NSP

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April 22, 1993

Report Required by
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

Reactor Protection System Actuation
From High Pressure Caused By Inadequate Procedure

The Licensee Event Report for this occurrence is attached. This report contains the following new NRC commitments.

1. LER 93-006 and the lessons learned from the event will be presented in Engineering Technical Staff Continuing Training. This training will emphasize the importance of procedure development and technical review.

2. LER 93-006 will be presented in Licensed Operator Requalification Training. The training will include use of the simulator to demonstrate the proper operation of the Steam Pressure Control System.

3. The simulator software change process will be revised to include independent review by a qualified individual.

Please contact Marv Engen, Sr Licensing Engineer, at (612) 295-1291 if you require further information.

Roger O. Anderson

Roger O Anderson
Director
Licensing and Management Issues

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

Attachment

IE22 / 1

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)
MONTICELLO NUCLEAR GENERATING PLANT	05000 263	1 OF 6

TITLE (4)
Reactor Protection System Actuation From High Pressure Caused By Inadequate Procedure

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
03	23	93	93	-- 006 --	00	04	22	93	FACILITY NAME	DOCKET NUMBER 05000

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

OPERATING MODE (9)	N	20.402(b)	20.405(c)	X 50.73(a)(2)(iv)	X 73.71(b)
POWER LEVEL (10)	10%	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 (Specify in Abstract below and in Text, NRC Form 366A)
		20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
		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)
Steve Engelke, Superintendent, Electrical & I&C Engineering	(612) 295-1329

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
B	AA	SOL	A609	Y					

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)

During startup from a refueling outage a Reactor Protection System trip from high reactor pressure occurred. The trip occurred while performing the main turbine Stop Valve tightness test. The cause of the event was an inadequate procedure. A contributing factor was inadequate knowledge by the individuals involved in the procedure revision and incorrect simulator modeling. During the event one control rod scram solenoid failed to operate resulting in a slow rod insertion time. The Turbine Stop and Control Valve Tightness Test procedure has been revised. The plant simulator was modeled to reflect actual operation. The failed scram solenoid pilot valve was replaced and all scram solenoid pilot valves were verified to function properly. This event and the lessons learned from the event will be presented in Engineering Technical Staff Continuing Training and License Operator Requalification training.

REQUIRED NUMBER OF DIGITS/CHARACTERS
FOR EACH BLOCK

BLOCK NUMBER	NUMBER OF DIGITS/CHARACTERS	TITLE
1	UP TO 46	FACILITY NAME
2	8 TOTAL 3 IN ADDITION TO 05000	DOCKET NUMBER
3	VARIES	PAGE NUMBER
4	UP TO 76	TITLE
5	6 TOTAL 2 PER BLOCK	EVENT DATE
6	7 TOTAL 2 FOR YEAR 3 FOR SEQUENTIAL NUMBER 2 FOR REVISION NUMBER	LER NUMBER
7	6 TOTAL 2 PER BLOCK	REPORT DATE
8	UP TO 18 -- FACILITY NAME 8 TOTAL -- DOCKET NUMBER 3 IN ADDITION TO 05000	OTHER FACILITIES INVOLVED
9	1	OPERATING MODE
10	3	POWER LEVEL
11	1 CHECK BOX THAT APPLIES	REQUIREMENTS OF 10 CFR
12	UP TO 50 FOR NAME 14 FOR TELEPHONE	LICENSEE CONTACT
13	CAUSE VARIES 2 FOR SYSTEM 4 FOR COMPONENT 4 FOR MANUFACTURER NPRDS VARIES	EACH COMPONENT FAILURE
14	1 CHECK BOX THAT APPLIES	SUPPLEMENTAL REPORT EXPECTED
15	6 TOTAL 2 PER BLOCK	EXPECTED SUBMISSION DATE

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

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

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LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Monticello Nuclear Generating Plant	05000 263	93	- 006 -	00	2 OF 6

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

Description

On March 23, 1993, at 0618 hours, during startup from a refueling outage, a Reactor Protection System (EIIS System: JC) trip from high reactor pressure occurred. With the reactor at about 10% rated power and 940 psig, a surveillance procedure, Main Turbine Stop and Control Valve Tightness Test, was being performed. The Steam Pressure Control System (EIIS System: JI) was controlling reactor pressure using the Main Steam Bypass Valves (EIIS Component: V). The Mechanical Pressure Regulator was controlling with the Load Limit set at 100%. The Main Turbine (EIIS Component: TRB) was at 1800 RPM under control of the Speed Load Changer. In the performance of the procedure the stop valves are closed and it is verified that the turbine decelerates to 1200 RPM. When the Stop Valves were closed the turbine began to decelerate and reactor pressure began to increase slowly. After about 110 seconds the Bypass Valves began to close and reactor pressure began increasing more rapidly. A full scram on high reactor pressure occurred 137 seconds after the Stop Valves were closed. One safety relief valve (EIIS Component: RV) operated in the Low-Low Set mode. Immediately following the scram the operator noticed that all control rods (EIIS System: AA) were not full in as indicated by the one-rod-out permissive light. The Rod Worth Minimizer Display indicated one control rod not full in. Shortly there-after all control rods indicated full in by all indications. Subsequent review of control rod scram times recorded by the Rod Worth Minimizer revealed that one control rod exhibited a delayed scram. All other systems functioned as designed. The immediate and subsequent procedure actions for the scram were performed.

This event resulted in an automatic trip of the Reactor Protection System and is reportable per 10CFR50.73(a)(2)(iv).

Cause

Investigation of the event determined that the Steam Pressure Control system performed as designed.

To understand the following explanation, refer to the attached Figure 1, Main Steam Pressure Control Functional Control Diagram. Stop Valve closure terminated steam flow to the turbine. This initiated a slow increase in reactor pressure and allowed the turbine to decelerate. The increase in reactor pressure resulted in a slowly increasing output from the Mechanical Pressure Regulator. The decrease in turbine speed resulted in an increasing

output from the Speed Governor. Initially, the output of low value gate 3 (LVG3 on Figure 1) was limited by the speed governor signal. The difference between the pressure signal and the output of LVG3, minus a bias of 3%, (output of E4) controlled the opening of the bypass valves. Since the output of LVG3 was increasing at about the same rate as the pressure signal, the bypass valves did not respond to the initial slow pressure increase. When the pressure signal reached the setting of the Flow Limit, the pressure signal applied to E4 was held at the limited value (100%). As the speed signal to LVG3 continued to increase, the difference signal (output of E4) decreased, and the bypass valves began to close. This caused a faster pressure increase.

The cause of the event was an inadequate procedure. The procedure was revised on March 3,1993. During the revision process a step to place the Load Limit to 25% to limit the speed governor signal was deleted. The revised procedure was validated on the plant simulator, but subsequent investigation has determined that the simulator did not model the control system correctly. A contributing factor was inadequate knowledge by the Licensed Operators and system engineer involved in the procedure revision and incorrect simulator modeling.

The single control rod delayed scram was determined to be caused by a scram solenoid pilot valve (EIIS Component: SOL) which did not function. The solenoid was one of 154 replaced during the 1993 refueling outage. Post maintenance testing had verified proper operation prior to start-up. The valve was disassembled and inspected. Separation of the solenoid core from the core spring had occurred due to improper assembly by the manufacturer. Separation cannot occur with proper assembly. Bench testing confirmed that, after improper assembly, failure would occur after several valve cycles.

Analysis

This event represents an unnecessary challenge to the Reactor Protection System and an unnecessary plant transient. All systems responded as designed and all parameters remained within analyzed values with the exception of the delayed control rod scram. Since the Stop Valve Tightness test is only performed at low power levels this event could not have occurred under other conditions which would have resulted in more severe consequences. There were no consequences to the health and safety of the public.

The failure of the scram solenoid pilot valve was determined to be an isolated case. All 22 spare solenoid valves were inspected and were found to be properly assembled. Each installed solenoid was cycled five times and verified to function properly.

A single rod delayed scram which occurs due to the action of the backup scram valves represents an insignificant decrease in scram reactivity insertion rate. The Technical Specifications contain Limiting Condition for Operations (LCOs) governing core average and four rod array scram times. These LCOs were not exceeded.

Corrective Actions

The following actions have been completed:

1. The Turbine Stop and Control Valve Tightness Test procedure has been revised.

2. The plant simulator was revised to properly model operation of the Steam Pressure Control System.

3. The failed scram pilot solenoid valve was replaced and all installed scram pilot solenoid valves were cycled and verified to function properly.

4. All remaining solenoids of this type in stock were inspected and no other problems were found.

The following actions will be completed:

1. This event and the lessons learned from the event will be presented in Engineering Technical Staff Continuing Training. This training will emphasize the importance of procedure development and technical review.

2. This event will be presented in Licensed Operator Requalification Training. The training will include use of the simulator to demonstrate the proper operation of the Steam Pressure Control System.

3. The simulator software change process will be revised to include independent review by a qualified individual.

Additional Information

Failed Component Identification

Scram Pilot Solenoid Valve
Manufacturer: ASCO
Model: HVA 90-405

Previous similar events

None

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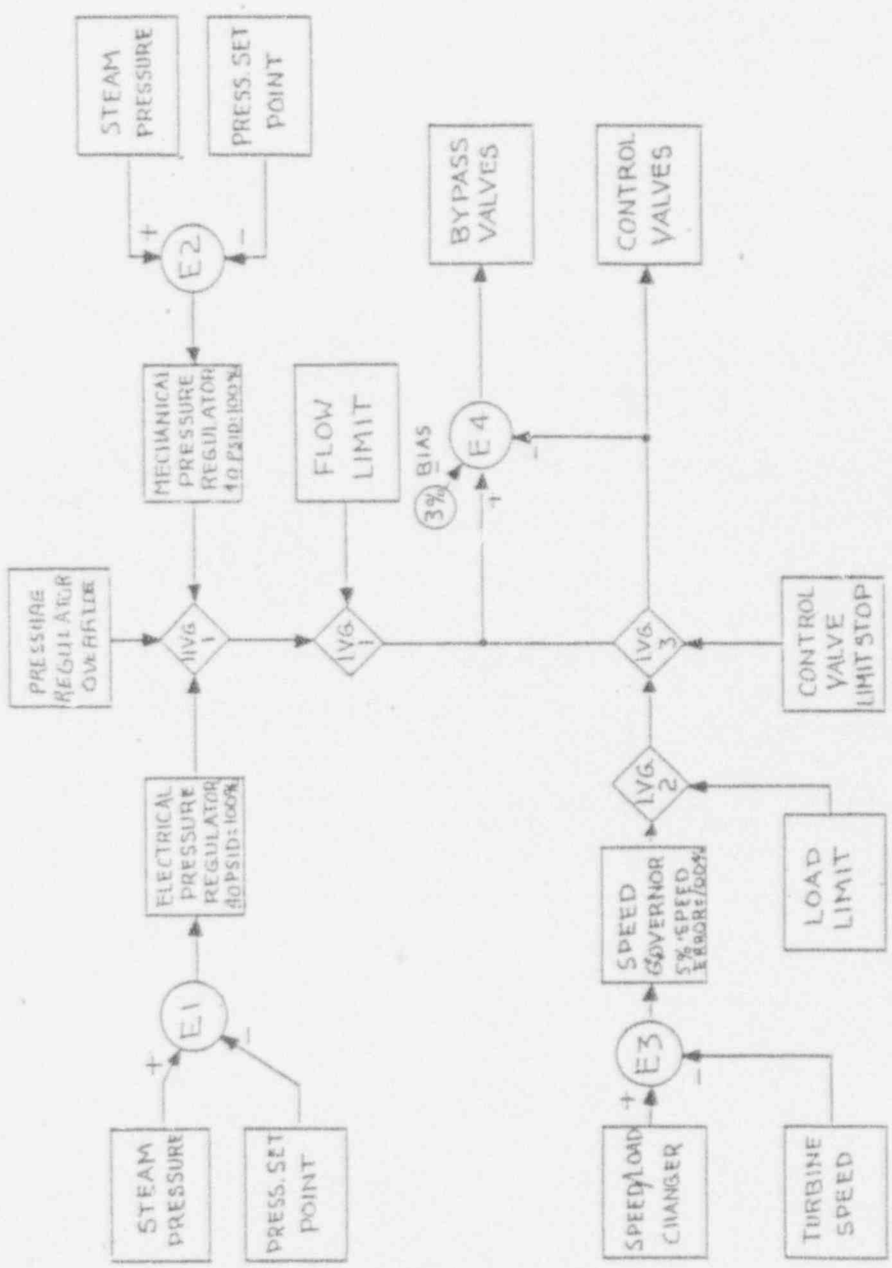

FIGURE 1

Main Steam Pressure Control Functional Control Diagram