Northern States Power Company

Prairie Island Nuclear Generating Plant

1717 Wakonade Dr. East
Welch, Minnesota 55089

November 22, 1994

10 CFR Part 50
Section 50.73

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

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
Docket Nos. 50-282 License Nos. DPR-42
50-306 DPR-60

Design Basis Reconstitution Effort Uncovered
Design Deficiency in Containment Isolation Function

The Licensee Event Report for this occurrence is attached. In the report, we made no new NRC commitments.

This event was reported via the Emergency Notification System in accordance with 10 CFR Part 50, Section 50.72, on October 29, 1994. Please contact us if you require additional information related to this event.

Michael Wadley for

Roger O Anderson
Director
Licensing and Management Issues

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

Attachment

280018

JE22

NRC FORM 366 (5-92) — U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

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.

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Prairie Island Nuclear Generating Plant U1	05000 282	1 OF 4

TITLE (4) Design Basis Reconstitution Effort Uncovered Design Deficiency in Containment Isolation Function

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
									Prairie Island U2	05000 306
10	29	94	94	-- 09 --	00	11	22	94	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	N
POWER LEVEL (10)	100

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)
	20.405(a)(1)(i)		50.36(c)(1)	X	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)		
	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)
Arne A Hunstad	612-388-1121

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)

Design basis reconstitution of the containment integrity and containment isolation functions uncovered a design deficiency.

The chemical and volume control system letdown line from the reactor coolant system has redundant containment isolation valves; inboard, the 3 letdown orifice isolation valves, and outboard, the letdown isolation valve. The RHR to letdown line, which is used for shutdown purification and overpressure protection of the RHR piping, ties in to the letdown line between the redundant containment isolation valves.

Following a LOCA, and after depletion of the refueling water storage tank, it is necessary for the suction of the RHR pumps to be transferred to the containment sump to provide long term cooling. Once on long term recirculation, failure of the outboard letdown line containment isolation valve in the open position would allow the RHR system to pump containment sump water to the letdown system outside of containment. Assuming no operator action, there is a potential to exceed the USAR offsite dose calculation values. Additionally, with containment sump water in the auxiliary building in locations other than those evaluated, it may not be possible to fully implement all the emergency procedures within the assumed radiation dose.

EVENT DESCRIPTION

Design basis reconstitution of the containment integrity and containment isolation functions determined that failure of a single containment isolation valve could provide a path, during long-term recirculation following a loss-of-coolant accident (LOCA), for containment sump water out of containment via the residual heat removal (RHR) to letdown line.

The chemical and volume control system letdown line from the reactor coolant system has redundant containment isolation valves; inboard, the 3 letdown orifice isolation valves, and outboard, the letdown isolation valve (Unit 2 has equivalent valves). The RHR to letdown line, which is used for shutdown purification and overpressure protection of the RHR piping, ties in to the letdown line between the redundant containment isolation valves. (See attached Figure.)

Following a loss of coolant accident and after depletion of the refueling water storage tank, it is necessary for the suction of the RHR pumps to be transferred to the containment sump to provide long term cooling. Once on long term recirculation, failure of the outboard letdown line containment isolation valve in the open position would allow the RHR system to pump containment sump water to the letdown system outside of containment. Assuming no operator action, there is a potential to exceed the USAR offsite dose calculation values. Additionally, with containment sump water in the auxiliary building in locations other than those evaluated, it may not be possible to fully implement all the emergency procedures within the assumed radiation dose.

Procedures have been revised to close the manual valve downstream of the outboard letdown isolation valve in preparation for transfer to recirculation following a LOCA.

CAUSE OF THE EVENT

Cause of the event is apparent oversight during original plant design.

ANALYSIS OF THE EVENT

The event is reportable pursuant to 10CFR50.73(a)(2)(v) since a single failure could provide a path through the containment boundary while on recirculation flow after a LOCA.

The probability of a LOCA coincident with the failure of the outboard letdown line containment isolation valve to close is very low. Operator action would have been needed to mitigate the postulated event. The indications that had been available to alert the operator of the flow of containment sump water into the auxiliary building were:

VCT high level alarm
VCT high pressure alarm
Letdown line high radiation alarm

Various area radiation monitors and alarms
Control board valve position lights
Containment isolation monitor lights
Emergency response computer containment isolation valve monitoring

The indications in the control room were sufficient to allow the operator to diagnose and respond to the event. Alternate valves with remote control from the control room were available to isolate the flow path.

Additionally, the letdown system was maintained in a leak tight condition which reduced the potential for any release to the outside environment.

Health and safety of the public were unaffected by this event.

CORRECTIVE ACTION

Procedures have been revised to close the manual valve downstream of the outboard letdown isolation valve in preparation for transfer to recirculation following a LOCA.

FAILED COMPONENT IDENTIFICATION

None.

PREVIOUS SIMILAR EVENTS

The Design Basis Reconstitution effort has yielded other reportable events, but this is the only one involving containment isolation capability.

NRC FORM 366A (5-92) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 5/31/95
LICENSEE EVENT REPORT (LER) TEXT CONTINUATION	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)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Prairie Island Unit 1	05000 282	94	-- 09 --	00	4 OF 4

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