Exelon. Nuclear

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SVP-03-084

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

Quad Cities Nuclear Power Station, Unit 1
Facility Operating License No. DPR-29
NRC Docket No. 50-254

Subject: Licensee Event Report 254/03-001, "Reactor Shutdown due to Reactor Head Vent Steam Leak Constituting Pressure Boundary Leakage."

Enclosed is Licensee Event Report (LER) 254/03-001, "Reactor Shutdown due to Reactor Head Vent Steam Leak Constituting Pressure Boundary Leakage," for Quad Cities Nuclear Power Station.

This report is submitted in accordance with the requirements of the Code of Federal Regulations, Title 10, Part 50.73(a)(2)(i)(A), which requires reporting of the completion of any nuclear plant shutdown required by the plant's Technical Specifications.

Should you have any questions concerning this report, please contact Mr. W. J. Beck at (309) 227-2800.

Respectfully,

TJ Tulon

Timothy J. Tulon
Site Vice President
Quad Cities Nuclear Power Station

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Quad Cities Nuclear Power Station

IE22

NRC FORM 366 (7-2001) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004
LICENSEE EVENT REPORT (LER)	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 Reactor Shutdown due to Reactor Head Vent Steam Leak Constituting Pressure Boundary Leakage

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME N/A	DOCKET NUMBER N/A
05	20	03	03	– 001	– 00	07	21	03	FACILITY NAME N/A	DOCKET NUMBER N/A

9. OPERATING MODE	2	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)			
10. POWER LEVEL	000	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)	X 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)
Wally Beck, Regulatory Assurance Manager	(309) 227-2800

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)	X	NO				

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

On May 20, 2003, at approximately 0240 hours, a small (approximately 12" plume) steam leak was identified on the Unit 1 Reactor Head Vent line. At the time that the leak was identified, the reactor was subcritical in Mode 2 with a planned shutdown in progress and all Emergency Core Cooling (ECCS) systems operable. The leak was identified during the initial drywell entry for the reactor shutdown.

The Reactor Head Vent is utilized to vent non-condensable gases from the reactor vessel head during operation. The vent line attaches to the vessel head at a 4" flanged connection and reduces to a 2" line constructed of A106 Grade B carbon steel, schedule 80, piping with socket welded fittings. The leak was located adjacent to an original construction weld in the 2" section of piping inboard of the isolation valves.

The root cause of this event was inadequate verification of weld quality. The significance of the event was minimal. The leakage rate was low and all ECCS systems were operable. Also, the consequences of a postulated worse case failure of this pipe are within analyzed limits. Corrective actions included replacement of a section of piping and inspection of additional welds on Unit 1 and Unit 2.

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PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor, 2957 Megawatts Thermal Rated Core Power

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

EVENT IDENTIFICATION

Reactor Shutdown due to Reactor Head Vent Steam Leak Constituting Pressure Boundary Leakage

A. CONDITION PRIOR TO EVENT

Unit: 1	Event Date: May 20, 2003	Event Time: 0240 hours
Reactor Mode: 2	Mode Name: Startup	Power Level: 000%

Startup (2) - Mode switch in Startup/Hot Standby position (or in Refuel position with all reactor vessel head closure bolts fully tensioned) with average reactor coolant temperature at any temperature.

B. DESCRIPTION OF EVENT

On May 20, 2003, at approximately 0240 hours, a steam leak was identified on Unit 1 in the 2" Reactor Head Vent [SB] line. The leak was identified during the initial drywell entry for a reactor shutdown (Q1M16).

The Reactor Head Vent is utilized to vent off non-condensable gases from the reactor vessel head during operation. The vent line attaches to the vessel head at a 4" flanged connection and reduces to a 2" line constructed of A106 Grade B carbon steel, schedule 80, piping with socket welded fittings.

The reactor was subcritical in Mode 2 with a planned shutdown in progress and all Emergency Core Cooling (ECCS) systems operable. The individual performing the inspections discovered a small steam leak on the Unit 1 reactor head vent line. The leak was observed as a steam plume approximately 12 inches high. The leak was located in a section of piping inboard of the normally closed head vent isolation valves [ISV] and inboard of the normally open continuous head vent isolation valve. The vent line and coupling [CPLG] are ASME Section XI Class 2 components. The leak was determined to be from a 60 degree circumferential crack in a fillet weld on a 2-inch coupling on the reactor head vent line. The weld is original construction (1970).

At 0630 hours on May 20, 2003, a 4-hour Emergency Notification System notification was made per 10CFR50.72(b)(2)(i) for a reactor shutdown required by Technical Specifications due to reactor coolant pressure boundary leakage (Technical Specification 3.4.4, Condition C).

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C. CAUSE OF EVENT

The root cause of this event was inadequate verification of an original construction weld. The socket weld was poor quality as evidenced by significant porosity, lack of fusion and excessive overlap in the failure region. The defects propagated due to long-term corrosion and possibly fatigue to form the through-wall leak.

D. SAFETY ANALYSIS

The significance of the event was minimal. The leakage rate was low and all ECCS systems were operable.

The Drywell leakage [IJ] is monitored every four hours and a Drywell Continuous Air Monitoring system [IK] provides a Control Room alarm if particulate activity reaches a predetermined value. Also, the Drywell air temperature [IM] is trended daily in accordance with QCOS 1600-53.

The consequence of a postulated worse case failure is within analyzed limits. The area of the leaking line is less than 0.12 square feet even if it is doubled to account for the potential loss of reactor coolant from both ends of a postulated worst-case vent line break.

UFSAR Section 3.4.1.2.3, Protection of the Drywell and Torus, documents the evaluation of the internal flooding measures in the containment [NH], so that the worst-case failure of the reactor head vent line would not result in an accumulation of water beyond the analyzed limits.

UFSAR Section 15.6.5, Loss of Coolant Accidents Resulting from Piping Breaks Inside the Containment, documents the evaluation of the primary system piping failures, so that the consequences of the worst-case failure of the reactor head vent line would remain within the bounds of an analyzed small steam line break.

E. CORRECTIVE ACTIONS

Immediate Actions

The reactor shutdown that was in progress was continued to completion, and a 4-hour ENS notification was made.

An ultrasonic thickness examination of the line was performed to verify minimum wall thickness.

Corrective Actions Completed:

The failed weld, coupling and two-foot section of pipe were removed, and a new section of pipe and couplings was installed. The new welds were visually inspected and liquid penetrant tested.

A visual inspection was performed of the additional socket welds on the Unit 1 line and on the similar coupling welds on Unit 2. No additional weld defects were identified.

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F. PREVIOUS OCCURRENCES

On February 27, 1998, during a Reactor Vessel Class One Leak Test, a leak was identified in the heat affected zone of a coupling weld on the reactor vessel bottom head drain line. Failure analysis identified the failure as outside diameter initiated stress corrosion cracking. This event was reported in LER 1-98-012.

G. COMPONENT FAILURE DATA

There were no component failures associated with this event.