LICENSEE EVENT REPORT (LER) — Form Rev. 2.0

Facility Name (1)	Docket Number (2)	Page (3)
Quad Cities Unit One/Two	0 5 0 0 0 2 5 4	1 of 0 5

Title (4)

Engineering Calculations performed in response to NRC Generic Letter 96-06 indicated that several isolable piping sections on each unit may experience stresses above UFSAR Allowables following a Loss of Coolant Accident due to inadequate original system design.

Event Date (5)			LER Number (6)			Report Date (7)			Other Facilities Involved (8)	
Month	Day	Year	Year	Sequential Number	Revision Number	Month	Day	Year	Facility Names	Docket Number(s)
0 1	2 7	9 7	9 7	-- 0 0 2 --	0 0	0 2	2 4	9 7	QC Unit 2	0 5 0 0 0 2 6 5
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 1 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

	20.402(b)		20.405(c)		50.73(a)(2)(iv)		73.71(b)
	20.405(a)(1)(i)		50.36(c)(1)		50.73(a)(2)(v)		73.71(c)
	20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)		Other (Specify in Abstract below and in Text)
	20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		
	20.405(a)(1)(iv)	X	50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)		
	20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (AREA CODE)
Charles Peterson, Regulatory Affairs Manager, ext. 3609	3 0 9 6 5 4 - 2 2 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

		Expected Submission Date (15)	Month	Day	Year
X YES (If yes, complete EXPECTED SUBMISSION DATE)	NO		0 5	3 1	9 7

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

ABSTRACT:

On 012797, with both Units in Mode One at 100% power, it was determined that several isolable piping sections could experience stresses above Updated Final Safety Analysis Report Allowables due to Post-Loss of Coolant Accident thermal pressurization. In response to NRC Generic Letter 96-06, several calculations were performed to evaluate piping and containment penetrations for thermal overpressurization caused by a Main Steam Line Break(MSLB)/Loss of Coolant Accident(LOCA). These calculations identified five penetrations(per Unit) that could experience overpressurization under accident conditions. The affected piping systems include: Reactor Building Closed Cooling Water, Reactor Recirculation Sample lines, Residual Heat Removal, Reactor Water Clean-up, and Clean Demineralized Water. The affected piping sections were determined to be operable, but degraded. The cause of the event was inadequate original design.
Based on the results of the operability assessment, there is minimal safety significance to the station or the health and safety of the public as a result of this event. The affected penetrations would maintain containment integrity following an accident. In the event of an accident, any radiological release would remain within analyzed limits. The corrective actions taken to resolve this issue are being developed and will be provided in a supplemental report.

LER254\97\002.WPF

PLANT AND SYSTEM IDENTIFICATION:

General Electric - Boiling Water Reactor - 2511 MWt rated core thermal power.

EVENT IDENTIFICATION: Engineering Calculations performed in response to NRC Generic Letter 96-06 indicated that several isolable piping sections on each unit may experience stresses above UFSAR Allowables following a Loss of Coolant Accident due to inadequate original system design.

A. CONDITIONS PRIOR TO EVENT:

Unit: One Event Date: 012797 Event Time: 2100
Reactor Mode: 1 Mode Name: POWER OPERATION Power Level: 100%

Unit: Two Event Date: 012797 Event Time: 2100
Reactor Mode: 1 Mode Name: POWER OPERATION Power Level: 100%

This report was initiated by Licensee Event Report 254\97-002.

Power Operation (1) - Mode switch in the RUN position with average reactor coolant temperature at any temperature.

B. DESCRIPTION OF EVENT:

On 012797, with both Units in Mode One at 100% power, it was determined that several isolable piping sections on both units could experience stresses above Updated Final Safety Analysis Report(UFSAR) Allowables due to Post Loss of Coolant Accident(LOCA) thermal pressurization. The allowable piping stresses are described in UFSAR Section 3.9.

In response to NRC Generic Letter(GL) 96-06, "Assurance of Equipment Operability and Containment Integrity During Design - Basis Accident Conditions", piping systems that penetrate containment were evaluated to determine if they were susceptible to thermal overpressurization. This evaluation resulted in the development of four calculations. Calculation QDC-1600-M-0299 determined the maximum temperature change in water trapped between two closed valves during accident conditions for a variety of containment penetration configurations. Calculation QDC-1600-M-0297 developed a methodology to calculate an incremental pressure increase per degree Fahrenheit(F) as a function of pipe outside diameter to wall thickness ratio and final water temperature. Pressure changes in the isolated pipe sections could then be predicted using the pipe material and size, the pressure increase per degree F, and the calculated temperature rise. Calculation QDC-1600-M-298 determined the maximum permissible piping pressures for design, upset, emergency, and faulted conditions to determine the pressure retaining capability of isolated pipe sections under thermal pressurization conditions. Calculation QDC-1600-M-0296 reviewed containment penetrations and compared the predicted pressure increase with the maximum permissible piping pressures to determine if maximum permissible pressures were exceeded.

Five penetrations(per Unit) were determined to be susceptible to the overpressurization conditions discussed in GL 96-06. The penetration and associated descriptions are as follows:

Penetration X-24 - Reactor Building Closed Cooling Water(RBCCW) Return Piping from the Drywell
Penetration X-41 - Reactor Recirculation[AD] System Sample Line Piping*
Penetration X-12 - Residual Heat Removal(RHR) Shutdown Cooling[BO] Water Suction Piping*
Penetration X-14 - Reactor Water Cleanup(RWCU)[CE] Pump Suction Piping*
Penetration X-20 - Clean Demineralized Water[KC]Piping

These penetrations would be heated during a LOCA or a Main Steamline(MSL) break inside containment. Penetrations designated by an asterisk close automatically on various group isolation signals.

An operability assessment was completed 012797 to document the basis for continued operability of the affected penetrations and systems/components. The basis for operability of these affected penetrations includes consideration of one or more of the following: expansion of the trapped fluid in voided areas of the isolated piping section, leakage from one of the containment isolation valves (seat, packing, or body to bonnet flange), insulation of piping to delay the temperature increase, lifting of air operated valves due to the pressure increase, and plastic straining of the affected pipe to accommodate the pressure increase. The criteria provided in ASME Section III, Appendix F was utilized to evaluate the operability of the affected piping sections.

C. <u>APPARENT CAUSE OF EVENT</u>:

The cause of this event is attributed to inadequate original design. The potential for thermal pressurization of isolated containment penetrations due to elevated post accident conditions was not adequately anticipated in the original penetration/piping design. The root cause of the original design error is unknown.

D. <u>SAFETY ANALYSIS OF EVENT</u>:

There is minimal safety significance to the station or the health and safety of the public as a result of this event. The results of the operability assessment concluded that the affected penetrations were Operable, but degraded. The affected penetrations would maintain containment integrity following an accident. In the event of an accident, any radiological release would remain within analyzed limits.

Additionally, it should be noted that for three of the penetrations, it is unlikely that the conditions needed for a thermal overpressurization condition to develop would be present following an accident.

The penetrations associated with RWCU(X-14) and the Reactor Recirculation System Sample Line(X-41) normally contain "hot" process flow and would not experience thermal heat-up during an accident. These penetrations are only susceptible to thermal overpressurization if the lines are isolated during operation. The valves associated with these penetrations are currently open.

The penetration associated with the RBCCW piping(X-24) is normally open, and must remain open during operation because it supplies cooling water to the Reactor Recirculation Pumps. RBCCW is a closed loop system and does not interface with the Reactor Coolant Pressure Boundary(RCPB). This penetration is only susceptible to thermal overpressurization if the lines are water solid <u>and</u> isolated following an accident. The valves associated with this penetration do not have an automatic isolation signal and are procedurally isolated if a break in the RBCCW piping is detected. Therefore, in order for thermal overpressurization to occur, the LOCA event, a RBCCW system breach, and water solid conditions between the isolation valves must occur concurrently.

E. <u>CORRECTIVE ACTIONS</u>:

<u>Corrective Actions Completed/In Progress</u>:

1. An Initial Operability Evaluation was completed on 012797 which concluded that the affected penetrations were operable and containment integrity was intact.

2. On 013097, the Operability Evaluation was revised to address additional corrective actions that will be taken.

3. During 1996, Quad Cities performed several reviews intended to assure conformance with the UFSAR. A comprehensive UFSAR Compliance/Design Basis Review Initiative is under development.

<u>Corrective Actions Scheduled:</u>

1. Alternative solutions to resolve the potential thermal overpressurization will be evaluated. The specific corrective actions and implementation schedule for each affected penetration will be provided to the NRC in an Update to the Quad Cities Generic Letter 96-06 response by 053197.[NTS 2541809700201;Engineering]

2. Quad Cities will evaluate the acceptability of partially draining the piping associated with penetrations X-12 and X-20. This action will be complete by 022897. [NTS 2541809700202;Engineering]

3. Subject to the above approval(Item 2), the piping associated with penetrations X-20, Clean Demineralized Water to the Drywell, will be partially drained. Provisions will be made to ensure that this piping is drained prior to Unit start-up in the future. This action will be complete by 032897. [NTS 2541809700203;Operations]

4. Subject to the above approval(Item 2), the piping associated with penetrations X-12, RHR Shutdown Cooling Suction piping, will be partially drained. Provisions will be made to ensure that these actions remain in place until final actions to resolve this concern are implemented. Appropriate procedure changes will also be made to ensure the RHR Shutdown Cooling is filled and vented prior to operation. This action will be completed during the next cold shutdown for Unit 1 and prior to startup from refueling outage, Q2R14 for Unit 2. [NTS 2541809700204;Engineering]

5. The modification process will be evaluated to ensure that future modifications account for overpressurization concerns.[NTS 2541809700205;Engineering]

6. A discussion covering the issues raised by NRC GL 96-06 will be included in the training program for Engineering personnel.[NTS 2541809700206;Engineering]

F. PREVIOUS EVENTS:

A review of previous Licensee Event Reports (LER) at Quad Cities Station Units One and Two, since 010195 concerning primary containment issues or deficiencies related to plant design identified the following previous events:

LER Number	Description
1-95-002	POTENTIAL PROBLEM WITH THE SIZING OF THE THERMAL OVERLOADS ON THE BOOSTER FANS FOR THE CONTROL ROOM HVAC
1-95-003	TIP SYSTEM RESPONSE TO A PCIS SIGNAL
1-95-007	RPS SCRAM DISCHARGE VOLUME LOGIC CIRCUIT IS DESIGNED SUCH THAT A SINGLE COMPONENT FAILURE WOULD PREVENT A FULL SCRAM
1-96-002	DRESDEN STATION NOTIFIED THE STATION THAT OUR CONTROL ROOM VENTILATION ISOLATION MAY BE DEFICIENT.
1-96-009	DEGRADED VOLTAGE CONCERNS DUE TO MCC 29-2 CABLE LENGTHS
1-96-011	NRC IN 92-18 DESCRIBES A CONDITION WHERE MOVS CAN BE DAMAGED DUE TO HOT SHORTS DUE TO FIRE
1-96-015	PIPE WHIP RESTRAINT JIHP-3
1-96-016	REACTOR BUILDING BLOWOUT PANELS
1-96-020	CONTROL ROOM BOOSTER FANS
1-96-022	B CONTROL ROOM HVAC REFRIGERATION UNIT HEATER
1-96-025	ECCS SUCTION STRAINERS HEAD LOSS VALUE IS INCORRECT.
2-95-006	U-2 MCC 29-2 TRIPPED ON OVERLOAD.

G. COMPONENT FAILURE DATA:

There is no component failure associated with this event. There have been no penetration failures at Quad Cities Station attributed to thermal overpressurization.