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10 CFR 50.73

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

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
Docket No. 50-282 License No. DPR-42

LER 1-02-02: Unit #1- Manual Reactor Trip and Auxiliary Feedwater Pump Start During Planned Shutdown

The Licensee Event Report for this occurrence is attached. In the report, we made no new NRC commitments. Please contact us if you require additional information related to this event.

This event was reported via the Emergency Notification System in accordance with 10 CFR Part 50, Section 50.72, on November 15, 2002.

Mano K. Nazar
Site Vice President
Prairie Island Nuclear Generating Plant

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

Attachment

IE22

NRC FORM 366 (1-2001) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 6-30-2001
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory information collection request 50 hours Reported lessons learned are incorporated into the licensing process and fed back to industry Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U S Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@nrc gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503 If a means used to impose information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection

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

TITLE (4)
Manual Reactor Trip and Auxiliary Feedwater Pump Start During Planned Shutdown

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
11	15	02	02	– 02 –	00	01	13	03	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	1
POWER LEVEL (10)	12

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply) (11)

	20 2201(b)		20.2203(a)(3)(ii)		50 73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
	20 2201(d)		20 2203(a)(4)		50 73(a)(2)(iii)		50.73(a)(2)(x)
	20 2203(a)(1)		50 36(c)(1)(i)(A)	X	50 73(a)(2)(iv)(A)		73.71(a)(4)
	20 2203(a)(2)(i)		50 36(c)(1)(ii)(A)		50 73(a)(2)(v)(A)		73.71(a)(5)
	20 2203(a)(2)(ii)		50 36(c)(2)		50 73(a)(2)(v)(B)		OTHER
	20 2203(a)(2)(iii)		50 46(a)(3)(ii)		50 73(a)(2)(v)(C)		Specify in Abstract below or in NRC Form 366A
	20 2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50 73(a)(2)(v)(D)		
	20 2203(a)(2)(v)		50.73(a)(2)(i)(B)		50 73(a)(2)(vii)		
	20 2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50 73(a)(2)(viii)(A)		
	20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Robert Alexander	651-388-1121

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
B	SB	BLL	P090	Y	B	SM	LT	M040	N

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 November 15, 2002, Unit 1 was undergoing an orderly shutdown in preparation for a planned refueling outage. Prior to the shutdown, an existing problem with the steam generator blowdown flash tank vent line bellows to 13A Feedwater Heater had been identified and contingency plans were in place for monitoring the level of the 13A Feedwater Heater during the load decrease. At approximately 2020 during the load decrease, Operators made the decision as planned, to manually trip the reactor when it was determined that a Hi Hi Level in the heater, could not be reduced. Reactor power at the time of the trip was 12%.

During performance of reactor trip recovery procedures, 11 Turbine Driven Auxiliary Feedwater Pump auto started at 2055 when the running Main Feedwater Pump was secured.

Following investigation and repair to the steam generator blowdown flash tank vent line bellows and completion of all refueling outage activities, the unit was returned to service on December 6, 2002.

EVENT DESCRIPTION

On November 15, 2002, Unit 1 was undergoing an orderly shutdown in preparation for a planned refueling outage. Several months prior to the refueling outage shutdown, the bellows[1] on the steam generator blowdown flash tank vent line[2] to 13A Feedwater Heater had apparently failed and difficulties in controlling level in 12A and 13A Feedwater Heaters[3] were expected. Contingency plans were in place for monitoring 13A Feedwater Heater levels during the load decrease. At approximately 45% power the 2 inch drain valves for 13A and 12A Feedwater Heaters were opened. At approximately 15% power, the level in the 13A Feedwater Heater was noted as rising and the dump valve to the condenser[4] did not open. Operations personnel directed engineering support to investigate and with the unit at approximately 12% power, they observed the 13A Feedwater Heater gage glass[5] full and the dump valve closed. Operators then opened the 13A Feedwater Heater dump valve to approximately 50% and engineering verified the valve open. At that time the 13A Feedwater Heater Hi Hi level alarm[6] started to flash in and out. Operations and Engineering personnel continued to monitor 13A Feedwater Heater level. At 2020 hours, it was decided to manually initiate a reactor trip at approximately 12% power when the 13A Feedwater Heater level alarm came in solid and would not clear. Integrated plant response to the manual trip was normal, except as noted below.

After the reactor trip, while progressing through the "Reactor Trip Recovery" procedure, both the 11 Turbine Driven Auxiliary Feedwater Pump (TDAFW)[7] and the 12 Motor Driven Auxiliary Feedwater Pump (MDAFW) were found not running. This was the expected plant response because the plant had tripped from a low power level and because the Steam Generator levels were at approximately 35%. The decision was made to only start the 12 MDAFW Pump and that starting 11TDAFW Pump would not be necessary to maintain Steam Generator[8] level. When the running Main Feedwater Pump[9] was stopped as part of the trip recovery, the 11 TDAFW pump auto started. The pump automatically started because the AFW selector switch for the pump had not been positioned to Shutdown Auto before stopping the MFW pump. The 11 TDAFW Pump was secured and Steam Generator levels were maintained with 12 MDAFW Pump.

CAUSE OF THE EVENT

The apparent cause of the high 13A Feedwater Heater level, which necessitated a manual reactor[10] and turbine[11] trip, was mechanical failure of the steam generator blowdown flash tank vent line bellows to

[1] EIIS System Identifier: SM Component Identifier: BLL
[2] EIIS System Identifier: SB Component Identifier: PSP
[3] EIIS System Identifier: SM Component Identifier: HX
[4] EIIS System Identifier: SG Component Identifier: LCV
[5] EIIS System Identifier: SM Component Identifier: LI
[6] EIIS System Identifier: SM Component Identifier: LA
[7] EIIS System Identifier: BA Component Identifier: P
[8] EIIS System Identifier: SB Component Identifier: HX
[9] EIIS System Identifier: SJ Component Identifier: P
[10] EIIS System Identifier: AB Component Identifier: RCT
[11] EIIS System Identifier: TA Component Identifier: TRB

13A Feedwater Heater. This failure caused the 13A Feedwater Heater to have insufficient pressure to aid in equilibrium balance between the cascading train of feedwater heaters. With the 13A Feedwater Heater in vacuum, moisture was essentially trapped and continued to build until the turbine was manually tripped and the extraction steam isolated. During the shutdown, the feedwater heater dump valve did not open and the high level alarm did not actuate. However, because the feedwater heater was in vacuum, the drain would not have functioned and the manual trip would have been required regardless. The cause of the dump valve and alarm failure is discussed in the "Corrective Actions" discussed below.

The cause of the 11 Turbine Driven AFW Pump auto start is attributed to insufficient guidance in the "Reactor Trip Recovery" procedure. This procedure assumes the AFW pumps have automatically started on low Steam Generator level during a reactor trip. For reactor trips occurring at low power levels this may not be the case and directs the control room operators to start the AFW pump(s). During this occurrence, operators had started the 12 Motor Driven AFW Pump and recognized that operation of the 11 Turbine Driven AFW Pump would not be necessary. Since the procedure assumes both 11 and 12 AFW pumps started, the procedure does not address the auto-start logic. Thus the 11 TDAFW pump automatically started, when the remaining Main Feedwater Pump was secured.

ANALYSIS OF THE EVENT

This event is reportable under 10CFR50.73(a)(2)(iv)(A) as a manual actuation of the reactor protection system and unplanned automatic actuation of an auxiliary feedwater pump. The health and safety of the public were unaffected since the plant systems responded as designed following the actuations.

RISK SIGNIFICANCE

A Risk Significance evaluation was completed for this event. Using the current Level 1 PRA model, the contribution of normal reactor trip-initiated events (assuming their normal initiating event frequency) is 9.168E-7/yr, which is less than 5% of the total Core Damage Frequency (CDF). With the event frequency set to logical (TRUE), the calculated Conditional CDF is 1.671E-6/yr. Using 12 hours as the time the plant was in the trip condition, the Conditional Core Damage Probability (CDP) is 2.29E-09. The Conditional CDP is the most appropriate risk measure for describing the risk incurred due to this event. In general, both Conditional CDP and Conditional Large Early Release Probability (LERP) should be calculated to assess the risk significance due to an accident event. Since the Conditional CDP for this event is calculated to be 2.29E-9 that is much lower than "1E-6" - the cutoff value for risk significance consideration and this event has no impact on Conditional LERP, this is a non-risk significant event.

SAFETY SYSTEM FUNCTIONAL FAILURE PERFORMANCE INDICATOR

Since no system, structure, or component was inoperable, this event did not involve either a partial or complete loss of a safety system function and is not reportable per 10 CFR 50.73(a)(2)(v).

CORRECTIVE ACTIONS

Actions Taken

1) Both the interior condenser and exterior condenser expansion joint bellows have been replaced on the steam generator blowdown flash tank vent line.

2) The cause of Magnetrol level transmitter problem was determined to be two missing screws that hold the centering plate for the float stem. These were replaced. Inspections were performed on the 11A, 12A, and 14A heater transmitters with no problems identified.

Planned Actions to Prevent Recurrence

1) The "Reactor Trip Recovery" procedure will be revised to provide improved guidance to the operators when transitioning between Main and Auxiliary Feedwater at low power levels.

2) The failure and maintenance history for expansion joint bellows will be reviewed to determine the expected service life for this application. Based on the results of the review, a replacement frequency will be established and included in the Preventative Maintenance Program.

FAILED COMPONENT IDENTIFICATION

1) The level transmitter failure discussed in the LER is a Magnetrol; Model No. 249-C-SIM4D; Serial No. 487003

2) The expansion joint bellows is a Pathway Bellows Inc.; Model No. BSWW15012L

PREVIOUS SIMILAR EVENTS

The Unit 2 turbine trip / reactor trip events of November 9, 1998 (LER 1998-05) and April 28, 2000 (LER 2000-01) are considered to be similar.