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October 5, 1995
JAFP-95-0440

United States Nuclear Regulatory Commission
Document Control Desk
Mail Station P1-137
Washington, D.C. 20555

SUBJECT: DOCKET NO. 50-333
LICENSEE EVENT REPORT: 95-013-00

Loss of Feedwater Flow Transient Due to Personnel Error

Dear Sir:

This report is submitted in accordance with 10 CFR 50.73(a)(2)(iv). That is, a condition which resulted in automatic initiation of the Reactor Protection System.

There are no commitments contained in this report.

Questions concerning this report may be addressed to Mr. Robert Bruns at (315) 349-6575.

Very truly yours,

HARRY P. SALMON, JR.

HPS:RWB:las
Enclosure

cc: USNRC, Region I
USNRC Resident Inspector
INPO Records Center

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NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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

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TITLE (4)

Loss of Feedwater Flow Transient Due to Personnel Error

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
09	05	95	95	-- 013 --	00	10	05	95		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)
Mr. Robert Bruns, Senior Licensing Engineer	(315) 349-6575

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)

On 9/05/95 at 1303 hours, with the plant operating at 100 percent power, a feedwater control system fuse was mistakenly removed due to personnel error. Removal of the fuse caused a rapid reduction in feedwater flow and a reactor scram due to low reactor water level. Isolation of the containment, initiation of high pressure coolant injection and reactor core isolation cooling systems, and initiation of alternate rod insertion and recirculation pump trip signals occurred in response to lowering reactor water level. The safety significance of this event was minimal because the plant responded as designed to lowering reactor water level and the event is bounded by the previously analyzed loss of feedwater flow transient. A post transient evaluation was performed in order to determine the cause of the event, evaluate plant and personnel response, and provide corrective actions. All corrective actions required for continued plant operation were completed prior to restart.

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

EIIS Codes are in []

Event Description

On September 5, 1995, the plant was operating at 100 percent power with reactor pressure vessel (RPV) [AD] water level being automatically maintained at the normal level of approximately 202 inches above the top of active fuel (TAF). At 1303 hours while operators were hanging protective tags in panel 09-5 to allow replacement of a Primary Containment Isolation System (PCIS) [JM] reset switch, Feedwater Control System [JB] fuse 6A-F8 was removed in error. Removal of fuse 6A-F8 caused a loss of power to feedwater control manual/automatic transfer stations and an exponential decay in the speed demand signals to Reactor Feedwater Pump Turbines (RFPT) [SJ] A and B. When the Feedwater Control System sensed a loss of the speed demand signal, lock-up signals were initiated to lock RFPT speed at the last demanded value. Since RFPT speed demand signals decayed before the lock-up signals were initiated, the resultant feedwater flowrate was substantially less than required to maintain RPV water level. This caused RPV water level to rapidly lower. The Nuclear Control Operator (NCO) unsuccessfully attempted to take manual control of the feedwater pumps to maintain level. He then attempted to shut down the reactor by inserting a manual scram signal; however, an automatic reactor scram occurred first due to low RPV water level (177 inches above TAF). In addition to an automatic reactor scram, the following automatic actions occurred at low RPV water level:

- PCIS Group II Isolation
- Standby Gas Treatment System (SGT) [BH] Initiation
- Reactor Water Clean-Up System (RWCU) [CE] Isolation
- Reactor Building [NG] Isolation

RPV water level continued to lower resulting in initiation of the following at low-low RPV water level (126.5 inches above TAF):

- Alternate Rod Insertion
- High Pressure Coolant Injection (HPCI) [BJ]
- Reactor Core Isolation Cooling (RCIC) [BN]
- Reactor Water Recirculation (RWR) [AD] Pump Trip

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

Plant operators carried out the actions of Abnormal Operating Procedure AOP-1, Reactor Scram, and Emergency Operating Procedure EOP-2, RPV Control. Operators examined the possibility of restarting RWR pumps but recognized that pump restart was prohibited due to differential temperature limitations between RWR loops and the RPV. Operators throttled Control Rod Drive (CRD) [AA] system flowrate to minimize the accumulation of relatively cool water in the RPV bottom head. Operators also recognized the need to establish flow from the RPV bottom head drain line and initiated actions to restore the RWCU System to service; however, replacement of the PCIS switch was required in order to reset the RWCU isolation. Despite these efforts, thermal stratification of the RPV occurred due to the flow of CRD water into the RPV bottom head and the lack of forced circulation or bottom drain line flow. During the subsequent forced cooldown, operators recognized that RPV bottom head temperature and pressure were approaching the limits of the enveloping Technical Specification pressure-temperature curve. Operators stopped forced cooldown and commenced raising RPV water level to increase natural circulation flow. Forced cooldown was stopped before raising RPV water level to avoid multiple evolutions which could affect cooldown rate. Increased natural circulation was desired to provide greater margin to Technical Specification limits by reducing bottom head cooldown rate. These efforts were unsuccessful in that bottom head temperature and pressure exceeded the limits of the enveloping Technical Specification curve. The specific RPV bottom head pressure-temperature limits were not exceeded. Operators restored pressure and temperature to within the limits of Technical Specifications using a combination of forced cooldown to lower RPV pressure and RWCU blowdown flow to warm the bottom head.

The sequence of events leading up to and following the scram is presented below. All RPV water levels are referenced to inches above TAF:

12:59:00 - Plant is in normal operation at 100 percent rated power. RPV water level is in the normal range (202 inches). Operations personnel commence hanging protective tags to allow replacement of the PCIS reset switch.

13:02:55 - Operators remove fuse 6A-F8 in error. The correct fuse was 16A-F8. Feedwater flow begins to lower.

13:03: (Approx) - NCO unsuccessfully attempts manual control of feedwater pumps.

13:03:10 - Automatic reactor scram occurs due to low RPV water level (177 inches).

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13:03:11 - PCIS Group II isolation, Reactor Building isolation, SGT initiation, and RWCU isolation occur due to low RPV water level (177 inches).

13:03:13 - RWR pump trip, alternate rod insertion, HPCI and RCIC initiation occur due to low-low RPV water level (126.5 inches).

- NCO inserts a manual reactor scram.

13:03:24 - Lowest RPV water level is reached (123.5 inches).

13:03:30 - RFPT A speed lockup occurs. RFPT B speed continues to lower. Both feedwater pumps are injecting into the RPV at a total flowrate of approximately 35 percent.

13:04:06 - RPV water level is restored to 180 inches.

13:04:43 - HPCI trip and RCIC shutdown occur due to high RPV water level (222.5 inches).

13:04:45 - Main Turbine [TA] trips due to high RPV water level (222.5 inches).

13:04:46 - RFPT A and B trip due to high RPV water level (222.5 inches).

13:11:52 - Operators replace fuse 6A-F8. RFPT A and B speed control is restored.

13:15:04 - Operators reset alternate control rod insertion and scram signals.

13:16 - Operators examine possibility of restarting RWR pumps but recognize that pump restart is prohibited due to differential temperature limitations.

13:34 - Operators place RFPT A in service to control RPV water level.

13:52 - Operators notify NRC of the event (ENS phone call).

13:54 to
13:55:09 - Turbine Bypass Valve (TBV) [JI] #1 throttles open to control RPV pressure at 940 psig. Operators observe oscillations in valve position and lower the pressure setpoint to further open the valve and eliminate oscillations. This causes RPV water level to swell.

13:55:09 - RFPT A trips on high RPV water level (222.5 inches).

14:00 - Operators return RFPT A to service.

14:13 - Operators exit EOP-2 with the plant stable in the Hot Shutdown Mode.

15:12 - Operators reset low RPV water level (177 inches) isolation signal and commence preparations for returning RWCU to service.

16:06 - Operators commence forced cooldown.

17:21 - Operators stop forced cooldown and raise RPV water level to promote natural circulation and provide greater margin to RPV pressure-temperature limits.

17:40 - RPV bottom head pressure-temperature exceed enveloping Technical Specification P-T limit curve. Specific pressure-temperature limits for the RPV bottom head are not exceeded.

18:00 - Operators recommence forced cooldown to restore bottom head pressure-temperature to within enveloping Technical Specification limit curve.

19:05 - Operators place RWCU in service in blowdown mode. RPV bottom head metal temperature begins to rise.

19:21 - Operators restore RPV bottom head pressure-temperature to within enveloping Technical Specification limit curve.

Plant cooldown was continued and Cold Shutdown was achieved at 0250 hours on September 6, 1995.

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Cause of Event

The scram was caused by personnel error while hanging protective tags to allow replacement of a PCIS reset switch. Inadequate self-verification by the two licensed operators performing a dual-concurrent verification of the fuse removal resulted in removal of an incorrect fuse. This error was a cognitive error in that the two licensed operators were confronted with an apparent discrepancy between the fuse labelling and the protective tag labelling and incorrectly reconciled the difference by concluding that the fuse label was wrong or faded by aging. The licensed operators also failed to adhere to their responsibilities as specified in the administrative procedure for protective tagging. The procedure dictates that operators are responsible for notifying the individual who authorized issuance of protective tags if any label discrepancies are discovered. This person was not notified of the apparent discrepancy prior to removing the fuse.

Analysis of Event

This event is bounded by the previously analyzed loss of feedwater flow transient as described in the FitzPatrick Updated Final Safety Analysis Report (FSAR). The plant responded as designed to a loss of feedwater flow. Adequate core cooling was maintained and RPV pressure was controlled throughout the event. Additionally, no safety relief valves [SB] were actuated and there was no challenge to the primary containment [NH]. Therefore, the safety significance of this event was minimal.

The fact that RPV bottom head pressure and temperature exceeded the limits of the enveloping Technical Specification curve was not significant because the specific bottom head pressure and temperature limits were not exceeded. The Technical Specification pressure-temperature (P-T) limit curve is a composite curve established by superimposing the limits of the most restrictive portions of the RPV. The specific P-T limits for the RPV bottom head are less restrictive than the Technical Specification P-T limits.

In addition to P-T limit considerations during this event, there was a thermal transient which produced heatup and cooldown stresses on the RPV bottom head. For analytical purposes, a cooldown rate of 200 degrees F per hour was selected for the bottom head. This rate was greater than the actual change in RPV bottom drain line temperature during the first hour following the scram. A 200 degree F per hour cooldown rate for the RPV bottom head is bounded by bottom head specific limits and previously analyzed transients; therefore, the bottom head cooldown rate was not safety significant.

The observed bottom head heatup rates during the transient were not safety significant because the rates were less severe than the analyzed cooldown rate. There were two instances of higher than normal heatup rates. The first occurred while restoring RPV bottom head pressure and temperature to within Technical Specification Limits. During this evolution, bottom head metal temperature rose approximately 120 degrees F in a one-hour period. The second instance occurred when shutdown cooling [BO] was initiated resulting in an approximate 80 degree F step change in RPV bottom head drain line temperature, although the normal cooldown rate limit of 100 degrees F in any one-hour period was not exceeded. Both of these instances were less severe than cooldown based on the magnitude of the change in a one-hour period and the fact that heatup is less restrictive than cooldown.

The stress cycle associated with cooldown and subsequent heatup of the bottom head was not safety significant because it is bounded by the loss of feedwater transient analysis.

This event requires a report under 10 CFR 50.73(a)(2)(iv). That is, an event that resulted in automatic actuation of the Reactor Protection System [JC].

Corrective Actions

1. An engineering evaluation of the RPV thermal transient was performed by Authority and General Electric personnel. The evaluation concluded that the reactor coolant temperature and pressure, as well as heatup and cooldown rates experienced during the transient are bounded by RPV bottom head specific pressure-temperature limit curves and by the previously analyzed loss of feedwater transient. Based on the evaluation, the thermal transient was within the design basis of the RPV. No fatigue criteria regarding the RPV were violated and no safety concerns exist following this event. Therefore, return to full power operation was determined to be acceptable. (COMPLETE)

2. Abnormal Operating Procedure AOP-1, Reactor Scram, was revised to start a plant cooldown without delay, in the event that recirculation pumps are tripped and cannot be started. Cooldown will lower RPV pressure and thus provide additional margin to pressure-temperature limits. (COMPLETE)

3. The two licensed operators involved in the erroneous removal of fuse 6A-F8 were removed from watchstanding duties pending completion of an investigation of the event. Prior to resumption of watchstanding duties, the licensed operators were counseled on their failure to meet plant standards and management expectations in the areas of self-verification, questioning attitude, and procedural responsibilities. (COMPLETE)

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4. All operations personnel were briefed on the details of this event and the lessons learned. Plant standards and management expectations were reinforced in the areas of self-verification and questioning attitude. All Operations personnel were briefed before assuming watchstanding duties during plant restart. (COMPLETE)

5. Administrative Procedure AP-12.01, Equipment and Personnel Protective Tagging, was revised to require that protective tags specify both the system fuse number and panel fuse number, if applicable. (COMPLETE)

6. An operator aid which shows fuse numbers and locations was installed inside panel 09-5. (COMPLETE)

7. Operating Procedure OP-28, Reactor Water Clean-Up System, was revised to add a procedure for rapid restoration of blowdown flow in order to minimize thermal stratification following a low RPV water level isolation. (COMPLETE)

8. The pressure control loop for turbine bypass valves was tested in an attempt to recreate the valve position oscillations that were observed by plant operators during the scram. The data obtained during testing did not reveal any malfunction of the instrument loop. A review of plant transient data indicated that RPV pressure remained stable throughout the transient; therefore any oscillations in valve position did not effect RPV pressure. A General Electric representative reviewed the data, witnessed control loop testing, and concurred that no adverse conditions existed. (COMPLETE)

9. The feedwater control loss of signal lockup feature was tested to verify proper operation. Testing revealed that the loss of signal lockup feature functioned as designed during the event. The function of the lockup feature is to lock RFPT speed at the last demanded value upon a loss of signal. The nature of the signal loss during the transient was an exponential decay rather than an abrupt loss; this caused RFPT speed to lower before the lockup feature actuated. (COMPLETE)

10. Simulator response will be modified to more accurately reflect recirculation loop cooldown following pump trip. (Planned Completion Date: November 15, 1995)

11. A plant computer display will be developed to plot pressure-temperature limit curves and provide alarms when curves are approached or exceeded. (Planned Completion Date: November 15, 1995)

12. Operator aids will be developed for fuse identification in other panels containing critical circuits. (Planned Completion Date: December 1, 1995)

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13. Design changes will be evaluated to eliminate the failure of fuse 6A-F8 from causing a reactor scram. (Planned Completion Date: March 15, 1996)

14. The logic circuits which initiate RWR pump trip on low RPV water level will be evaluated to determine if logic changes are appropriate. (Planned Completion Date: March 15, 1996)

15. Technical Specification changes regarding restart requirements for recirculation pumps will be evaluated. (Planned Completion Date: March 15, 1996)

16. Technical Specification changes regarding RPV bottom head heatup/cooldown rates and pressure-temperature limits will be evaluated. (Planned Completion Date: March 15, 1996)

Previous Similar Events: LER-90-009 describes a similar event in which a loose part in the Feedwater level control circuitry resulted in a false low reactor water level signal and unit trip.

LER-93-009-02 describes a similar event in which a loose electrical connection caused a loss of feedwater flow and reactor scram.