NRC FORM 366 (6-1998) U.S. NUCLEAR REGULATORY COMMISSION

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LICENSEE EVENT REPORT (LER)

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

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TITLE (4)

Automatic Reactor Trip Due to Flow Transmitter Low Flow Bistable Actuation on Reactor Coolant Loop 3 While Unisolating Another Transmitter as a Result of an Inadequate Work Package

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
03	09	1999	1999	-- 003 --	01	07	2	1999		05000
									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)

	20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
	20.2203(a)(2)(i)		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)
Troy Schaefer, Instrumentation & Control Engineer	(914) 736-8797

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
X	AB	ISV	W165	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)

On March 9, 1999, an automatic reactor trip (RT) occurred after Instrumentation and Control personnel completed installation of a temporary modification (TM). The TM was installed downstream of the normally closed low pressure side blowdown line isolation valve for Reactor Coolant System (RCS) flow transmitter FT-436. FT-436 is one of three differential pressure transmitters on RCS loop 3. To implement the TM the transmitter's low pressure supply side isolation valve (ISV) was closed (and the bistable placed in trip) then opened to return the transmitter to service. Upon opening, a RT occurred and auxiliary feedwater actuated. The plant was stabilized in the hot shutdown condition. The direct cause of the RT was due to satisfying the reactor protection system (RPS) logic of two out of three RCS low flow in a single loop. The RPS logic was satisfied because the bistable for FT-436 was in the tripped condition and the bistable for FT-435 actuated due to a pressure oscillation in the common high pressure side transmitter sensing line. The oscillation, caused by unisolating FT-436, was sufficient to actuate (trip) the low flow bistable for FT-435. The root cause was inadequate planning and change management for the TM, due to a failure to assess the risks, consequences and system interactions associated with the TM. A contributing cause was the sensitivity of FT-435. Corrective actions included an inspection of like valves to confirm leak tightness, installation of a TM on FT-435, evaluation of the TM procedure for adequacy in addressing plant mode and system interactions, and a procedure revision to prevent work on RCS FTs above P-8. FT-435 and FT blowdown valves not previously replaced will be replaced, and an Operating Experience report was issued. The event had no effect on public health and safety.

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Note: The Energy Industry identification system Codes are identified within the brackets {}

DESCRIPTION OF EVENT

On March 9, 1999, at approximately 1423 hours, with reactor power at approximately 100 percent, an automatic reactor trip (RT) occurred. The RT occurred during steady state power operation as a result of satisfying the two out of three logic for reactor coolant system (RCS) {AB} low flow in a single loop. With reactor power above the permissive P-8 {IEL} setpoint (approximately 48% power) a RT is generated by the reactor protection system {JC} when a low flow trip signal is generated in any single RCS loop.

On March 8, Operations observed that the RCS loop 3 flow for transmitter FT-436 {FT} started to trend up, which is indicative of leakage on its normally closed low pressure side blowdown isolation valve (RC-FT-436-BDVL) {ISV}. The leaking valve causes an increase in differential pressure sensed by flow transmitter (FT-436) resulting in a false high flow indication on its associated Control Room (CR) {NA} indicator (FI-436) {FI}. Each RCS loop has three (3) flow transmitters, with a loop low flow RT logic that requires two out of three bistables to be tripped in any loop. The three transmitters have a common high pressure side supply line with isolation valves {ISV}, and individual low pressure side supply lines, each with isolation valves {ISV}. Each transmitter also has low pressure and high pressure drain lines (stainless steel tubing) {TBG} with isolation valves.

Management decided to correct the inaccurate RCS flow indication by installing a temporary modification (TM) to prevent leakage from the low pressure side blowdown isolation valve RC-FT-436-BDVL. The TM would cut and remove a portion of the tubing which connects the blowdown valve (RC-FT-436-BDVL) to the containment {NH} trench common header tee. A cap would be installed on the tubing from the header tee and a plug installed on the downstream side of the blowdown valve. A Work Request (WR) was prepared to install the TM. On March 9, a containment entry was made to install the TM. Prior to installing the TM, Instrumentation and Control (I&C) personnel performed a temperature check of the line downstream of the low pressure side blowdown isolation valve. The measured temperature was approximately 175 degrees F and increasing, while the other valves were found to be at ambient temperature. This confirmed that the low pressure side blowdown isolation valve was leaking. Attempts to close the valve further were unsuccessful. I&C personnel concluded the increasing temperature would exceed the WR limit (180 degrees F or less). Consequently, in accordance with the WR step text, I&C initiated actions to close the low pressure side transmitter supply line isolation valve.

On March 9, at approximately 1348 hours, Control Room (CR) Operators entered the Technical Specification (TS) Limiting Condition for Operation (LCO) Action statement for FT-436 after being notified that the low pressure side supply valve was to be closed. Subsequently I&C personnel closed the low pressure side transmitter isolation valve to isolate the leakage and install the TM.

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At approximately 1415 hours, Operations placed the bistable for transmitter FT-436 in the tripped condition per off-normal operating procedure (ONOP)-RPC-1. Implementation of the TM was initiated. At approximately 1421 hours, after TM installation, I&C personnel slowly opened the low pressure side transmitter isolation valve to return the transmitter to service. During valve opening a RT occurred.

CR Operators observed the rod bottom lights and Reactor Trip First Out Annunciator, "Loss of Flow Single Loop." Operators then entered Emergency Operating Procedure (EOP) E-0, "Reactor Trip or Safety Injection," initiated event recovery, stabilized the plant and transitioned to the hot shutdown condition. Plant protective equipment operated as expected in response to the event; all RT (JC) breakers {BKR} opened, all control rods {AA} fully inserted, main feedwater isolated {SJ}, and auxiliary feedwater {BA} automatically started. No safety injection actuation {JE} occurred, nor was one required. RCS pressure remained below the setpoint for pressurizer PORV and code safety valve actuation. Offsite power {EB} remained available during the event. At approximately 1617 hours, the bistable for FT-436 was returned to normal. At 1622 hours, ONOP-RPC-1 was exited and FT-436 returned to operable at 1626 hours.

Equipment that failed to perform properly following the trip included the pressurizer pressure master controller {JD} which was slow to respond to the event and allowed RCS pressure to rise to approximately 2290 psig until taken to manual. Additional equipment that failed to perform properly included the 31 Auxiliary Boiler Feedwater Pump (ABFP) which was observed to have a steaming inboard packing, the 32 Main Boiler Feedwater {SJ} Pump high pressure steam isolation valve {ISV} which had dual indication in the CR, Main Steam {SB} Reheat valves MS-MOV-6-3 and MS-MOV-6-4 which had dual indication in the CR, and the 34C Extraction Steam {SE} Feedwater heater {HX} and the 33B MSR {SB} non-return check valves {V} which didn't indicate closed in the CR. Required safe shutdown equipment performed its function.

An investigation into the cause of the trip was initiated and a post transient evaluation performed. A review of plant recorders and event data showed that a pressure oscillation occurred in the other two channels of RCS loop flow, channel "A" (FT-434) and channel "B" (FT-435) while unisolating transmitter FT-436. The pressure oscillation exceeded the low flow setpoint for channel "B" (FT-435) actuating its associated trip bistable, resulting in satisfying the two out of three RT logic which initiated a RT. It was known that disturbances on the high side instrument lines can create a trip signal since this line is common to all three transmitters in the loop. The risk of installing the TM at power (i.e., above P-8) had been discussed prior to the work.

The RCS loop low flow signals are interlocked by permissives P-7 and P-8. Above permissive P-7 (approximately 10% power), RT will occur for a loss of flow in any two loops. Above permissive P-8 (approximately 48% power), RT will occur for a loss of flow in any single loop. The TM did not contain any actions to close the low pressure supply side transmitter isolation valve, but this action was added to the WR as a contingency to address fluid release and temperature concerns when the blowdown line was cut and capped.

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At 1659 hours, a four hour non-emergency notification (Incident Log No. 35452) was made to the NRC for a reactor protection system (RPS) actuation.

An extent of condition assessment was performed that included a sampling of other similar transmitter isolation valves with similar configurations. The sample lot of valves inspected and checked included reactor coolant flow transmitter and steam generator level transmitter isolation valves. The results of the inspection showed that all low side and high side isolation valves of the sample to be tight with no movement, indicating fully seated isolation valves. With stable RCS flow indications and the results of the sample inspection, no other transmitter isolation valves are considered to be leaking.

CAUSE OF EVENT

The direct cause of the RT was the opening of the RT breakers because the RPS logic for two out of three channels of RCS low flow in a single loop was satisfied. The RPS logic was satisfied because the bistable for transmitter FT-436 was in the tripped condition and the bistable for transmitter FT-435 actuated due to a pressure oscillation in the common high pressure side transmitter sensing line. The oscillation, caused by unisolating FT-436, was sufficient to actuate (trip) the low flow bistable of FT-435. With channel "C" in the tripped condition and power above permissive P-8, the two out of three logic for low RCS flow in a single loop was made up for initiating a RT.

The root cause was inadequate planning and change management for the TM. The inadequacy resulted from internal behavioral factors of, the mindset and overconfidence from successful past experiences. The major difference with the implementation of this TM and previous TM's was the performance of a step in the WR that allowed the closure of the FT low pressure supply side isolation valve. Overconfidence led to less questioning of the activity and a mindset where only the risks associated with the high pressure side were evaluated. These behavioral factors resulted in a failure to adequately assess the risks, consequences and system interactions associated with the TM. The dynamic effects of unisolating the transmitters' low pressure supply side isolation valve was not thoroughly understood. A contributing cause was the sensitivity of FT-435. Troubleshooting FT-435 found that its output is noisy relative to the other transmitters which is believed to be a result of a more sensitive transmitter diaphragm.

CORRECTIVE ACTIONS

The following corrective actions have been or will be performed under the Authority's corrective action program to address the cause of this event.

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- A sample of transmitter blowdown isolation valves were inspected and confirmed to be closed. A similar TM was installed for the same valve on FT-435 because of past leakage. No other transmitter isolation valves were considered to be leaking and RCS flow indications were confirmed to be stable.

- A training session was conducted for appropriate I&C department personnel on the lessons learned from this event.

- A root cause analysis was performed.

- The temporary modification (TM) procedure (AP-13) was evaluated for adequacy in addressing plant mode and system interactions. The procedure was determined to be adequate. The procedure requires assessment of the affects of the TM on a flow path required for plant mode, RT signal, and system interaction.

- The blowdown isolation valves for RCS flow transmitter's not previously replaced are scheduled to be replaced in the next refueling outage (RO 10).

- FT-435 is scheduled to be replaced in the next refueling outage (RO 10).

- The installation of TMs at power for RCS flow transmitters will be limited until the procedure that provides risk assessment is revised to prevent work on RCS flow transmitters above the P-8 setpoint (approximately 48% reactor power). The procedure is scheduled to be revised by the end of July 1999.

- An Operating Experience (OE) report was prepared and issued on the lessons learned. The Authority's industry operating experience process will distribute the OE to the appropriate personnel for review and assessment.

ANALYSIS OF EVENT

The event is reportable under 10 CFR 50.73 (a) (2) (iv). The licensee shall report any event or condition that resulted in a manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS).

This event meets the reporting criteria because an automatic RT occurred as a result of satisfying the reactor protection system logic for a reactor trip. Two out of three RCS low flow in a single RCS loop above the P-8 setpoint generated a trip signal. Auxiliary feedwater automatically started in response to the RT. In response to the event, Operations notified the NRC of an RPS actuation in accordance with 10 CFR 50.72 (b)(2)(ii), (ENS Log No. 35452). A review of the past two years of Licensee Event Reports (LER) for events that involved RTs identified LER 97-025, LER 97-024, and LER 97-005. There were no previous events identified that were due to the manipulation of instrument valves which caused a RT.

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LER 97-024 was similar because the RT was a result of an inadequate turbine trip test; LER 97-025 and LER 97-005 were due to equipment failures. The trip test was revised but failed to include a step for resetting the generator lockout relays. The cause was poor work practices and inadequate work organization and planning due to overconfidence and an inappropriate technical review. The corrective actions for LER 97-024 would not have prevented this event because this event was a TM not a test, had different systems/personnel, and an unknown technical issue.

SAFETY SIGNIFICANCE

This event had no effect on the health and safety of the public. There were no actual safety consequences for the event because there were no RCS loop flow conditions that required the low flow transmitter to actuate. RCS pressure, level, and flow were normal and the reactor tripped as a result of a pressure oscillation in the RCS flow transmitter sensing lines. Following the RT, the plant was maintained stable in the hot shutdown condition.

There were no potential safety consequences of the event under postulated accident conditions. The protection systems are designed to fail into a safe state. Each RT circuit was designed so that a trip occurs when the circuit is de-energized. RT is a fail safe condition. The primary function of the RCS low flow protection is to protect the core from exceeding DNB limits during loss of reactor coolant flow by tripping the reactor. This capability was maintained for this event. Reactor coolant flow would be lost following loss of power to one or more Reactor Coolant Pumps (RCP), a station blackout or similar events, RCP locked rotor or shaft break.

A loss of coolant flow condition is sensed by four methods; measured low flow in the reactor coolant piping (two out of three), detecting RCP breaker open position (two out four), sensing RCP bus undervoltage conditions (two out four), and sensing RCP bus underfrequency conditions (two out four). In this event, a pressure oscillation in RCS loop 3 flow transmitter sensing line exceeded the transmitter low flow limit and its associated bistable tripped as designed. There was another RCS loop 3 flow transmitter available (FT-434) that could have initiated RT on sensed low RCS flow.

Equipment that failed to perform properly as a result of the event had no safety significant effect on the event and no effect on the health and safety of the public. The slowness of the pressurizer pressure master controller to respond to the RT was recognized by the operators who are trained to place the controller in manual. The reactor protection system has a high pressurizer pressure trip to protect against RCS over pressure. The 31 Auxiliary Boiler Feedwater Pump (ABFP), which was observed to have a steaming inboard packing, was corrected by operators by loosening the packing. The Auxiliary Feedwater System has a redundant motor driven pump (33 ABFP) and a turbine driven pump (32 ABFP) to perform the safety function. The 32 Main Boiler Feedwater Pump high pressure isolation valve and the Main Steam Reheat valves MS-MOV-6-3 and MS-MOV-6-4 dual indication in the CR was determined to be caused by a problem with limit switches and the valves were verified to be closed. The 34C Extraction Steam Feedwater heater non-return check valve and the 33B MSR non-return check valve failure to indicate closed in the CR were due to problems with limit switches and the valves were verified to be closed.