NRC Form 366 (6-89) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
PLANT E I. HATCH, UNIT 2	0 5 0 0 0 3 6 6	1 OF 6

TITLE (4)
"AS-FOUND" PRIMARY CONTAINMENT INTEGRATED LEAKAGE RATE TEST FAILURE

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQ NUM	REV	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
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0 1	2 6	9 3	9 3	0 0 2	0 0	0 2	0 2	9 3		0 5 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR (11)

OPERATING MODE (9): 1	[ ] 20.402(b)	[ ] 20.405(c)	[ ] 50.73(a)(2)(iv)	[ ] 73.71(b)
POWER LEVEL: 1 0 0	[ ] 20.405(a)(1)(i)	[ ] 50.36(c)(1)	[X] 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)
	[ ] 20.405(a)(1)(iii)	[ ] 50.73(a)(2)(i)	[ ] 50.73(a)(2)(viii)(A)	
	[ ] 20.405(a)(1)(iv)	[ ] 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	
STEVEN B TIPPS, MANAGER NUCLEAR SAFETY AND COMPLIANCE, HATCH	912	367-7851

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

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORT TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORT TO NPRDS
X	B B	I S V	P 3 0 5	Y E S		X	B B	V	R 3 4 4	N O
X	B B	V	R 3 4 4	N O		X	B B	S E A L	F 1 5 3	N O

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
[ ] YES (If yes, complete EXPECTED SUBMISSION DATE)	[X] NO				

ABSTRACT (16)

On 1/26/93, Unit 2 was in the Run mode at 2436 CMWT (approximately 100 percent of rated thermal power). An Integrated Leakage Rate Test (ILRT) of the Unit 2 primary containment had been previously performed on November 6-7, 1992 during the tenth refueling outage. The as-left containment leakage rate was found to be 0.8858 weight percent per day, which was within the Technical Specification limit of 0.9 weight percent per day (.75La). Subsequent to the outage, the Architect Engineer (A/E) compiled the local leakage rate testing (LLRT) results, which determine the leakage rate improvements, and calculated the "as found" containment integrated leakage rate. The "as found" leakage rate was determined to be 1.3357 weight percent per day. This was in excess of the Technical Specification limit of 1.2 weight percent per day (La). The ILRT report was subsequently completed by the A/E and submitted for management review on 1/14/93. On 1/26/93, in reviewing the report, plant management determined that the excessive "as found" integrated containment leakage rate represented a reportable condition. No corrective actions were required at this time since the leakage rate had been previously restored to the appropriate Technical Specification limit prior to restart from the refueling outage. The cause of the event was degradation of components comprising the primary containment boundary, in particular, those comprising the boundary for penetration X-222B. Corrective actions included repairing valves and replacing a flange gasket.

TEXT

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor Energy Industry Identification System Codes are identified in the text as (EIIS Code XX).

DESCRIPTION OF EVENT

On 1/26/93, Unit 2 was in the Run mode at 2436 CMWT (approximately 100 percent of rated thermal power). At that time, plant management, in reviewing the Unit 2 primary containment (EIIS Code NH) Integrated Leakage Rate Test (ILRT) report for the fourth periodic ILRT determined that a reportable condition existed in that the "as found" leakage rate for primary containment had exceeded the Technical Specification limit. The ILRT had been performed on November 6-7, 1992 prior to starting up from the tenth refueling outage. The purpose of the test is to demonstrate Unit 2 primary containment integrity and is required by 10 CFR 50, Appendix J and the Unit 2 Technical Specifications on a 40 $\pm$ month frequency.

The "as left" leakage rate of the Unit 2 primary containment as determined by the ILRT on November 6-7, 1992, was within the Technical Specification limits. Local leak rate testing of the primary containment penetrations, also required by 10 CFR 50, Appendix J and the Unit 2 Technical Specifications, was also performed during the outage. The results of the tests were later compiled by the Architect Engineer (A/E) and used to calculate the "as found" integrated leakage rate of primary containment. The "as found" leakage rate based on the total time analysis method as described in ANSI N45.4 - 1972, "Leakage-rate Testing of Containment Structures for Nuclear Reactors," was determined to be 1.3357 weight percent per day and was in excess of the design leakage limit for the primary containment of 1.2 weight percent per day, also a Technical Specification limit. The ILRT report required by 10 CFR 50, Appendix J was subsequently compiled by the A/E and submitted to plant management.

On 1/26/93, in reviewing the report, plant management determined that a reportable condition existed in that the "as found" ILRT leakage rate results were in excess of the Technical Specification limit. A notification was made to the NRC in accordance with 10 CFR 50.72 (b)(2)(iii). No further actions were required since the leakage rate had been previously restored within the appropriate Technical Specification limit prior to restart from the refueling outage.

TEXT

CAUSE OF EVENT

The cause of this event was degradation of components comprising the primary containment boundary. During the operating cycle, service related wear of components is expected. Expected wear is an acceptable phenomenon and is accounted for by the Technical Specifications in that the leakage rates at the beginning of each test period are required to be significantly less than that assumed in an accident at the postulated peak accident pressure of 57.5 psig. In this case, the primary cause of exceeding the containment leakage rate was an apparent failure of valve 2T49-F004B, Recombiner "B" Discharge Isolation Valve, to fully seat in combination with two instances of valve packing leakage and a flange leak in the Recombiner System (EIIS Code BB).

Valve 2T49-F004B provides the inboard isolation for primary containment penetration X-222B. The outboard isolation boundary is provided by the Recombiner System piping, which is a closed loop system taking suction from and discharging to primary containment. During local leak rate testing, valve 2T49-F004B was found to be leaking in excess of 0.549 weight percent per day. Additionally, several leaks in the Recombiner System, specifically, packing leaks on valves 2T49-F003B and 2T49-F008B and a flanged connection leak, resulted in a leakage rate of 0.331 weight percent per day. Thus, the minimum pathway leakage for penetration X-222B was 0.331 weight percent per day, which was approximately 25 percent of the overall containment leakage rate and resulted in the "as found" leakage rate exceeding the Technical Specification limit.

It is apparent from the amount of leakage experienced with valve 2T49-F004B and from the condition of the valve seating surfaces that the valve did not fully seat during the leakage testing. It appears that the torque switch tripped the valve motor operator on high torque before the valve was fully seated resulting in a large amount of leakage past the valve seats. The reason for the torque switch tripping before the valve fully seated could not be determined. The results of the local leak rate test performed on this valve on 4/15/91 indicated no leakage passed the valve seats. Plant personnel reviewed the maintenance work history on this valve between the 4/15/91 test and the local leak rate test performed on 10/12/92. No work was performed on the valve during this approximately 18-month period which would have resulted in a change in the torque switch setting. Furthermore, no work was performed which would have required an increased motor operator torque to close the valve. It is apparent that the torque required to fully seat the valve increased at some point during the operating cycle due to wear and/or increased friction associated with the valve and/or valve drive train.

The leakage from the flange connection of the "B" Recombiner reaction chamber discharge piping was most likely due to less than optimum installation of the flange connection gasket. The leakage rate of this connection alone could not be quantified. It is possible that the leakage was at an acceptable level. However, the gasket was replaced since it contributed to the excessive leakage from the Recombiner System. Packing leaks associated with Recombiner system valves 2T49-F003B and F008B were apparently due to service related degradation.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This report is required pursuant to 10 CFR 50.73 (a)(2)(v) in that the integrated leakage rate of primary containment was in excess of that allowed by the Technical Specifications. Specifically, the "as found" leakage rate was 1.3357 weight percent per day as opposed to the Technical Specification limit of 1.2 weight percent per day.

The purposes of the Unit 2 primary containment and its penetration sealing devices are to contain any radioactive material which might be released during and following a postulated design basis accident and to limit leakage paths and associated leakage rates to those assumed in the accident analyses. The primary containment is designed to limit the unrestricted release of radioactive material to the secondary containment (EIIS Code NG) to 1.2 weight percent per day in the unlikely event of a design basis accident resulting in a peak containment pressure of 57.5 psig. The Standby Gas Treatment System (SGTS, EIIS Code BH) draws down the secondary containment to a negative pressure within 120 seconds after the initiating event. Any leakage from the primary containment into secondary containment after the 120 second draw down time would be filtered through 95 percent efficient filters in SGTS and then released via the Main Stack elevated release point. These systems function in combination to limit the site boundary radiation doses and the doses to personnel in the Main Control Room to within the limits of 10 CFR 100.

The Unit 2 Technical Specifications require that periodic verification testing of the leak-tight integrity of the primary containment isolation barriers be performed. This testing allows timely detection of degradation such that maintenance and repairs can be performed as necessary to restore leakage rates to within their Technical Specifications limits. The maintenance of primary containment integrity in conjunction with the Technical Specifications leakage rates will maintain the potential site boundary radiation doses below the limits of 10 CFR 100 during postulated accident conditions.

In this event, the "as found" leakage rate for the primary containment was 0.1357 weight percent per day greater than that assumed in the accident analysis. A preliminary analysis of site boundary doses and doses to the Main Control Room personnel with the increased release rate showed that the doses remained within the 10 CFR 100 limits. A formal analysis will be completed by 2/28/93 and a revision to this report will be submitted if the results are contrary to the preliminary conclusion.

Based on the above information, it as concluded that this event had no adverse impact on nuclear safety. This assessment applies to all operating conditions.

TEXT

CORRECTIVE ACTIONS

Recombiner "B" Discharge Isolation Valve 2T49-F004B was repaired. The valve disc and seats were cleaned, and the torque switch was replaced. The torque switch setting was increased from the "as found" value of 2.5 to 3.0 per maintenance engineering instructions and plant procedure 52GM-MEL-022-OS, "Limitorque Valve Operator Electrical Maintenance." Also, a second lantern ring was added to the stem packing to reduce the stem-to-packing friction. Another local leak rate test was then performed on the valve on 10/31/92. The valve leakage rate was found to be at an acceptable level.

The gasket on the "B" Recombiner reaction chamber discharge piping was replaced, and Recombiner System valves 2T49-F003B and F008B were repacked. Another local leak rate test was then performed on the system piping on 11/12/92 and the leakage rate was found to be at an acceptable level. These corrective actions decreased the minimum pathway leakage rate through primary containment penetration X-222B from 0.3310 weight percent per day to less than 0.0004 weight percent per day.

Personnel responsible for ILRT activities have been informed of the need to initiate a deficiency card to document non-conformances and obtain reportability evaluations even if a deficiency has been corrected at the time of discovery.

A formal analysis of the affects of the increased leakage rate on doses to the Main Control Room personnel and at the site boundary will be completed by 2/28/93. An update to this report will be submitted if the analysis shows that 10 CFR 100 limits could be exceeded.

ADDITIONAL INFORMATION

No systems other than those previously identified in this report were affected by this event.

One previous similar event was reported in the past two years in which the degradation of primary containment components resulted in a primary containment leakage rate limit being exceeded. This event was reported in LER 50-366/91-008, dated 4/24/91, and involved redundant Main Steam Isolation Valves (MSIVs) in the same penetration each exceeding their leakage limit of 11.5 standard cubic feet per hour (SCFH). Corrective action for the event involved refurbishing the valves and retesting to demonstrate leakage integrity. This action would not have prevented this event because it involved the MSIVs and not the components involved in this event.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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TEXT

Failed Components Information:

Master Parts List Number: 2T49-F004B
Manufacturer: William Powell Co.
Model Number: 1523WE
Type: 4-inch motor-operated gate valve
Manufacturer Code: P305
EIIS System Code: BB
EIIS Component Code: ISV
Reportable to NPRDS: Yes
Root Cause Code: X

Master Parts List Number: 2T49-F003B
Manufacturer: Rockwell Manufacturing
Model Number: 3624
Type: 3-inch motor-operated gate valve
Manufacturer Code: R344
EIIS System Code: BB
EIIS Component Code: V
Reportable to NPRDS: No
Root Cause Code: X

Master Parts List Number: 2T49-F008B
Manufacturer: Rockwell Manufacturing
Model Number: 3624
Type: 3-inch motor-operated gate valve
Manufacturer Code: R344
EIIS System Code: BB
EIIS Component Code: V
Reportable to NPRDS: No
Root Cause Code: X

Master Parts List Number: None
Manufacturer: Flexitallic
Model Number: CG API601
Type: Gasket
Manufacturer Code: F153
EIIS System Code: BB
EIIS Component Code: SEAL
Reportable to NPRDS: No
Root Cause Code: X