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10 CFR 50.73

W3F1-2019-0021

March 15, 2019

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

Subject: Licensee Event Report (LER) 2019-003-00
Relevant Indications Identified in Reactor Coolant Pump 1A and 2A Suction Drain Nozzle Dissimilar Metal Welds Resulting in the Condition of the Nuclear Power Plant, Including its Principal Safety Barriers, Being Seriously Degraded

Waterford Steam Electric Station, Unit 3 (Waterford 3)
NRC Docket No. 50-382
Renewed Facility Operating License No. NPF-38

The enclosed report is being sent pursuant to 10 CFR 50.73.

There are no regulatory commitments contained in this correspondence.

If you have any questions or require additional information, please contact the Acting Regulatory Assurance Manager, John V. Signorelli, at (504) 739-6032.

Respectfully,

John V. Signorelli

JVS/mmz

Enclosure: Waterford 3 Licensee Event Report 2019-003-00

cc: NRC Region IV Regional Administrator
NRC Senior Resident Inspector – Waterford Steam Electric Station, Unit 3
NRR Project Manager

ENCLOSURE

W3F1-2019-0021

Entergy Operations, Inc.

Waterford 3 Licensee Event Report 2019-003-00

NRC FORM 366 (04-2018) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 03/31/2020

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 Information Services Branch (T-2 F43), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by e-mail to Infocollects.Resource@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)

(See Page 2 for required number of digits/characters for each block)

(See NUREG-1022, R.3 for instruction and guidance for completing this form http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1022/r3/)

1. Facility Name	2. Docket Number	3. Page
Waterford Steam Electric Station, Unit 3	05000382	1 OF 4

4. Title

Relevant Indications Identified in Reactor Coolant Pump 1A and 2A Suction Drain Nozzle Dissimilar Metal Welds Resulting in the Condition of the Nuclear Power Plant, Including its Principal Safety Barriers, Being Seriously Degraded

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
01	17	2019	2019	- 003	- 00	03	15	2019	FACILITY NAME	DOCKET NUMBER

9. Operating Mode	11. This Report is Submitted Pursuant to the Requirements of 10 CFR §: *(Check all that apply)*			
6	☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☒ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
	☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
	☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
	☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
10. Power Level	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
	☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(C)	☐ 73.77(a)(1)
0%	☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(D)	☐ 73.77(a)(2)(i)
	☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(vii)	☐ 73.77(a)(2)(ii)
		☐ 50.73(a)(2)(i)(C)	☐ OTHER (Specify in Abstract below or in NRC Form 366A)	

12. Licensee Contact for this LER

LICENSEE CONTACT	TELEPHONE NUMBER *(Include Area Code)*
John V. Signorelli - Manager, Regulatory Assurance	(504) 739-6032

13. Complete One Line for each Component Failure Described in this Report

Cause	System	Component	Manufacturer	Reportable to ICES	Cause	System	Component	Manufacturer	Reportable to ICES
B	AB	NZL	n/a	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 14 single-spaced typewritten lines)

On January 17 and 18, 2019, relevant indications were detected on the Reactor Coolant System (RCS) Loop Reactor Coolant Pump (RCP) suction drain nozzle to safe-end butt welds 07-009 (RCP 1A) and 11-007 (RCP 2A) during the performance of Phased Array Ultrasonic Examination of Alloy 600 dissimilar metal piping welds during planned inspections which were being conducted during Refueling Outage 22. This resulted in an event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded because they did not meet acceptance criteria referenced in American Society of Mechanical Engineers Section XI, IWB-3514-2 and Code Case N-770-2. The RCS was declared inoperable per Technical Requirements Manual (TRM) requirement 3.4.9, "Structural Integrity."

The cause of this event was Primary Water Stress Corrosion Cracking. Action was taken to restore full component integrity and assure structural integrity of the affected welds by applying dissimilar metal weld full structural weld overlays over the affected components. The RCS was subsequently declared operable on February 18, 2019 and TRM 3.4.9 was exited.

1. FACILITY NAME	2. DOCKET NUMBER	3. LER NUMBER		
		YEAR	SEQUENTIAL NUMBER	REV NO.
Waterford Steam Electric Station, Unit 3	05000382	2019	- 002 -	00

NARRATIVE

EVENT DESCRIPTION

A. Plant Status

When the indications were identified, Waterford 3 was in Mode 6 (Refueling). There were no other structures, systems or components out of service that contributed to this event.

B. Event Chronology and Description of Relevant Indications Detected

On January 17 and 18, 2019, relevant indications were detected on the Reactor Coolant System (RCS) [AB] Loop Reactor Coolant Pump (RCP) [P] suction drain nozzle [NZL] to safe-end butt welds 07-009 (RCP 1A) and 11-007 (RCP 2A) during the performance of Phased Array Ultrasonic Examination of Alloy 600 dissimilar metal piping welds during planned inspections which were being conducted during Refueling Outage 22. The welds contribute to maintaining the RCS pressure boundary. The RCP suction drain nozzles are categorized as American Society of Mechanical Engineers (ASME) Code Class 1 components. The indications are axially oriented and located within the butt weld and weld butter. The axial indications are inside surface connected and are exposed to the reactor coolant.

The indication on weld 07-009 is in the axial direction and measures 0.59” which is 55% through-wall. The width of the indication is 0.40”. The total thickness of the weld is 1.08”. The remaining ligament 0.49” is between the flaw and the outer diameter surface.

The indication on weld 11-007 is in the axial direction and measures 0.58” which is 50% through-wall. The width of the indication is 0.60”. The total thickness of the weld is 1.17”. The remaining ligament 0.59” is between the flaw and the outer diameter surface.

On January 17, 2019, at 1858 CST, following identification of the indication on weld 07-009, the RCS was declared inoperable in accordance with Technical Requirements Manual requirement 3.4.9, "Structural Integrity," due to the weld did not meet acceptance criteria referenced in ASME Section XI, IWB-3514-2 and ASME Code Case N-770-2. Operations entered TRM 3.4.9 action a, which requires that with the structural integrity of any ASME Code Class 1 component(s) not conforming to the ASME requirements, restore the structural integrity of the affected component(s) prior to increasing the RCS temperature more than 70 degrees Fahrenheit above the minimum temperature required by nil ductility temperature (NDT) considerations.

This event is reportable pursuant to 10 CFR 50.73(a)(2)(ii)(A), "Any event or condition that resulted in: (A) The condition of the nuclear power plant, including its principal safety barriers, being seriously degraded," due to welding defects in the RCS that cannot be found acceptable under ASME Section XI.

C. Event Causes

A failure mode analysis and an equipment failure evaluation were performed. This identified that the most likely cause of the indications is Primary Water Stress Corrosion Cracking (PWSCC).

CORRECTIVE ACTIONS

(1) Review for extent of condition per ASME Code Case N-770-2 paragraph 2430(a), which requires that if an examination reveals an unacceptable flaw, then the scope of Inservice Inspections (ISI) must be extended to include additional weld inspections during the current outage. The result of this review was that all welds which would have been included in the code required extent of condition were scoped into the refueling outage and examined using Phased Array Ultrasonic Examination as part of the regularly scheduled outage activities. (complete)

(2) Restore full component integrity and assure structural integrity of the affected welds by obtaining NRC approval of request for relief for and application of dissimilar metal weld full structural weld overlays over the affected components. (complete)

(3) Update the Alloy 600 Project technical report and issue additional actions to revise the Alloy 600 Aging Management program contingency or mitigation plans for any identified deficiencies. (planned)

(4) Assess the likelihood of other cold leg weld failures occurring and, as appropriate, generate action to revise the Alloy 600 Aging Management program contingency or mitigation plans for any identified deficiencies. (planned)

(5) Verify similar susceptible welds do not have unacceptable indications. (complete)

(6) Review Entergy ISI plans to identify welds that may require mitigation. (complete)

(7) Ensure Entergy ISI plans require encoded phased array inspection of all applicable dissimilar metal welds in the future. (planned)

(8) Ensure current ASME Section XI inspection intervals are sufficient to identify indications prior to their becoming unacceptable.

SAFETY EVALUATION

The actual consequence of this event was that the indications were not acceptable under ASME Section XI. There were no other actual consequences to safety of the general public, nuclear safety, industrial safety, and radiological safety for this event.

The potential consequence to safety of the general public, nuclear safety, industrial safety and radiological safety of this event if the indication was through-wall is a loss of the RCS pressure boundary. This would have resulted in a small, non-isolable reactor coolant leak that would have been detected by local radiation detectors and through RCS leakrate surveillance by plant personnel.

PREVIOUS OCCURRENCES

A review of the Waterford 3 corrective action program and previous Licensee Event Reports for the previous 12 years revealed no similar events as described in NUREG-1022 guidance.

Energy Industry Identification System (EIIS) codes and component codes are identified in the text as [XX].