NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104

EXPIRES 04/30/99

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Millstone Nuclear Power Station Unit 2	05000336	1 OF 3

TITLE (4)

Auxiliary Feedwater DC Power Supply Failure Not Considered in Safety Analysis

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 NAME	DOCKET NUMBER
09	10	98	98	-- 022 --	00	10	13	98	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
			20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
POWER LEVEL (10)	000		20.2203(a)(1)		20.2203(a)(3)(i)	X	50.73(a)(2)(ii)		50.73(a)(2)(x)
			20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
			20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
			20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or NRC Form 366A
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
R. G. Joshi, MP2 Regulatory Compliance Manager	(860) 440-2080

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		01	29	99

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On September 10, 1998, a condition report was issued based on the Independent Corrective Action Verification Program contractor finding which indicated that the Auxiliary Feedwater (AFW) system may not meet the loss of normal feedwater safety analysis for all single failures. Specifically, the loss of the Facility 2 125 VDC Bus can result in less AFW flow to the Steam Generators (SG) than is credited in the safety analysis for a Loss of Normal Feedwater event.

The cause of the condition is under investigation.

Prior to entering Mode 4 from the current outage, causal factors to determine the extent of the condition will be identified and appropriate corrective actions taken.

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Millstone Nuclear Power Station Unit 2	05000336	98	-- 022 --	00	2 OF 3

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

I. Description of Event

On September 10, 1998, a condition report was issued based on the Independent Corrective Action Verification Program contractor finding which indicated that the Auxiliary Feedwater (AFW) [BA] system may not meet the loss of normal feedwater safety analysis for all single failures. Specifically, the loss of the Facility 2 125 VDC Bus can result in less AFW flow to the Steam Generators (SG) [AB] than is credited in the safety analysis for a Loss of Normal Feedwater event. At the time of discovery the unit was defueled.

The plant safety analysis is documented in FSAR Chapter 14. FSAR section 14.0 states in part that,

"All of the reactor operating conditions allowed by the plant technical specifications are examined to ensure that bounding subevents are identified for each SRP event category. This ensures that the safety analysis will support the complete range of allowable operating conditions."

FSAR Section 14.0.11, "Plant Licensing Basis and Single Failure Criteria" states in part that,

"The event scenarios considered in the safety analysis depend on single failure criteria."

"The safety analvsis is structured to demonstrate that the plant systems design satisfies these single failure criteria."

"The ESFs required to function in an event are assumed to suffer a worst single failure of an active component."

The safety analysis for a loss of normal feedwater does not satisfy the design basis single failure criterion. Loss of the Facility 2 125 VDC Bus (201B) [EJ] results in a loss of control power to the TDAFWP and the facility 2 MDAFWP. The Facility 2 125 VDC Bus provides control power required to automatically start the Facility 2 MDAFWP and to remote manually start the TDAFWP. Therefore, for a postulated Facility 2 125 VDC Bus failure, control power for starting the TDAFWP and the Facility 2 MDAFWP would be lost, leaving only the Facility 1 MDAFWP available to perform the system function of decay heat removal within the assumed 10 minute timeframe.

Therefore, the loss of Facility 2 125 VDC Bus can result in less AFW flow to the SG than is credited in the safety analysis for the Loss of Normal Feedwater event.

This condition is being reported in accordance with 10 CFR 50.73(a)(2)(ii)(B), a condition that was outside of the design basis of the plant.

II. Cause of Event

The cause of the condition is under investigation.

III. Analysis of Event

The AFW system provides feedwater for the removal of sensible and decay heat to cool the RCS to 300°F or maintain hot standby conditions whenever normal feedwater is not available. This safety function mitigates the following FSAR Chapter 14 events: Steam Line Break, Loss of Normal Feedwater Flow, Steam Generator Tube Rupture, Small Break Loss of Coolant, Station Blackout and Appendix R Fire.

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Millstone Nuclear Power Station Unit 2	05000336	98	-- 022 --	00	3 OF 3

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

The AFW system has two MDAFWPs and a TDAFWP which is approximately twice the capacity of the MDAFWPs. The MDAFWPs receive an automatic start signal on a low SG level from an Auxiliary Feedwater Automatic Initiation Signal (AFAIS). AFAIS is provided to ensure delivery of sufficient feedwater to the SGs in the event of the loss of normal feedwater. Upon actuation, two MDAFWPs start and the flow control valves to both SGs open. The TDAFWP can be remote manually started either from the control room or the hot shutdown panel. The TDAFWP can also be locally started at the pump without DC control power. TDAFWP operation outside the control room is not credited in the current analysis due to the time constraints for AFW flow initiation.

The postulated loss of the DC Bus 201B is required to satisfy General Design Criterion requirements for the single failure of an engineered safety feature. Per IEEE-279, 1971, both active and passive failures must be separately considered in the long and short term for electrical equipment which performs a safety function.

Based on the following, this condition is not considered to be safety significant:

The analysis of a loss of feedwater, without assumed degraded MDAFWP pump performance, shows that sufficient cooling is provided with one MDAFWP to preclude opening of the RCS PORV. This analysis addresses system reliability / availability with a single MDAFWP feeding the SG. The results show that, although RCS pressure initially rises, sufficient cooling is provided with one MDAFWP with nominal pump performance to preclude opening of the RCS PORV. This "best estimate" analysis credits the SG atmospheric dump valves for removal of decay heat.

The loss of two AFW pumps requires a specific passive failure of DC electrical equipment. The two DC Buses are considered to be highly reliable. The redundant and independent buses are each connected to a battery and battery charger supplied by vital power. Although loss of a DC bus occurred twice at Millstone Unit 2, it was not due to a failure of DC equipment, and was not coincident with any ongoing events.

Existing abnormal operating procedures identify that a loss of DC Bus 201B affects the TDAFWP. Existing emergency operating procedures provide direction for local manual operation of the TDAFWP with DC control power unavailable.

IV. Corrective Action

Prior to entering Mode 4 from the current outage, causal factors to determine the extent of the condition will be identified and appropriate corrective actions taken.

V. Additional Information

Similar Events

The following previous similar events involving analysis of Engineered Safety Features were identified.

LER 98-002	Emergency Core Cooling System Single Failure Vulnerability
LER 98-004	Auxiliary Feedwater Pump Performance Degraded
LER 97-025	Single Failure Vulnerability of the AFW System via the Condenser Hotwell Make-up Valve
LER 97-023	Minimum HPSI Flow Used in FSAR Accident Analysis May be Non-conservative

Energy Industry Identification System (EIIS) codes are identified in the text as [XX].