NRC FORM 366 (4-95) US 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 4

TITLE (4)

Potential Inability to Close Auxiliary Feedwater Regulator Valves after a High Energy Line Break

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
08	26	98	98	-- 019 --	00	09	25	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 in 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)
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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
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

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

On August 26, 1998, a Nuclear Regulatory Commission (NRC) review of the auxiliary feedwater system (AFW) questioned the historical reportability of crediting non-safety grade components to mitigate the consequences of a design basis accident.

On September 16, 1998, a NRC review also identified that following a high energy line break (HELB) in the turbine building (TB), a higher-than-previously analyzed ambient temperature may also have challenged the ability of the AFW regulating valve backup air equipment to remotely close and isolate flow. Subsequently, the licensee identified that other components of the AFW regulating valve also may be challenged. Current estimates predict a maximum TB temperature of 326 degrees Fahrenheit compared to the earlier prediction of 220 degrees Fahrenheit.

The cause of the August 26, 1998 and September 16, 1998 conditions previously were addressed by LER 50-336/97-031, as inadequate review by the original facility analysis for postulated HELBs. The resultant corrective action for the earlier reported condition currently is in progress.

No immediate corrective action was required as the AFW system is not required to be operable in the current mode. As a pending corrective action, the auxiliary feedwater system will be upgraded, as required, to ensure compatibility with analyzed HELB conditions.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Millstone Nuclear Power Station Unit 2	05000336	98	-- 019 --	00	2 OF 4

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

I. Description of Event

On August 26, 1998, a Nuclear Regulatory Commission (NRC) review of the auxiliary feedwater system (AFW) [BA] questioned the historical reportability of crediting non-safety grade components to mitigate the consequences of a design basis accident.

In 1981, the AFW regulating valves [V] were changed from motor operated valves (MOV) to air operated valves (AOV) to provide electrical independence from a failure which could preclude the valves from opening and supplying AFW to the steam generators (S/G) [SG]. Both normally closed valves were designed to open by an Auto AFW Initiation (AAFWI) signal or fail open upon loss of instrument air (IA) [LD]. In 1992, nonsafety grade backup air was installed to provide the ability to close the valves in the event of a beyond design basis main feedwater system [SJ] line break in the turbine building (TB), i.e., a high energy line break (HELB), coincident with a feedwater check valve failure. This change was an effort to improve the unit's core melt frequency, however, it was not recognized at that time that more limiting HELBs could affect the TB. The current main steam line break (MSLB) analysis assumes operator action to isolate the faulted S/G ten (10) minutes after the event. However, closure of the AFW regulating valve and isolation of the S/G may not be assured.

On September 16, 1998, a NRC review also identified that following a HELB in the TB, a higher-than-previously analyzed ambient temperature may also have challenged the ability of the AFW regulating valve backup air equipment to remotely close and isolate flow. Subsequently, the licensee identified that other components of the AFW regulating valve also may be challenged. Current estimates predict a maximum TB temperature of three-hundred-twenty-six (326) degrees Fahrenheit compared to the earlier prediction of two-hundred-twenty (220) degrees Fahrenheit. At the time these conditions were identified, the unit was defueled.

Based on the recently analyzed HELBs, both the ability of the AFW regulating valve to be remotely closed, as well as the operator's ability to enter the TB and manually close the valve to the faulted S/G within the required ten minute time limit, were challenged. Therefore, from 1981 through the present, when the plant operated in Modes 3 or higher, the capability to manually close the AFW regulating valves to isolate flow out the faulted S/G as analyzed, could not be assured and a potential for operating outside of the plant's design basis existed. As a result, this condition is being reported pursuant to 10 CFR 50.73(a)(2)(ii)(B), as a condition outside the plant's design basis.

Since the AFW system is not required in the present mode, there were no immediate operator actions required.

II. Cause of Event

The cause of the August 26, 1998 and September 16, 1998 conditions previously were addressed by LER 50-336/97-031 as inadequate review by the original facility analysis for postulated HELBs. The resultant corrective action for the earlier reported condition currently is in progress.

III. Analysis of Event

The AFW system is designed to provide feedwater to the steam generators for the removal of sensible and decay heat from the Reactor Coolant System [AB] in the event of loss of normal feedwater flow. The AFW system is used to mitigate transients and accidents such as: Loss of Normal Feedwater, Small Break Loss of Coolant Accident, S/G Tube Rupture and Steam Line Break. The AFW system has two redundant electric motor driven pumps [P] (MDAFWP) and one turbine [TRB] driven pump (TDAFWP). An automatic initiation signal starts both MDAFWPs. The TDAFWP is manually initiated by a control room operator. A normally opened AFW cross-tie valve can be closed to provide isolation between the MDAFWPs and the TDAFWP. Auxiliary feedwater flow to the S/Gs is initiated and controlled by two (2) regulating valves which are normally closed during power operation and open upon receipt of an AAFWI signal.

For a MSLB event, AFW is assumed to be isola' ` to the faulted generator ten minutes after initiation of the event. A MSLB in the TB results in elevated temperatures that exceed the current rating of the AFW regulating valve and its backup air supply, and the AFW cross-tie valve which potentially affects their ability to operate. After an AAFWI signal, if the AFW regulating valve supplying the faulted SG and the AFW cross-tie valve cannot be closed, isolation of the faulted SG may be delayed. Delay of the MSLB assumed operator action affects the long term temperature and pressure profiles used for the environmental qualification of equipment and increases the rate of consumption of the feedwater supply.

Although non-safety, the backup air supply to the AFW valves has been designed to operate in a harsh environment. In this instance, it is reasonable to assume that the higher area temperatures would not inhibit the equipment (AFW valves, backup air supply, cross-tie valve, etc.) from performing their intended safety function, but rather to potentially adversely affect the equipment's operating life, i.e., the time the equipment is anticipated to be operating in a harsh environment may be shortened due to the higher area temperatures. Although, an increased consumption of feedwater has not been analyzed for this case, it is expected that the AFW components would be able to perform their accident function. Additionally, it would also be expected that an adequate amount of time would be available for an operator to remotely manipulate alternate components (such as the cross-tie valve) in order to isolate flow to the S/G.

Although these conditions challenged the post-accident results from the MSLB analysis and as such, could have potentially placed the unit outside of its design basis, the impact on safety was minimal and are considered to be of low safety significance.

IV. Corrective Action

No immediate corrective action was required as the AFW system is not required to be operable in the current mode. The following corrective action will be performed:

1. Prior to entering Mode 3 from the current outage, the auxiliary feedwater system will be upgraded, as required, to ensure compatibility with analyzed HELB conditions.

V. Additional Information

Similar Events

Previous similar conditions involving the issues described in this report are as follows:

LER 91-010: Supplement 4 to this LER (reported on March 26, 1993), identified a condition involving potential unconservative assumptions made in the 1979 MSLB with respect to power level, break size, and single active failure. Subsequently discussed in the analysis of event, was a scenario involving failure to isolate AFW flow. The discussions concluded that an inability to isolate AFW within the 10 minute timeframe would not adversely impact safety, however, there was no discussion involving potential safety implications due to a failure to isolate AFW flow after that timeframe.

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FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Millstone Nuclear Power Station Unit 2	05000336	98	-- 019 --	00	4 OF 4

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

LER 97-031: This LER reported that certain safety-related components and structures required for safe shutdown of the plant are not properly protected from the dynamic and environmental effects of pipe ruptures in high energy lines. The cause of this condition was a failure to adequately evaluate potential effects from a HELB on safety-related equipment located outside containment. A corrective action was to evaluate present pipe rupture design configurations against postulated high energy line breaks and to address deficiencies as they are identified.

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