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Robert J. Murillo
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W3F1-2009-0071

December 16, 2009

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

Subject: Licensee Event Report 09-005-00
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) 09-005-00 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 report contains no new commitments. Please contact Robert J. Murillo at (504) 739-6715 if you have questions regarding this information.

Sincerely,

For R. Murillo

RJM/WJS

Attachment: Licensee Event Report 09-005-00

IE22
NRR

(w/Attachment)
cc: Mr. Elmo E. Collins, Jr.
Regional Administrator
U. S. Nuclear Regulatory Commission
Region IV
612 E. Lamar Blvd., Suite 400
Arlington, TX 76011-4125

NRC Senior Resident Inspector
Waterford Steam Electric Station Unit 3
P.O. Box 822
Killona, LA 70066-0751

U. S. Nuclear Regulatory Commission
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Office of Environmental Compliance
Surveillance Division
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R.K. West, lerevents@inpo.org - INPO Records Center

Attachment

W3F1-2009-0071

Licensee Event Report 09-005-00

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

LICENSEE EVENT REPORT (LER)

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

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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	05	00	12	16	2009	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, Robert J. Murillo	(504) 739-6715

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 most likely 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.

The pilot valve spring fracture appeared to be due to a high speed (single load) brittle fracture. Based on metallurgical analysis performed, the spring material became brittle while in service. This was caused by a problem during the manufacturing process. The failed spring was in service 3 years prior to failure.

CORRECTIVE ACTIONS

The carbon steel pilot springs in all six MSR relief valves were replaced with stainless steel springs during refueling outage 16.

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.

LICENSEE EVENT REPORT (LER)
CONTINUATION SHEET

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NARRATIVE

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 [ ].