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November 17, 2008
JAFP-08-0119

United States Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Subject: **Docket No. 50-333**
License No. DPR-59

LICENSEE EVENT REPORT: LER-2008-001-00
Loss of Shutdown Cooling Resulting From Invalid PCIS Actuation Signal

Dear Sir or Madam:

This report is submitted in accordance with 10 CFR 50.73(a)(2)(iv)(A), "Any event or condition that resulted in manual or automatic actuation of... (B)(2) general containment isolation signals affecting containment isolation valves in more than one system" and 10 CFR 50.73(a)(2)(v)(B), "Any event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to... (B) remove residual heat."

There are no commitments contained in this report.

Questions concerning this report may be addressed to Mr. Gene Dorman, Acting Licensing Manager, at (315) 349-6810.

Very truly yours,

Pete Dietrich
Site Vice President

PD:jm
Enclosure

cc: USNRC, Region 1
USNRC, Project Directorate
USNRC Resident Inspector
INPO Records Center

IE22
NRR

NRC FORM 366 (9-2007) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 08/31/2010

Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@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 an 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.

LICENSEE EVENT REPORT (LER)

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4. TITLE

Loss of Shutdown Cooling Resulting from Invalid Primary Containment Isolation System Actuation Signal

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									N/A	05000
09	16	2008	2008	- 001 -	00	11	17	2008	FACILITY NAME: N/A	DOCKET NUMBER: 05000

9. OPERATING MODE

5

10. POWER LEVEL

000

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

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☑ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☑ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

CONTACT NAME	TELEPHONE NUMBER *(Include Area Code)*
Mr. Gene Dorman, Acting Licensing Manager	(315) 349-6810

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

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED

○ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* ◉ NO

15. EXPECTED SUBMISSION DATE

MONTH	DAY	YEAR

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On September 16, 2008 at 0734, with the plant shutdown and in the refueling mode (Mode 5), a fuse was removed during an equipment tag-out evolution resulting in the closure of two Primary Containment Isolation System (PCIS) valves that resulted in a loss of Residual Heat Removal (RHR) Shutdown Cooling (SDC). At the time of the event cavity flood-up was in progress. In response to this event, the Reactor Protection System (RPS) response and logic circuits functioned as designed. The appropriate Technical Specification LCO Actions were entered and shutdown cooling was restored at 0827. The event was reported to the NRC Operations Center as EN# 44492 via the Emergency Notification System pursuant to 10 CFR 50.72(b)(3)(v) for loss of SDC (8-hour report).

The cause of the event was ineffective implementation of the outage risk assessment procedure. There were no safety system functional failures. There were no nuclear, radiological or industrial safety consequences associated with this event. All systems performed as designed and there were no component or system failures. Loss of the common SDC suction line impacts both systems/trains, however, the system remained available through manual realignment of the suction isolation valve. Therefore, barriers providing safety to the public were not compromised and the safety significance of this event is considered low.

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

BACKGROUND

The Primary Containment Isolation System (PCIS) [EIIS=JM] initiates automatic isolation of appropriate process lines which penetrate the primary containment whenever monitored variables exceed preselected operational limits.

The Residual Heat Removal System (RHR) [EIIS=BO] removes decay heat from the reactor core when the plant is in the refueling mode (Mode 5). RHR Shutdown Cooling (SDC) Inboard Isolation Valve, 10MOV-18 [EIIS=ISV], isolates the common shutdown cooling suction line to all RHR pumps [EIIS=P]. With the "B" RPS system de-energized removal of fuse 16A-F15 [EIIS=FU] completely de-energizes power to PCIS Logic Relay 16A-K29 [EIIS=86] resulting in closure of 10MOV-18.

EVENT DESCRIPTION

On September 16, 2008 at 0734, while the James A. FitzPatrick Nuclear Power Plant (JAF) was shutdown, in Refueling Mode (Mode 5), with cavity flood-up in progress, a fuse (16A-F15) was removed during an equipment tag-out evolution that resulted in the closure of two PCIS valves that resulted in a loss of SDC to the reactor core. The tag-out was being performed on a portion of the Reactor Protection System (RPS) [EIIS=JC], PCIS circuitry, to support relay replacement activities. When removing fuse 16A-F15 as directed by the tag-out, isolation logic for the RHR SDC Inboard Isolation Valve, 10 MOV-18, and Recirculation Loop Inboard Sample Isolation Valve 02-2AOV-39 [EIIS=ISV] actuated. Closure of the SDC Inboard Isolation valve isolates the common shutdown cooling suction line to all RHR Pumps, thereby isolating SDC to the reactor core. At the time of the isolation, the time to boil was greater than 5.5 hours.

Immediately following the event, Operators restored power to the RPS, restored SDC and suspended further work in the system pending the results of an investigation into the cause of the event. Additional controls were established that included a review of all electrical jumper / protective tagging interactions specified in the remaining work activities prior to installation.

The event was reported to the NRC Operations Center as EN# 44492 via the Emergency Notification System pursuant to 10 CFR 50.72(b)(3)(v) for loss of SDC (8-hour report). The event also requires written NRC notification within sixty (60) days in accordance with 10CFR50.73(a)(2)(iv), "Any event or condition that resulted in manual or automatic actuation of ...(B)(2) General containment isolation signals affecting containment isolation valves in more than one system ... and, 10CFR50.73(a)(2)(v), "Any even or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to, ...(B) Remove residual heat."

EVENT ANALYSIS

The RPS system response and logic circuits functioned as designed. The event resulted in isolation of two primary containment isolation valves and loss of shutdown cooling. Both systems were quickly restored and had minimal impact on overall plant safety.

CAUSE OF EVENT

The cause of the event was ineffective implementation of the outage risk assessment procedure. The 'B' RPS tag-out was intended to be implemented while RHR Shutdown Cooling was not required to be in service and the DHR system was in service. When the work task was relocated from the SDC outage to a period in which SDC would be in service the outage risk assessment team failed to identify the potential impact of hanging the "B" RPS tag-out on plant conditions.

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

EXTENT OF CONDITION

The inadvertent actuation of a safety system (PCIS Group II Isolation) while performing a tag-out evolution could occur in any work task associated with a plant safety system.

FAILED COMPONENT IDENTIFICATION

There were no component failures as a result of this event. The event was caused by human error due to ineffective implementation of the Outage Risk Assessment Procedure.

CORRECTIVE ACTIONS

Immediate Corrective Actions:

1. Restored power to the affected portion of the RPS.
2. Restored SDC within 53 minutes.
3. Suspended further work in the RPS system pending the results of an investigation into the cause of the event.
4. Established additional controls to prevent recurrence of this event, including a review of all electrical jumper / protective tagging interactions specified in remaining work activities prior to installation.

Completed Corrective Actions:

1. Performed a Root Cause Analysis.
2. Performed a Human Performance Error Review.
3. Performed a Plant Impact Assessment for all tag-outs that were installed from 09/21/08 to the completion of the refuel outage (R18) that involved pulling fuses and/or lifting leads. Additional emphasis was provided on tag-outs affecting protective logic circuitry.
4. All surveillance tests categorized as "High Risk Activity" that were found to contain additional controls were classified as an Infrequently Performed Tests and Evolutions (IPTE) and controls required by the IPTE governing procedure (EN-OP-116) were implemented as appropriate to provide additional oversight.
5. Provided special consideration when determining risk assessments to address the potential configuration when a half isolation signal is present, similar to having one train of SDC available, with contingencies developed that address single point failures on the other isolation logic circuitry.
6. Addressed risk reviews for emergent work with additional rigor.
7. Evaluated plant procedures to ensure that specific components necessary to protect SDC are identified for administrative controls.

Planned Corrective Actions:

1. Upon completion of the Root Cause investigation, this event will be shared with the industry through the INPO Operating Experience Program.
2. A benchmark will be performed on SDC procedures from Vermont Yankee, River Bend Station, and Nine Mile Point Unit 2 as appropriate, to incorporate best industry practices into JAF site specific procedures.
3. Evaluate for potential enhancements, JAF procedures relating to equipment tag-out and protected equipment administrative controls and performance of risk assessment reviews for plant impact.

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ASSESSMENT OF SAFETY CONSEQUENCES

The event began on September 16, 2008 at 0734 and was terminated within 53 minutes at 0827 after restoration of SDC. During the period that SDC was isolated, the reactor was shutdown in Mode 5 for refueling, cavity flood-up was in progress and the time to boil was greater than 5.5 hours. The reactor water temperature at the onset of the event was 100°F and 108°F upon restoration of SDC. The resulting increase of 8°F did not significantly alter plant conditions.

There were no nuclear, radiological or industrial safety consequences associated with the event. All systems performed as designed and there were no component or system failures. In the event that the RHR pumps could not be restarted, an alternate train of redundant RHR pumps was available for SDC. In the event that the SDC suction isolation valves would not open from the control room, manual re-opening of the valves could have been performed if directed by the shift manager. Therefore, barriers providing safety to the public were not compromised and the safety significance of this event is considered low.

SIMILAR EVENTS

No similar events at JAF have occurred during the past ten (10) years.

REFERENCES

JAF Condition Report CR-JAF-2008-02997, SDC Isolation While Hanging PTR