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[10 CFR 50.73]

W3F1-2003-0096

December 18, 2003

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Subject: Waterford 3 SES
Docket No. 50-382
License No. NPF-38
Reporting of Licensee Event Report (LER) 03-003-00

Gentlemen:

Attached is Licensee Event Report (LER) 03-003-00 for Waterford Steam Electric Station Unit 3. This report provides details of the discovery of evidence of RCS pressure boundary leakage. Three indications of leakage were identified during inspections performed during Waterford 3's Refuel 12 Outage. Evidence of leakage was found at a hot leg instrument nozzle and at two heater sleeves on the pressurizer. The conditions are being reported pursuant to 10CFR50.73(a)(2)ii(A) as conditions of the nuclear power plant, including its principal safety barriers, being seriously degraded.

There are no commitments contained in this submittal. If you have any questions, please contact Oscar Pipkins at (504) 739-6707.

Very truly yours,

G. Sen
Manager, Licensing

GS/OPP/cbh
Attachment

IE22

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NRC FORM 366 (7-2001) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004
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: 50 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.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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4. TITLE

RCS Pressure Boundary Leakage Due to Primary Water Stress Corrosion Cracking (PWSCC)

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
Month	Day	Year	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME N/A	DOCKET NUMBER N/A
10	24	2003	2003	003	00	12	18	2003	FACILITY NAME N/A	DOCKET NUMBER N/A

9. OPERATING MODE	5
10 POWER LEVEL	0

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

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

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Oscar P. Pipkins / Senior Licensing Engineer	(504) – 739-6707

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			15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE).	✓	NO				

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

During Waterford 3's Refuel 12 outage, three indications of Reactor Coolant System pressure boundary leakage were identified. The first indication of leakage was identified on October 24, 2003 during an Alloy 600 Program Plan inspection of the small bore nozzles on Hot Leg #2. The leakage was identified on instrument nozzle RC-IPT-0106B. The other two leakage indications were identified on October 26, 2003, while inspecting the pressurizer bottom nozzles and heater sleeves in accordance with the same inspection program plan. One of the two pressurizer leakage indications was identified on heater sleeve C-1, while the other was on heater sleeve C-3. All three leakage indications are believed to be the results of primary water stress corrosion cracking. Nozzle RC-IPT-0106B was repaired using a permanent partial nozzle welded repair. The heater sleeves were repaired using the Mechanical Nozzle Seal Assembly (MNSA-2). The conditions were found during the planned, scheduled inspection program before they could develop into significant leakage. The aggressive inspection and repair program reduces the potential safety significance, via early detection. Plant operations were unaffected and this event did not compromise the health and safety of the public or plant personnel. The conditions are not considered Safety System Functional Failures (SSFF).

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

REPORTABLE OCCURRENCE

On October 24, 2003 and again on October 26, 2003 a total of three indications of Reactor Coolant System leakage was identified during Alloy 600 Program Plan inspections performed during Waterford 3's Refuel 12 outage. The leakage identified constituted Reactor Coolant System pressure boundary leakage, which is evidence of pressure boundary degradation. The conditions were reported on October 25, 2003 and October 26, 2003 to the NRC Operations Center within 8 hrs of their discoveries (EN# 40277 and EN# 40278) in accordance with 10 CFR 50.72(b)(3)(ii)(A) as degraded conditions. The conditions are herein being reported under the 60-day reporting requirements of 10 CFR 50.73(a)(2)(ii)(A) as conditions of the nuclear power plant, including its principal safety barriers, being seriously degraded.

INITIAL CONDITIONS

The plant was in Mode 5 for Refuel 12 with inspections being performed under the Alloy 600 Program Plan.

EVENT DESCRIPTION

On October 23, 2003, during Refuel 12, shortly after entry was allowed in Containment, while the insulation was still on, an inspection was conducted of the Pressurizer and RCS piping for evidence of gross leakage. No evidence of leakage was observed on the exterior surface of the insulation.

On October 24, 2003 at 2300, after insulation had been removed, during inspections of the small bore nozzles on hot leg #2, in accordance with the Alloy 600 Program Plan, evidence of a leak was identified on the instrument nozzle [NZL] connected to instrument RC-IPT-0106B. A localized wetness was observed right around the penetration of the nozzle.

On October 26, 2003 at 1900, evidence of leakage was also identified on pressurizer heater sleeves C-1 and C-3. There was a white substance extruding from a portion of the penetration at heater sleeve C-3 (later confirmed to be boric acid). The metal surface in that area was wetted, however not dripping wet. In addition to that, a thin white film ring at heater sleeve C-1 penetration was also observed. The metal surface at C-1 was not wetted. After the plant cooled down further, the wetted area around C-3 was observed to expand.

CAUSAL FACTORS

The apparent cause for all three cases of identified leakage is believed to be Primary Water Stress Corrosion Cracking (PWSCC) that produced axial flaws which resulted in RCS leakage. This is the same apparent cause as determined during Refuel 9 for the three small bore instrument nozzle failures on the hot legs, and two small bore instrument nozzle failures on the pressurizer. The Alloy 600 Nozzles were determined to be susceptible to PWSCC. The nozzles with evidence of leakage identified during Refuel 12 are of the same material and are susceptible to the same failure mechanism.

CORRECTIVE ACTIONS

The nozzle RC-IPT-0106B condition was corrected by performing a permanent partial nozzle

welded repair.

The condition at heater sleeves C-1 and C-3 was corrected by installing temporary Mechanical Nozzle Seal Assemblies (MNSA-2). Application of MNSA-2 as a temporary repair system at Waterford 3 was authorized via a NRC letter dated July 3, 2003 for use up to two operating cycles.

There are nineteen small bore instrument nozzles on the hot legs and six were previously weld repaired. The leaks are readily discovered by visual inspection. Several repair techniques have been developed to deal with cracked nozzles which include: partial nozzle replacement, weld repair, Mechanical Nozzle Seal Assembly (MNSA and MNSA-2) or a welded plug. The alloy 600 Project will evaluate proactive replacement of the remaining Alloy 600 hot leg nozzles.

There are thirty heater sleeves and seven small bore instrument nozzles on the pressurizer. Four of the instrument nozzles were previously weld repaired and heater sleeve F-4 was repaired with a welded plug during Refuel 10. The leaks are also readily discovered by visual inspection. The repair techniques discussed above for hot leg nozzles are also the choices for pressurizer nozzle repairs. The alloy 600 Project will evaluate proactive replacement or repair of the remaining Alloy 600 pressurizer instrument nozzles and heater sleeves.

SAFETY SIGNIFICANCE

During Waterford 3's RF-12 refueling outage, two Pressurizer heater sleeve penetrations were identified by visual examination as having evidence of RCS leakage. In order to quantify the source of this leakage, a decision was made to disassemble and volumetrically inspect these two penetrations. The NDE methods employed for these inspections were Materials Reliability Program (MRP) / Electric Power Research Institute (EPRI) qualified ultrasonic and eddy current techniques, which were delivered to the penetrations using a Westinghouse rotating probe device similar to that which is used for Reactor Vessel Head penetration inspection. The inspections of these two penetrations revealed that each one contained a single axially oriented PWSCC-type flaw in the wall of the heater sleeve. The location of both flaws was in the region of the weld, which attaches the heater sleeve to the lower head of the Pressurizer. Both flaws began slightly above the upper extent of the weld and extend through to below the lower extent of the weld, providing a leakage path through the wall of the heater sleeve material.

The other location identified during the outage as having evidence of RCS leakage was on the hot leg #2 instrument nozzle connected to instrument RC-IPT-0106B. That leakage area was also consistent with leakage previously observed on small bore instrument nozzle leaks and previously reported at numerous other Pressurized Water Reactors utilizing the Alloy 600 materials. NDE was not performed on this nozzle due to the extensive industry experience with axially oriented flaws in the heat affected zone where the nozzle was welded to the wall of the RCS piping.

Other than light surface pitting, there was no measurable or significant corrosion of the base materials on any of the three leaking penetrations and the surface was cleaned and restored with only light surface buffing or grinding with a wire brush.

In 1993, the CEOG and the other PWR Owners Groups submitted a Safety Evaluation to the NRC on the consequences of cracking of Reactor Vessel Head penetrations (CEDM nozzles). The documents submitted were CEN-607, "Safety Evaluation For and Consequences of Reactor Vessel Head Penetration Alloy 600 ID Initiated Nozzle Cracking" and CEN-614 that addressed OD initiated cracking. The evaluations concluded that there is no new safety significance to the occurrence of an axial through-wall crack. The cracks, if they occur, are expected to be small, stable and axially oriented. All CEOG plants currently have adequate operational procedures and Technical Specifications in place to quickly respond in the event that small leakage occurs. The evaluation also documented that corrosion tests were performed that demonstrated boric acid leaking from a nozzle would not cause significant corrosion on the alloy base material. The safety evaluation included the results of an extensive finite element stress analysis that concluded that the stresses are primarily hoop stresses (circumferential), which result in cracks in the axial direction. This conclusion is supported by industry experience, wherein the indications being detected are axial.

The safety evaluation included CEDM nozzles on the reactor head, however the analysis conservatively bounds any leakage from small bore instrument nozzles. The CEDM nozzles have a larger OD and are subjected to higher stresses due to the internal pressure/surface area relationships.

Based on review of industry correspondence and meetings with owners groups, the NRC developed a NUREG on the issue. NUREG/CR-6245, "Assessment of Pressurized Water Reactor Control Rod Drive Mechanism Nozzle Cracking" concluded "CRDM nozzle cracking is not a short term safety issue." The report stated that it is highly unlikely that growth of an axial crack will lead to nozzle rupture (Page 42).

The CE Owners Group has also been studying Inconel instrument nozzle cracking and the potential safety significance for some time. The conclusion of these studies is that the PWSCC cracks or indications, if they occur, do not significantly challenge the safety of the nuclear plant as follows:

1. The crack is expected to grow axially not circumferentially.

Studies on expected crack growth are documented by CEOG report CE-NPSD-690-P, "Evaluation of Pressurizer Penetrations and Evaluation of corrosion after unidentified leakage develops," issued January 1992. Safety Evaluation CEN-607 dated May 1993, was developed by the CEOG for these types of CEDM nozzle failures. These documents report that crack growth is axial, and therefore, the potential for a catastrophic or circumferential failure is reduced. Slow axial crack growth and slowly increasing leakage allows the plant personnel to detect and react to identified leakage.

2. The crack may grow to a length of 2 inches axially without exhibiting unstable crack growth.

A fracture mechanics analysis by the CEOG CE-NPSD-690-P shows that even with a crack 2 inches long at normal steady state temperatures and pressures, there is a factor of safety of 43 against any additional crack growth due to mechanical means. Fracture mechanic analysis considers plant operational transient conditions. RCS integrity is not affected, as ASME Code requirements do not consider fatigue during accidents. Although the crack may proceed from chemical corrosion

cracking, it will proceed gradually allowing time for detection as addressed in item 3 and 4 below.

3. A potential leak will be detected well before its corresponding crack would grow to an unstable size. It is expected that significant RCS leakage from a crack would be detected by the leakage detection system so that investigations can be made to identify and correct the leakage in a timely manner. The leakage detection system provides extensive and diverse methods for determining a pressure boundary leak. Historically, RCS leakage much less than the Technical Specification allowable leak rate has been detected and compensatory actions taken. Leak detection system and plant practices regarding suspected RCS boundary leaks show that a potential leak in the RCS boundary would be investigated and mitigating actions taken prior to the crack propagating such that mitigating systems are challenged.

The reactor coolant pressure boundary leakage detection system is designed to detect radioactive and non-radioactive leakage inside containment. Equipment is provided to continuously monitor the environmental conditions within containment such that an increase over background level is indicative of higher than normal leakage so that possible leakage from the primary system and components may be identified. Deviations from normal containment environmental conditions provide positive indication to the control room of increases in leakage rates. Methods of detecting RCS leakage consists of the radiation monitoring system, the containment sump inlet flow rate and the RCS water inventory balance. A method of trending noble gases in the containment atmosphere has also been used effectively in the past to identify leaks.

4. Catastrophic failure evaluation:

As discussed above, evidence shows that a catastrophic failure of an Inconel Alloy 600 nozzle is not expected to occur. However, in the event that this type of failure does occur, the consequences will not impact the safety of the plant. Specifically, this type of accident has been previously evaluated in the SAR and a postulated failure of one RCS instrument nozzle is bounded by this analysis. The probability of occurrence of these events are unchanged because with the exception of Palo Verde, all of the industry experience to date indicates that circumferential failure is not expected to occur. In October of 2003, Palo Verde reported circumferential cracking on six pressurizer heater sleeves. The indications were all located in the non-pressure boundary area of the heater sleeve which is above the attachment weld. The hot leg nozzles do not have this nozzle remnant protruding above the weld. As mentioned above, NDE was performed on the two repaired pressurizer heater sleeves and only a single axially oriented indication was found on each penetration in the area of the attachment weld. Circumferential flaws are not expected in the pressure boundary. However, even if they were to occur, based on a heater sleeve safety assessment being developed by the WOG, visible leakage would be apparent for multiple refuel cycles prior to circumferential cracking propagating to the extent of the nozzle ejection becoming possible. Additionally Entergy performs bare metal inspection on these nozzles each outage and would be able to detect such leakage. In the unlikely event that circumferential indications were to form on the repaired small bore RCS hot leg nozzles they would not have safety significance because the partial nozzle weld repair relocates the pressure boundary attachment weld to the outside surface of the RCS piping.

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Small break LOCA events evaluated in the SAR are listed by the size of the break. The smallest break analyzed is listed as 0.01 square foot. The consequences of smaller breaks are deemed to be bounded by the breaks evaluated in the SAR. In the case of RCS nozzles, the break area is smaller than the smallest analyzed small break LOCA area.

The emergency core cooling system (ECCS) is designed to provide sufficient core cooling for all line breaks in the reactor coolant system up to and including the unlikely double ended break in the largest reactor coolant pipe (LOCA). Therefore, failure of a small bore nozzle in the RCS is bounded by the existing safety analysis for the plant.

This event is / is not considered a Safety System Functional Failure (SSFF).

SIMILAR EVENTS

LER 99-002-00 reported the discovery, on February 25, 1999 (during Refuel 9), of evidence of Reactor Coolant System leakage on two Inconel 600 instrument nozzles on the top head of the Pressurizer, One on RCS Hot Leg #1 RTD nozzle, one on RCS hot leg #1 sampling line, and one on RCS hot leg #2 differential pressure instrument nozzle. No evidence of leaks was found on the RCS cold legs or Steam Generators at that time. The apparent cause of the leaks was determined to be axial cracks near the heat-affected zone of the nozzle partial penetration welds resulting from PWSCC. The two leaking nozzles located on the Pressurizer were repaired using welded nozzle replacements in accordance with ASME Section XI. The three leaking nozzles located on hot legs #1 and #2 were temporarily repaired using MNSAs.

LER 00-011-00 reported the discovery, on October 17, 2000 (during Refuel 10), of evidence of Reactor Coolant System leakage at Pressurizer heater sleeve F-4, and at two of the three MNSA clamps that had been temporarily installed during the Refuel 9 outage. The apparent cause of the leakage cases were PWSCC, a MNSA clamp flange not being flat against the pipe, and a MNSA clamp seating itself. The conditions were corrected by plugging the pressurizer heater sleeve, and by removing the MNSA clamps and making permanent weld repairs on the nozzles.

ADDITIONAL INFORMATION

Energy Industry Identification System (EIIS) codes are identified in the text within brackets [ ].