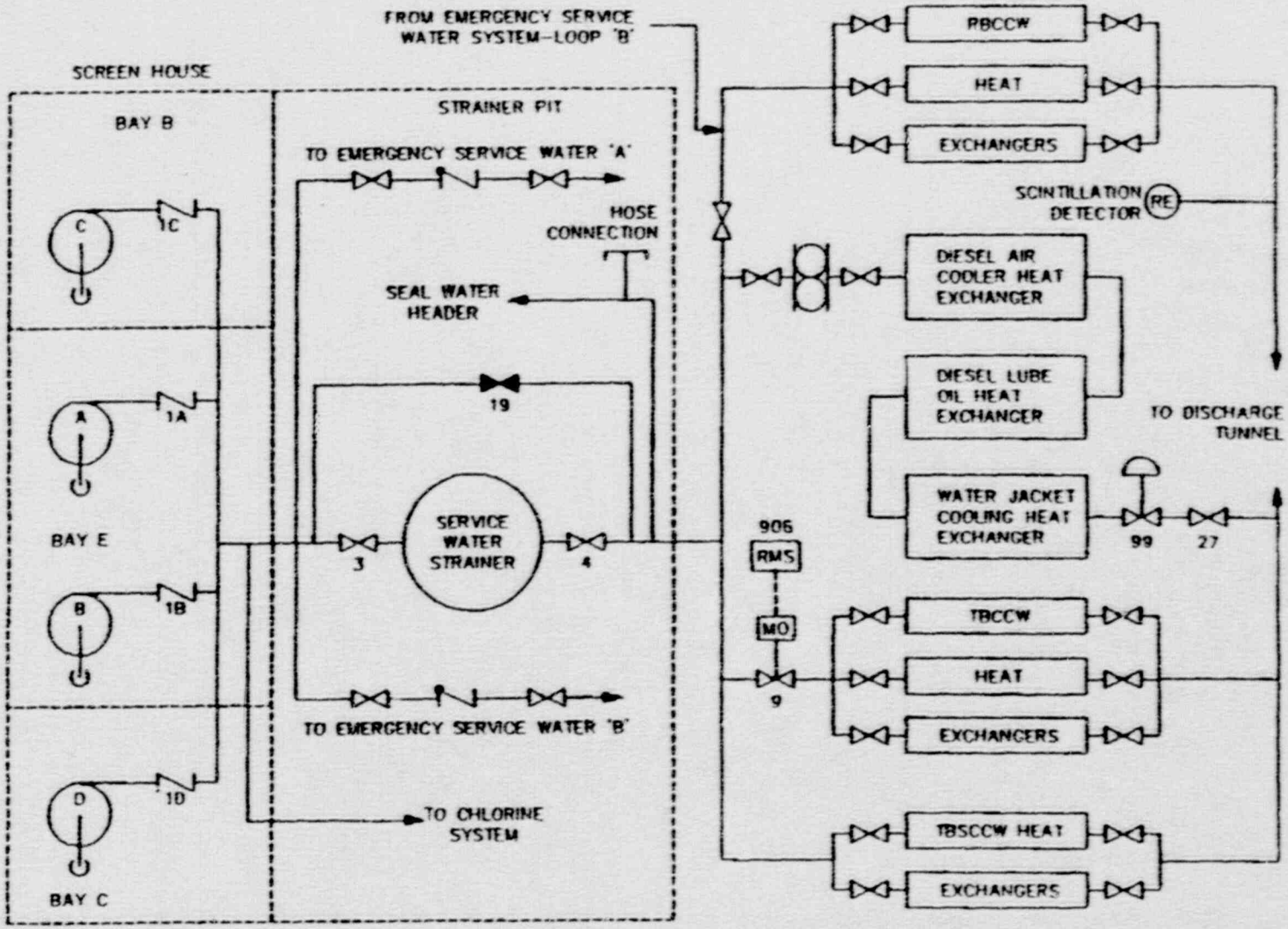


FIGURE 3

SERVICE WATER SYSTEM

NRC Form 366
(6-89)

NRC Form 366A
(6-89)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

APPROVED OMB NO. 3150-0104
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FACILITY NAME (1)

Millstone Nuclear Power Station
Unit 1

DOCKET NUMBER (2)

0	5	0	0	0	2	4	5

LER NUMBER (6)

YEAR		SEQUENTIAL NUMBER		REVISION NUMBER	
9	0	–	0	1	6	–	0	1	0

PAGE (3)

OF	1	0

TEXT (If more space is required, use additional NRC Form 366A's) (17)