NRC FORM 366 (4-95)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 04/30/98
LICENSEE EVENT REPORT (LER)		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.

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TITLE (4)
Unit 3 Shutdown Cooling Isolation from an invalid Low Reactor Water Level Signal due to Procedural Deficiency

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 N/A	DOCKET NUMBER
02	20	99	99	001	00	03	19	99	FACILITY NAME N/A	DOCKET NUMBER

OPERATING MODE (9)	4	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)	000		20.2203(a)(2)(i)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
			20.405(a)(1)(ii)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
			20.2203(a)(2)(ii)		20.2203(a)(4)	X	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)
Ralph M. Fenili (Operations Staff) ext.: 2917	**(815) 942-2920**

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)

At 0414 hours on February 20, 1999, Operations had successfully completed DOS 0201-02, Unit 3 ASME B & PV Code 1000 psi System Leakage Test / Hydrostatic Test. Following the testing, the two Equipment Attendants were tasked with the return to service of various reactor pressure instruments on the 2203-5 rack, located in the Unit 3 Reactor Building. As the two Equipment Attendants valved in the reactor pressure instruments, a reactor scram signal and Groups 2 and 3 isolations occurred causing a Shutdown Cooling (SDC) isolation for sixteen minutes. The Equipment Attendants were contacted by radio, work was stopped and an investigation initiated. A lack of backfill pressure in the instrument reference line, between the instrument isolation valve and the actual instrument, had caused a momentary oscillation in reference leg pressure. The reactor pressure instruments being returned to service share a common sensing line with reactor level instruments. As a result of the small pressure oscillation seen when the isolation valve was opened, the level instruments sensed the momentary low level condition resulting in initiation of the reactor scram and Groups 2 and 3 Isolations from low reactor water level. Actual reactor level remained at +32 inches for the duration of the event.

The cause was determined to be a procedural deficiency, evident by the failure for the governing procedure to adequately control sequencing of the task. Corrective actions include revision to Unit 2 and 3 hydro procedures, evaluation of the feasibility of electronically isolating the pressure switches in lieu of manual valving out of the pressure switches, review of similar procedures for related issues, and evaluation of the event for future training.

This LER is being submitted pursuant to 10 CFR 50.73(a)(2)(iv), which requires the reporting of any event or condition that results in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection system (RPS). Based on plant conditions at the time of the event, the safety significance was determined to be minimal.

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

PLANT AND SYSTEM IDENTIFICATION:

General Electric - Boiling Water Reactor - 2527 MWt rated core thermal power

Energy Industry Identification System (EIIS) Codes are identified in the text as [XX] and are obtained from IEEE Standard 805-1984, IEEE Recommended Practice for System Identification in Nuclear Power Plants and Related Facilities.

EVENT IDENTIFICATION:

Unit 3 Shutdown Cooling Isolation from an invalid Low Reactor Water Level Signal due to Procedural Deficiency

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: 3	Event Date: February 20, 1999	Event Time: 0414 CST
Reactor Mode: 4	Mode Name: Cold Shutdown	Power Level: 0

Unit 3 was in Mode 4 following the successful completion of DOS 0201-02, Unit 3 ASME B & PV Code 1000 psi System Leakage Test / Hydrostatic Test. At the time of the event various plant instruments were Out-Of-Service (OOS), with their subsequent return to service resulting in the ESF actuation.

B.1 DESCRIPTION OF EVENT:

On the midnight shift of February 20, 1999, following the successful completion of DOS 0201-02, Unit 3 ASME B & PV Code 1000 psi System Leakage Test / Hydrostatic Test, Operations prepared to realign the plant in accordance with the surveillance's Checklist A, Test In Progress Card Checklist. A pre-job briefing was held with two Equipment Attendants [Non-Licensed], assigning them the task of clearing the hydro surveillance informational cards in the Unit 3 Reactor Building. The briefing discussed the realignment of components in the 2203-5 and 6 racks, stating that the Equipment Attendants would be performing valving on sensitive instruments and that caution was to be used. Upon completion of the briefing, the Equipment Attendants proceeded to the Unit 3 Reactor Building to begin the plant realignment.

The Equipment Attendants arrived at the 2203-5 rack and began clearance of reactor pressure instruments that had been isolated and vented to atmosphere for the hydro test, in accordance with the testing procedure checklist. Instruments returned to service included:

- 3-263-51A, U3 MSIV Lo Vac Byp Rx Press PS.
- 3-263-55A, U3 Rx Hi Press Scram PS.
- 3-263-55B, U3 Rx Hi Press Scram PS.
- 3-263-53A. U3 Isol Cond Hi Press Scram PS.
- 3-263-53B, U3 Isol Cdsr Init PS.
- 3-263-20A, U3 Rx Press Xmtr ATWS PT.
- 3-263-20C, U3 Rx Press Xmtr ATWS PT.

For each instrument to be returned to service, the Equipment Attendants replaced the instrument test line vent plug, closed the instrument vent plug isolation valve, and then unisolated the instrument reference leg isolation valve. (Attachment A - typical pressure switch configuration) For each instrument, one Equipment Attendant would perform a verification of the component alignment concurrent with a second Equipment Attendant who performed the component manipulation.

Upon completion of all valving manipulations for the 2203-5 rack, the Equipment Attendants exited the instrument rack area, at which time they were notified by radio that a reactor scram and group isolations had occurred. Additionally, further manipulations were halted until completion of an investigation into the cause for the ESF actuations.

B.2 Investigation of the Issue

The reactor pressure instruments on the 2203-5 rack share a common sensing line with the reactor level instruments. Normally, with reactor pressure at zero pounds, there is a 13-pound pressure seen at each instrument due to the height of water present in the instrument reference line. When the Equipment Attendants replaced the cap on the instrument vent line, a pressure of zero existed in the pressure instrument and associated piping. As the operator proceeded to valve the reactor pressure instrument into service, a sudden pressure drop occurred as the 13-pound pressure equalized with the zero pound instrument pressure. The system saw the overall effect as a momentary pressure oscillation in the reference piping. This small pressure oscillation was determined to be sufficient to affect the sensitive level instruments, causing the instruments to see a momentary (indicated) level drop causing the reactor scram and Group Isolations to occur.

This issue of utilizing operations personnel to return these instruments to service was discussed with a sampling of Instrument Maintenance personnel. Normally, recovery of these instruments is the responsibility of the Instrument Maintenance group, who utilize specific procedures and instruction to recover the instruments. All Instrument Maintenance personnel interviewed were aware of a need to utilize a hand pump to pressurize the small section of piping (approximately 6 inches in length) between the instrument isolation valve and the reactor pressure instrument, raising the pressure to 13-pounds prior to opening of the isolation valve. Historically Instrument Maintenance personnel would accompany the operator to the instrument rack to perform pressurization of the instrument, after which the operator would open the instrument isolation valve.

The procedure utilized for the Unit 3 hydro (DOS 0201-02), and the Unit 2 equivalent procedure (DOS 0201-01) were reviewed finding the existence of a caution step which stated "RPV level must be >60 inches indicated before valving in Reactor pressure and level instrumentation to prevent possible trip actuations due to common sensing line spiking". This caution was provided to the operator at the point where depressurization of the reactor vessel begins following completion of the actual hydro of the reactor vessel. After depressurization, level reduction began from an initial value of approximately +467 inches. The level reduction continued for the next 4 hours and was terminated at +40 inches. Procedural guidance stated to lower level to "Normal Operating Level". As the Control Room worked to establish the new level band of +30 inches, the Equipment Attendants were attending the job briefing in the Work Execution Center for the recovery of the plant alignment, coordinating completion of surveillance's Attachment A and the OOS in place for configuration control. As a result of the inappropriate location and structuring of the caution and subsequent steps, the Operations personnel planning for recovery of plant components remained unaware that their upcoming actions could cause the instrument trips to occur. Also, at the point in the procedure where the caution occurs, reactor level was +467 inches, meeting the conditions. The caution step, being inappropriately placed, provided no value to the operator at the controls. Evaluation of the surveillance concluded that it was lacking in three specific areas: (1) failing to procedurally control the evolution for lowering of reactor level through placement of a hold point above the +60 inch caution, (2) directing the performance of Attachment A after attaining normal operating level, and (3) failure to place a caution within the checklist prior to manipulation of components on the 2203-5 and 2203–6 racks.

C. CAUSE OF EVENT:

The cause of this event was determined to be a defective procedure (NRC Cause Code D), as demonstrated by the failure for the governing surveillance to adequately control sequencing of this task. Though the need for specific mitigating action was required to assure prevention of an ESF actuation, the surveillance inappropriately had placed the caution at a point where the condition was met. After a period of time had lapsed and recovery of plant components was being performed, no further awareness to the caution step existed.

Contributing to the event was a knowledge deficiency (NRC Cause Code E) regarding the existence of a significant pressure within the reference line, and the need for pre-pressurization of the instrument prior to opening of the instrument isolation valve.

D. SAFETY ANALYSIS:

Since the pressure oscillation to the Unit 3 level instruments was only momentary, the loss of heat removal capability was experienced for only sixteen minutes, terminated by Operations personnel who promptly unisolated and restarted the Shutdown Cooling [BO] system. The unit had a low decay heat load, having just completed refueling operation, and no temperature rise was noted during the event. Should the unit had experienced rising temperatures as a result of delays in restoring Shutdown Cooling, Operational procedures were in place which offered various methods for alternate heat removal. Since Shutdown Cooling could be quickly restored and alternate heat removal systems were available for use, the safety significance was minimal.

E. CORRECTIVE ACTIONS:

1. DOS 0201-01 and DOS 0201-02 will be revised to incorporate adequate guidance requiring the assistance of Instrument Maintenance to recover instruments. This will include establishing a procedural step to stop lowering reactor level at +70 inches for the recovery of sensitive instruments, and placement of the "Caution" statement in the Attachment A prior to performing tasks for the 2202(3)-5 and –6 racks. (2491809900101)
2. Operations will identify and review outage surveillances that may require non-operations support to return instruments/components to service. An action plan will be created to revise the surveillances prior to their next performance. (2491809900102)
3. Operations Management will determine where ownership boundaries for in-plant instrument manipulations exist. (2491809900103)
4. An evaluation of the feasibility of electronically isolating the pressure switches in lieu of manual valving out of the pressure switches will be performed. (2491809900104)
5. The Operations Curriculum Review Committee will review this event and determine actions to be taken regarding Initial and Continuing Licensed Operator Training. This will include sensitivity to performing valving on the 2202(3)-5 and –6 racks. (2491809900105)

F. PREVIOUS OCCURRENCES:

None.

G. COMPONENT FAILURE DATA:

None.

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Attachment A
Typical Pressure Switch Configuration