NSP

Northern States Power Company

Monticello Nuclear Generating Plant
2807 West Hwy 75
Monticello, Minnesota 55362-9637

May 15, 1997

10 CFR Part 50
Section 50.73

US Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

MONTICELLO NUCLEAR GENERATING PLANT
Docket No. 50-263 License No. DPR-22

LER 97-007
Inadequate NPSH for the ECCS Pumps For Certain
Single Failures During Loss of Coolant Events

The voluntary Licensee Event Report for this occurrence is attached. This report contains no new NRC commitments.

Please contact Tom Parker at (612) 295-1014 if you require further information.

William J Hill

William J Hill
Plant Manager
Monticello Nuclear Generating Plant

c: Regional Administrator - III NRC
Sr Resident Inspector, NRC
NRR Project Manager, NRC
State of Minnesota, Attn: Kris Sanda

Attachment

I/I

IE22

NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 4/30/98
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), 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.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4)
Inadequate NPSH for the ECCS Pumps For Certain Single Failures During Loss of Coolant Events

EVENT DATE (5)			LER NUMBER (6)			REPORT NUMBER (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
04	15	97	97	007	00	05	15	97	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
			20.402(b)		20.405(c)		50.73(a)(2)(iv)		73.71(b)
POWER LEVEL (10)	100 %		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
			20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		(Specify in Abstract below and in Text, NRC Form 366A)
			20.405(a)(1)(iv)		50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)		Voluntary
			20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Tom Parker	612-295-1014

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)	✔	NO				

ABSTRACT (LIMIT TO 1400 SPACES, I.E., APPROXIMATELY 15 SINGLE-SPACED TYPEWRITTEN LINES) (16)
NCR FORM 366 (4-95)

After reviewing an LER from Quad Cities Nuclear Station, it was discovered that the net positive suction head available for the core spray pumps was less than required. Two issues were identified: increased head losses associated with the ECCS suction strainers and a more limiting net positive suction head case. These issues were determined to be applicable to Monticello. An operability evaluation determined that the pumps were still operable. Following evaluation of an additional issue, debris generation from the drywell piping insulation during post design basis loss of coolant accident conditions, the prudency of continued operations was questionable and the plant was brought to cold shutdown. The plant will remain there until new ECCS suction strainers are installed.

Conditions Prior to the Event

The plant was operating at 100% power.

In May of 1996, Bulletin 96-03, Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors, was issued by the NRC staff. In a letter dated November 1, 1996, NSP committed to replace the ECCS Suction Strainers (EIIS Component: STR) in response to NRC Bulletin. The bulletin's concern was potential degradation of the ECCS suction strainers by piping insulation following a LOCA. Steam released by the LOCA could remove insulation from piping in the drywell, and this insulation would be transported to the torus. This insulation could collect on and degrade the ECCS suction strainer performance. A vendor had recently been selected to produce strainers with approximately 60 times more strainer area. These new strainers were to be installed in the next refueling outage in January, 1998.

Event Description

On February 7, 1997, a copy of LER 96-025 from Quad Cities Nuclear Station was reviewed by the plant engineering staff. Quad Cities had determined that the head losses associated with their Emergency Core Cooling System (ECCS) suction strainers was 5.8 feet of head loss per 10,000 gpm rather than 1 foot of head loss per 10,000 gpm. This issue was determined to be potentially applicable to Monticello and a Condition Report (9700446) was initiated to document the evaluation of this LER. Two actions were initiated:

1) determine the head loss per 10,000 gpm for the Monticello strainers, and

2) determine the minimum containment pressure available during a loss of coolant accident

These actions were completed on 4/15/97.

Action 1 Results: The head loss for the Monticello ECCS suction strainers was calculated to be 11.7 feet per 10,000 gpm. This increase in head loss will decrease the available NPSH.

Action 2 Results: The most limiting accident was determined to be a large break loss of coolant accident with the Low Pressure Core Injection (LPCI) Loop Selection Logic selecting the incorrect Recirculation loop. This single failure had not been previously determined to be the most limiting for NPSH. In this case, the Residual Heat Removal (RHR) (EIIS System Code: BO) flow is directed into the broken loop and spills into the containment atmosphere without flowing through the reactor vessel. This cool water rapidly decreases the containment pressure below the 10 psig assumed in the limiting NPSH calculation, decreasing the available NPSH.

Operability Evaluation

On April 15, 1997, another Condition Report (97001188) was initiated to document this condition. An operability evaluation was initiated. The assumption that only 3 of the 4 ECCS suction strainers were available was used in the initial operability evaluation.

During the first three minutes following the large break LOCA, the available NPSH was greater than the required NPSH. During this time, the reactor pressure rapidly decreases and the Core Spray pump (EIIS System Code: BG)(EIIS Component Code: P) refills the core with water. Containment pressure is sufficient to maintain the available NPSH greater than the required NPSH. Between 3 and 10 minutes, the Core Spray pump available NPSH decreases below the required NPSH. At 10 minutes, the limiting NPSH was calculated to be 23 feet, rather than the 47 feet previous calculated:

The increased strainer head loss dropped the available NPSH by 7 feet.
The lower containment pressure dropped the available NPSH by 17 feet.

Available NPSH = 47 feet - (7 feet + 17 feet) = 23 feet

Required NPSH remained = 33 feet

During this period, the core spray pumps were determined to deliver less flow than design but greater than assumed in the LOCA analysis. This was based on testing done by the Bingham Pump Company. Pumps were tested with the available NPSH less than the required by up to 10 feet. This testing was evaluated and found to be applicable to the Monticello Core Spray pumps. The NPSH deficit caused a reduction in flow rate. The pumps were inspected and found to be undamaged following testing.

The testing was used to support the conclusion that the Core Spray pumps would pump less during this time but greater than that assumed in the LOCA accident and remain undamaged.

Following 10 minutes, operator action can be assumed. Operators would recognize severe degradations in flow and take action to reduce flow or turn off unneeded pumps to reduce the flow and increase the available NPSH.

The RHR pumps were found to have more available NPSH than required at all times. The RHR and Core Spray pumps were determined to be operable. This conclusion was discussed with the NRC.

Debris Concerns

Following NRC questions concerning debris generation and the potential for debris increasing strainer flow head loss, the engineering staff: 1) researched NUREG CR-6224, Parametic Study of the Potential for BWR ECCS Strainer Bockage Due to LOCA Generated Debris, and 2) performed strainer clogging calculations. The engineering staff concluded that the debris had the potential to rapidly collect on the strainers and further degrade the available NPSH for the Core Spray pumps. Following evaluation of this additional degradation, the reactor was shutdown on May 9, 1997 and placed in cold shutdown on May 10, 1997.

Cause:

Three issues are involved: identification of the increased ECCS strainer head loss, the identification of a limiting case for NPSH and the debris issue as described in NRC Bulletin 96-03.

1) Identification of the Increased ECCS Strainer Head Loss

   General Electric specified 1 foot head loss per 10,000 gpm for the ECCS suction strainers. The strainer were never verified to meet this requirement.

2) Identification of a Limiting Case For NPSH

The failure of the LPCI Loop Selection Logic had not been identified as a limiting condition for Core Spray NPSH. The effect that the RHR flow would have on reducing containment pressure, if injected directly out the break rather than flowing through the reactor vessel and then out the broken loop, had not been realized.

3) Debris Issue

The debris issue was identified in NRC Bulletin 96-03 and was being addressed.

Analysis of Reportability

The event is being reported as a voluntary LER.

Safety Significance

The compilation of the three issues suggested that in the unlikely event of a large break LOCA with a LPCI Loop Selection Logic failure and strainer clogging, the ECCS flow rate assumed in the safety analyses might not be available. In the unlikely event a LOCA occurred, operator action would ensure proper core cooling by reducing flow to ensure the available NPSH is greater than the required NPSH. However, for the purposes of the safety analyses, no operator action can be assumed for 10 minutes. Since the flows assumed in the safety analysis could not be assured, the plant was shutdown until this issue is resolved.

Actions

Immediate Actions

An operability evaluation was promptly conducted.

Corrective Actions

The plant was brought to cold shutdown and will remain there until the new ECCS suction strainers are installed.

Preventative Actions

New larger ECCS suction strainers will be installed.

Failed Component Identification - None

Similar Events - None

SIMPLIFIED RHR SYSTEM SHOWING LARGE BREAK LOCA