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W3F1-2010-0036

May 13, 2010

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Subject: Licensee Event Report 2009-005-01
Waterford Steam Electric Station, Unit 3 (Waterford 3)
Docket No. 50-382
License No. NPF-38

Dear Sir or Madam:

Entergy is hereby submitting Licensee Event Report (LER) 2009-005-01 for Waterford Steam Electric Station Unit 3. This report provides the details concerning a manual reactor trip and automatic engineered safety feature actuation of emergency feedwater subsequent to the spurious opening of a moisture separator heater relief valve. The condition is reported herein pursuant to 10CFR50.73(a)(2)(iv)(A).

This is a revision to LER 2009-005-00 which was submitted on December 16, 2009 under Entergy Letter W3F1-2009-0071 (ADAMS Accession Number ML093560080). This revision updates the causal factors and corrective action sections of the report using information gathered subsequent to submittal of the original report. The information clarifies what aspect of the manufacturing process affected the spring metal and how the replacement springs support effective corrective action.

This report contains no new commitments. Please contact William J. Steelman at (504) 739-6685 if you have questions regarding this information.

Sincerely,

William Steelman

WJS/RJP/ssf

Attachment: Licensee Event Report 2009-005-01

IE22
NRR

cc: Mr. Elmo E. Collins, Jr.
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R.K. West, lerevents@inpo.org - INPO Records Center

Attachment

W3F1-2010-0036

Licensee Event Report 2009-005-01

NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION (9-2007)	APPROVED BY OMB NO. 3150-0104 EXPIRES 8/31/2010
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory information 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 Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@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 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.

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4. TITLE

Spurious Moisture Separator Reheater Relief Valve Opening Resulting in a Manual Reactor Trip and an Engineering Safety Feature Actuation (Emergency Feedwater Actuation)

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: NA	DOCKET NUMBER: 05000
10	19	2009	2009	- 005 -	01	05	13	2010	FACILITY NAME: NA	DOCKET NUMBER: 05000

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)			
1	☐ 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)
10. POWER LEVEL	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
100%	☐ 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 Specify in Abstract below or in NRC Form 366A
	☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
Waterford 3 Steam Electric Station, William J Steelman	(504) 739-6685

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
B	SB	RV	Crosby Valves & Gage Co.	Y					

14. SUPPLEMENTAL REPORT EXPECTED	15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES (If yes, complete 15. EXPECTED SUBMISSION DATE) ☒ NO				

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

On October 19, 2009, at approximately 09:44 with the plant operating at 100% power (Mode 1), Waterford 3 manually tripped the reactor due to lowering condenser hotwell level caused by a stuck open moisture separator reheater relief valve. The Plant Protection System (PPS) responded as designed, resulting in an uncomplicated reactor trip. Subsequently, Emergency Feedwater Actuation Signals (EFAS-1 and EFAS-2) were received due to low Steam Generator (SG) levels which is an anticipated response to the reactor trip with the plant at or near full power. The emergency feedwater system did not receive a signal to send water to the steam generators. The plant was then maintained in Mode 3 with both steam generators being fed from the main feedwater system with steam generator levels in the normal operational band for Mode 3. Failure Mode Analysis (FMA) identified a broken pilot valve spring on the stuck open moisture separator heater relief valve. The pilot valve spring was replaced on all six moisture separator heater relief valves prior to unit restart. Adequate water level was maintained in the steam generators to ensure decay heat removal from the Reactor Coolant System (RCS). The event is not considered a safety system functional failure. The event did not compromise the health and safety of the general public.

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NARRATIVE

REPORTABLE OCCURRENCE

On October 19, 2009 at 09:44, Waterford 3 manually tripped the reactor due to lowering condenser hotwell level caused by a stuck open reheat system [SB] moisture separator reheater (MSR) relief valve [RV] (RS-203B). Subsequently, Emergency Feedwater Actuation Signals (EFAS-1 and EFAS-2) were received due to low Steam Generator (SG) levels actuating the Emergency Feedwater (EFW) System [BA]. The condition was reported to the NRC Operations Center within four hours. The event was reportable within four hours under criteria 10CFR50.72(b)(2)(iv)(B) for a manual reactor trip of the plant to preclude receiving an automatic Reactor Protection System (RPS) [JC] trip while the reactor was critical. Additionally, the event was reportable within eight hours under criteria 10CFR50.72(b)(3)(iv) for the automatic actuation of EFAS upon low Steam Generator levels. The manual reactor trip is reportable in writing (Licensee Event Report) within 60 days in accordance with 10CFR50.73(a)(2)(iv)(A) due to the manual actuation of the RPS and due to the automatic actuation of the Emergency Feedwater system.

INITIAL CONDITIONS

Just prior to the initiating events, the plant was operating in Mode 1 at 100% power. There were no procedures being implemented specific to this condition. There were no Technical Specification Limiting Conditions of Operation specific to this condition in effect.

BACKGROUND

Waterford 3 has two MSRs. The purpose of the MSRs are to remove the moisture and to reheat the steam from the high pressure turbine [TRB] exhaust to the low pressure turbine inlet. Each MSR is provided with overpressure protection by three pilot-operated relief valves mounted on top of the shell. These relief valves discharge to atmosphere at approximately 250 psig to prevent damage to the MSR shell. The design pressure of the shell is 265 psig.

With the stuck open moisture separator relief valve (RS-203B) open, the calculated flow rate based on hotwell level loss was 2500 GPM. The maximum makeup rate to the condenser hotwell through the makeup line is approximately 1250 gpm.

EVENT DESCRIPTION

On October 19, 2009 at 09:15, while operating at 100% power moisture separator reheater relief valve, RS-203B, spuriously opened causing reactor power to increase from 100% to 100.27% Rated Thermal Power (RTP). At 09:16, operations promptly reduced main turbine generator load to restore reactor power to less than 100% RTP. At 09:17 hotwell emergency makeup valve (CMU-715) opened. At approximately 09:42, operations commenced a rapid plant shutdown. At 09:44, operations manually tripped the reactor due to lowering condenser hotwell level and entered OP-902-000 (Standard Post Trip Actions). Emergency Feedwater actuation signals EFAS-1 and EFAS-2 automatically initiated due to low Steam Generator levels, an anticipated response to the reactor trip with the plant at or near full power. Steam was isolated to the MSRs. The inventory loss through the open relief valve stopped and hotwell level recovered. The plant was maintained in Mode 3 with both Steam Generators being fed from the non-safety main feedwater [SJ] system with steam generator levels in the normal operational band for Mode 3. The EFAS actuation signals were reset. Following the event, a post trip review was performed.

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NARRATIVE

CAUSAL FACTORS

Results of condition investigations and failure modes analysis conclude that the cause of the condition was a broken pilot valve spring in RS-203B. When the pilot valve spring fractured, the pressure load on the main valve disc was lost and the main valve spring opened RS-203B.

Based on metallurgical analysis performed, the root cause for the fractured spring was that spring material had cracks introduced during the manufacturing process. The springs were made of ASTM A401 spring wire that achieves it strength by being heated to the austenitizing temperature and then quenched. Because this steel is brittle in the as-quenched condition, it is subsequently "tempered."

ASTM A401 wire for springs is drawn to size by pulling through a die. If during the drawing process there is an interruption of the drawing lubricant at the die, the surface of the wire will be heated to the austenizing temperature, and as the wire exits the carbide "nib" or throat of the die, the body of the wire will quench this locally hot surface and will become brittle. When combined with the tensile stresses from pulling the wire through the die, the wire will sometimes develop "cross cracking" on the surface of the wire. The "cross cracking" provides initiation points for propagation of stress cracks.

In the replacement springs, the substitution of A313, type 316 stainless steel wire in place of A401 carbon steel wire eliminates the mechanism that was the root cause of the failure of the original springs that were in these valves.

CORRECTIVE ACTIONS

The carbon steel pilot springs in all six MSR relief valves were replaced with stainless steel springs during refueling outage 16. The wire that was used in the replacement springs was ASTM type 316 stainless steel. This is an austenitic stainless steel that does not harden when heated and quenched. Type 316 stainless steel hardens due to the work hardening from drawing the wire to successively smaller diameters. Because type 316 wire cannot be hardened by heating and quenching, it is immune to embrittlement that leads to "cross cracking."

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NARRATIVE

SAFETY SIGNIFICANCE

The plant remained within safety limits throughout the event. The condition did not prevent the fulfillment of any safety function and did not result in a safety system functional failure (i.e. ability to shut down the reactor and maintain it in a safe shutdown condition, ability to remove residual heat, ability to control the release of radioactive material, and ability to mitigate the consequences of an accident) as defined by 10CFR50.73(a)(2)(v). The only engineered safety feature actuations were emergency feedwater actuation signals EFAS-1 and EFAS-2, which automatically initiated due to low steam generator levels. This is an anticipated response to the reactor trip with the plant at or near full power. Main feedwater maintained SG water level above the setpoint at which EFW control valves open; therefore, no injection of EFW occurred during the event.

Operations manually tripped the reactor which caused a turbine trip and steam to be isolated to the MSRs. There were no structures, systems, or components that were inoperable at the time of the event that contributed to this condition. Since engineered safety features actuated as required, and considering that primary system parameters were maintained within acceptable limits, the safety significance of this event is considered minimal.

SIMILAR EVENTS

A search was performed for other similar reported events at Waterford 3. No similar events were identified.

ADDITIONAL INFORMATION

Energy industry identification system (EIIS) codes are identified in the text within brackets [ ].