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

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98
ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

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

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TITLE (4)

Spent Fuel Pool Cooling System Potentially Inoperable Following an SSE Due to Failure of Non Seismic Connecting Piping.

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
10	02	96	96	037	00	11	01	96	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	5	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)	00	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 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)
J.M. Peschel, MP3 Nuclear Licensing Manager	(860)437-5840

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	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)

At 12:40 on October 2, 1996 with the plant in mode 5 of an extended outage, an Engineering review determined that failure of the non-seismic purification lines connected to the Spent Fuel Pool (SFP) could result in a loss of SFP cooling. The purification lines are connected to the SFP at the same elevation as the SFP cooling system suction lines. Because of this, drain down of the SFP to the level of the purification line penetrations would result in the SFP cooling line being partially out of the water. SFP cooling would be unavailable until repairs to, or isolation of, the purification lines could be accomplished and make up provided to restore SFP level.

The SFP cooling system was declared inoperable. An immediate notification was made at 12:47 hours on October 2, 1996 pursuant to 10CFR50.72(b)(1)(ii)(B) as a condition that results in the nuclear power plant being in a condition that is outside the design basis of the plant.

The SFP cooling system and/or purification system will be modified to preclude loss of cooling due to a failure of the purification system. Administrative controls are being implemented to minimize exposure to potential failures until modifications can be completed. The related calculations will be generated, updated and/or upgraded to reflect current information. The FSAR will be revised to clarify that the flow path for make up from the Refueling Water Storage Tank (RWST) is through the non seismic purification system.

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I. Description of Event

At 12:40 on October 2, 1996 with the plant in mode 5 of an extended outage, an Engineering review determined that failure of the non-seismic purification lines connected to the Spent Fuel Pool (SFP) could result in a loss of SFP cooling. The purification lines are connected to the SFP at the same elevation as the SFP cooling system suction lines. Because of this, drain down of the SFP to the level of the purification line penetrations would result in the SFP cooling line being partially out of the water. There would be no damage to the spent fuel pool cooling system itself provided the SFP cooling pumps were secured on low level as required by Abnormal Operating Procedures (AOP) for "Loss of Spent Fuel Pool Cooling". However, pool cooling would be unavailable until repairs to, or isolation of, the purification lines could be accomplished and make up provided to restore SFP level.

The spent fuel pool cooling system was declared inoperable and outside it's design basis. An immediate notification was made at 12:47 hours on October 2, 1996 pursuant to 10CFR50.72(b)(1)(ii)(B) as a condition that results in the nuclear power plant being in a condition that is outside the design basis of the plant.

II. Cause of Event

The root cause for the plant being outside its design basis, relative to the SFP cooling system, is the improper initial design of this system. The penetrations of concern are located at the elevations shown on the original issue (1978) of the piping drawings for the SFP cooling system. This condition has existed since the original design of the plant. This was 8 years prior to commercial operation of the unit. No justification of the basis for the location has been located.

The locations of the piping and siphon breakers were situated to ensure that SFP drain down levels resulting from a pipe break do not exceed the requirements of Reg. Guide 1.13, Rev 1 which requires that a failure of the purification system during an SSE must not result in uncovery of the fuel stored within the SFP. In addition, the design met the requirements of Standard Review Plan (SRP) 9.1.3 section III.1 e. which states that the pool connecting systems should be designed such that a failure of the inlet piping, or outlet piping, or drains would not cause SFP level to reach a level lower then 10 feet above the top of the active fuel. However, the original design did not take into account SRP 9.1.3 section III.5 which states that a failure of the non safety pool cleanup system at interconnections or interfaces "not preclude adequate functional performance of the cooling system".

The root cause evaluation also revealed the inability of personnel associated with several previous reviews of this system (from original design until September of 1996) to identify that the system design was outside of the design basis. Additionally, management did not recognize the significance of emerging issues associated with SFP design.

III. Analysis of Event

The Final Safety Analysis Report (FSAR), discusses potential drain down of the purification system. Specifically, it states that the purification system design prevents "...drain down of the fuel pool water to uncover spent fuel". FSAR question 410.14 asked, "Is there any portion of the SFP cooling and cleanup system designed to non-seismic requirements? If so, verify that failure of the non-seismic Category I portion in an earthquake will not affect the operation of the cooling trains." The response to this question stated that "Failure of the purification portion in an earthquake does not affect the operation of the cooling trains". This response was incorrect and did not consider the consequences of the drain down on the SFP cooling suction line. Failure of the purification portion does not adversely

affect the operation of the equipment by direct interaction but the loss of SFP level will cause loss of pool cooling function.

NRC Inspection and Enforcement Bulletin (IE) 84-03 was issued in response to the refueling cavity pool seal failure at the Haddam Neck Nuclear Plant. The review and response to this bulletin, conducted at Millstone Point Unit 3, treated the loss of pool inventory due to a pool seal failure as an event in which the only requirement was to maintain the fuel covered and to provide make up to the SFP. Other potential drain down paths were also reviewed as a result of this bulletin. The review determined that only the reactor cavity drain lines presented a potential drain path of concern. In order to ensure recovery from this event, a plant design change was implemented which installed additional supports on the refueling cavity drain lines to ensure that piping up to the isolation valves in the drains was seismically qualified. The valves are not normally closed during refueling but the applicable Abnormal Operating Procedure (AOP) was revised to direct operators to close the valves upon symptoms of a cavity seal failure. This would also be the response to a failure in the non seismic portion of these lines. This operator action is credited with maintaining the water level above the fuel. Continued operation of the SFP cooling system was not considered as a design basis requirement for these modifications.

On November 16, 1995 an Adverse Condition report (ACR) was initiated by plant personnel due to concerns over the proper sizing of the anti siphon holes in the purification lines. This ACR determined that the holes were adequateiy sized to perform their design function of maintaining at least 10 feet of water above the top of the fuel. The protection of the spent fuel pool cooling system was not a design function of the anti siphon holes and the consequential loss of the spent fuel pool cooling system was not identified.

Subsequent to the October 2, 1996 review, an engineering review of potential drainage paths was performed. The drainage paths reviewed were grouped into four major pathways: SFP purification lines; spent fuel shipping cask area; transfer canal area; and refueling cavity drains.

SFP purification lines

Failure of the Spent fuel pool purification lines to and from the purification pumps can result in drainage of the SFP to below the SFP cooling pump inlet line. The final elevation is greater than 10 feet above the top of the fuel in the SFP. After isolation or repair of the break, the SFP level can be restored by the make up system and the SFP cooling system can then be restarted.

Spent fuel shipping cask area

Potential failures in the shipping cask area are: the failure of the shipping cask gate when the cask area is drained down; failure of the 6"line from the SFP to the shipping cask area (line 176); or failure of the 4" drain line in the cask area (line 42).

The shipping cask gate was seismicly qualified but a review of the calculation has determined that it requires updating since the final design thickness differs from the original. However, the conclusions remain valid.

No calculation for line 176 has been located. The line is a 6" schedule 40 pipe extending approximately 6" into the cask area with a 6" gate valve butt welded to the end. This line was likely seismically qualified by inspection. A preliminary seismic evaluation has determined that the line is qualified.

Failure of drain line 42 in the cask area is not likely since the drain line is laid up dry which precludes a siphon and it is isolated by a blind flange. Failure of the piping between its penetration and the blind flange will not result in a drain down since the penetration and piping are above the normal water level.

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In summary, a drain down of the spent fuel pool into the shipping cask area does not appear to be credible. However, an preliminary evaluation of the consequences of failure of the gate (the worst case failure) shows that the SFP level would drop to below the SFP cooling suction but would remain greater than 10 feet above the fuel.

Transfer canal Area

Similar to the shipping cask area the failures analyzed were: failure of the transfer canal gate; failure of the 6" line from the transfer canal to the SFP (line 202); and failure of the 3" line from the transfer canal dewatering pump (line 175).

The transfer canal gate was seismicly qualified by the same calculation as the shipping cask area gate since the two gates are the same design.

No calculation was found for line 202. However, it is the same arrangement as line 176 in the shipping cask area and a preliminary review has determined that the line is qualified.

Line 175 from the dewatering pump enters the SFP above the normal water level and has an anti siphon hole located above the water line which will prevent back flow of SFP inventory into the transfer canal.

In summary, a drain down of the SFP into a drained transfer canal does not appear to be credible. However, a preliminary evaluation of the consequences of failure of the gate (the worst case failure) shows that the SFP level would drop to below the SFP cooling suction but would remain greater than 10 feet above the fuel. A similar evaluation for simultaneous failure of both gates also determined that the SFP level would remain above the fuel by over 10 feet.

Refuel Cavity drain failure

Failure of the refuel cavity drains was evaluated in response to IE bulletin 84-03. The results of this evaluation were reported to the NRC Staff on November 29, 1984 and supplemented on December 14, 1987. In these reports it was identified that a loss of SFP cooling would result from the failures evaluated. It was also identified that failure of these lines was acceptable since they could be isolated prior to uncovery of the fuel. If the fuel transfer tube gate valve is open, SFP level will drop by approximately 4 feet which will cause the SFP Cooling system to become unavailable.

The SFP cooling system therefore has been determined to be outside of it's design basis as documented within the FSAR. The FSAR states that failure of the purification lines will not impact the fur ction of the SFP cooling system. In fact failure of the purification lines will cause SFP cooling to be lost until such time as repair or isolation of the break can be performed and pool level restored. As diverse make up sources are available, as described in the FSAR, and a safety related seismic make up supply is available using the Service Water System, restoration of SFP level is assured. This event is reportable pursuant to both 10CFR50.72(b)(1)(ii)(B) and 10CFR50.73(a)(2)(ii)(B) as a condition that results in the nuclear power plant being in a condition that is outside the design basis of the plant.

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IV. Corrective Action

The SFP cooling system and/or purification system will be modified to preclude loss of SFP cooling due to loss of the purification system. These modifications will be completed prior to entry into mode 4 from the current outage. Short term actions are being implemented to minimize exposure to potential failures until modifications can be completed. These actions are:

1) The SFP purification system will be shutdown when not required to be in operation. The affected portions of the Spent Fuel Pool (SFP) purification system will be maintained isolated when not in active use.

2) Administrative controls will be established to prevent draining of the shipping cask and transfer canal areas

3) Administrative controls will be established to maintain the isolation valves on the refueling cavity drain lines (3SFC-V998 & 3SFC-V999) closed whenever the refuel cavity is filled and the fuel transfer gate is open until required modifications have been completed to the SFP cooling system and/or purification system.

In addition, the following long term corrective actions will be implemented:

1) The calculations for the SFP gates will be updated to reflect current wall thickness and construction details.

2) The preliminary seismic qualification evaluations for drain down to the transfer canal and shipping cask area will be finalized and approved.

3) A formal evaluations will be generated to demonstrate seismic qualification of SFP cooling lines 176 and 202 from the SFP to the transfer canal and spent fuel shipping cask area

4) The FSAR will be revised to clarify that the flow path for make up from the seismic Category I Refueling Water Storage Tank (RWST) is through the non seismic purification system.

Corrective actions to prevent similar future occurrences relative to the design review problems are as follows:

1) Establish a database linking licensing design basis to the component level for all Maintenance Rule system that are both safety related and risk significant prior to entry into mode 4 from the current outage.

2) Engineering managers and supervisors will review this event and the associated root cause evaluation report with their engineers for lessons learned.

3) At this time work is being assigned in accordance with the qualification records.

4) Establish an effective self assessment and corrective action program within the Millstone Unit 3 organization.

In addition, the FSAR is currently being annotated and verified for accuracy.

V. Additional Information

The unit's response in LER 96-033-00, "Spent Fuel Pool Storage Potentially Outside of Design Basis During Seismic Events as a Result of Boraflex Embrittlement" considered the consequences of this event during development of its corrective actions. The corrective actions include monitoring of boron concentration within the SFP and addition of boron to the pool if make-up from an unborated source is re[illegible] Recovery from a seismic event may require use of the unborated Service Water System. The sampling and addition requirements of LER 96-033-00 will preclude a criticality event in the SFP due to simultaneous failure of the boroflex and the SFP purification system

Similar Events

LER 96-007-00 Containment Recirculation Spray and Quench Spray System Outside Design Basis due to Design Errors

On April 3, 1996, at 13:55 with the plant in Mode 5 at 0-percent power, it was determined that the Containment Recirculation System (RSS) spray piping and supports were not adequately designed for loads resulting from accident temperatures. It was initially determined that the higher RSS temperatures could result from a postulated loss of service water to one or more RSS heat exchangers. It was subsequently determined that: a) unacceptable stresses in the RSS and Quench Spray System (QSS) piping and supports could also result from the design basis accident temperatures inside containment, and b) the original design basis piping analyses utilized support anchor movements which were non-conservative.

LER 96-013-00 Residual Heat Removal System Design Deficiency Due to Non-conservative Original Design Assumption

On June 12, 1996, with the plant in Mode 5 at 0-percent power, an engineering evaluation determined that a design deficiency in the Residual Heat Removal System (RHS) was a condition that was outside the design basis of the plant. A loss of control air could cause the RHS control valves to fail open. If this condition occurred during the initial phase of a plant cool down, the Reactor Plant Component Cooling Water System (CCP) temperatures could go above the 125°F used in the system stress analysis. The Safety Grade Cold Shutdown (SGCS) design requirements specify that the unit be capable of being brought to Cold Shutdown with limited operator action outside the control room. The original plant design did not consider that the RHS flow control valves failing open on a loss of air, could create unacceptably high RHS heat exchanger discharge temperatures.

LER 96-036-00 Safety Related Valves Controlled by Non-Safety Equipment

On September 29, 1996, with the plant in mode 5 of an extended outage, while performing an engineering evaluation, it was concluded that the High Pressure Safety Injection (SIH) and Low Pressure Safety Injection (SIL) systems were subject to degraded performance due to possible mis-positioning of normally closed safety related air operated valves (AOVs). Mis-positioning of these valves was postulated to occur as a result of failures related to non-qualified power and control

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circuits. Several components within the SIH and SIL systems were not properly analyzed for all potential failures in the original plant design.

Manufacturer Data

EIIS System Code:
Fuel Pool Cooling and Purification - DA