Commonwealth Edison
LaSalle County Nuclear Station
2601 N. 21st. Rd.
Marseilles, Illinois 61341
Telephone 815/357-6761

November 27, 1992

Director of Nuclea Reactor Regulation
U.S. Nuclear Regulatory Commission
Mail Station P1-137
Washington, D.C. 20555

Dear Sir:

Licensee Event Report #92-014-00, Docket #050-374 is being submitted to your office in accordance with 10CFR50.73(a)(2)(i).

WRo [signature]
G. J. Diederich
Station Manager
LaSalle County Station

GJD/MJO/mkl

Enclosure

xc: Nuclear Licensing Administrator
NRC Resident Inspector
NRC Region III Administrator
INPO - Records Center
IDNS Resident Inspector

300041

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1/1

LICENSEE EVENT REPORT (LER)

Form Rev 2.0

Facility Name (1)	Docket Number (2)	Page (3)
LaSalle County Station Unit 2	0 5 0 0 0 3 7 4	1 of 0 5

Title (4)

Improper Performance of Local Leak Rate Test on RWCU Return Isolation Valve Due to Procedural Inadequacy

Event Date (5) Month	Day	Year	LER Number (6) Year	Sequential Number	Revision Number	Report Date (7) Month	Day	Year	Other Facilities Involved (8) Facility Names	Docket Number(s)
									LaSalle Unit 1	0 5 0 0 0 3 7 3
1 0	2 8	9 2	9 2	0 1 4	0 0	1 1	2 7	9 2		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)	X	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
Michael Oclon, ISI Coordinator, Extension 2117	AREA CODE 8 1 5 3 5 7 - 6 7 6 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
D	C E			N					

SUPPLEMENTAL REPORT EXPECTED (14)

		Expected Submission Date (15)	Month	Day	Year
Yes (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

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

At 1240 hours, on October 12, 1992, with Unit 2 in Operational Conditions One (Run) at 100% power, it was determined that the Local Leak Rate Test (LLRT) on the Reactor Water Cleanup (RWCU, RT) [CE] Return Containment Isolation Valve 2G33-F040 was improperly performed because the isolation valve was not properly vented during the performance of the Local Leak Rate Test (LLRT).

It was determined that the surveillance requirements of Technical Specification 4.6.1.1.a was not satisfied. This resulted in non-compliance with Technical Specification 3.6.3 which required declaring valve 2G33-F040 inoperable. This appropriate Action Statement required a Unit 2 shutdown. However, at 1530 hours, LaSalle County Station Unit 2 received a waiver of compliance with regard to the above surveillance requirements as it pertains to valve 2G33-F040.

The cause of the improper test was the belief that a check valve could not be considered leak tight, especially at low differential pressures, and therefore would not be a barrier to a vent path. This assumption was erroneous. A satisfactory Type C Local Leak Rate Test on the 2G33-F040 valve will be performed at the first available outage in which Unit 2 is in Cold Shutdown for a duration of two weeks or greater and no later than the next refuel outage L2R05.

This is reportable to the NRC pursuant to 10CFR50.73(a)(2)(i)(B), a condition prohibited by the plant's Technical Specifications.

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor

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

A. CONDITION PRIOR TO EVENT

Unit(s): 2 Event Date: 10/28/92 Event Time: 1240 Hours

Reactor Mode(s): 1 Mode(s) Name: Run Power Level(s): 100%

B. DESCRIPTION OF EVENT

On October 28, 1992, Unit 2 was in Operational Condition 1 (Run) at 100 percent power. At 1240 CDT, it was determined that the Local Leak Rate Test (LLRT) on Reactor Water Cleanup (RWCU, RT)[CE] Return Containment Isolation Valve 2G33-F040 may have been improperly performed.

LaSalle County Station Unit 2 Technical Specification 3.6.3 requires that Primary Containment Isolation Valves remain operable in Operational Conditions 1, 2, and 3. Due to concerns over the validity of the results of 10 CFR 50 Appendix J Type C testing previously performed on the Reactor Water Cleanup Return to Feedwater Valve 2G33-F040, this valve was declared inoperable. Primary Containment Isolation is accomplished on these lines by two check valves on each feedwater line and valve 2G33-F040. It is not possible to isolate the penetration per action statement a.1., and Unit shutdown is therefore required by action statement a.2.

Therefore, Commonwealth Edison requested a waiver of compliance with Technical Specification 3.6.3 action a.2. be granted to allow continued unit operation until approval of an emergency Technical Specification amendment. The amendment will allow valve 2G33-F040 to be excluded from the list of Primary Containment Isolation Valves which require a 10 CFR 50 Appendix J Type C test in order to be considered operable in Operating Conditions 1, 2, and 3.

During the performance of 10 CFR 50 Appendix J Type C testing (Local Leak Rate Testing) on Unit 1 during its current refuel outage, a NRC Resident Inspector questioned the appropriateness of the vent path established for the Unit 1 Reactor Water Cleanup Return to Feedwater Valve, 1G33-F040. This valve is a four-inch flex wedge gate valve manufactured by Anchor Darling. The reactor coolant pressure boundary piping configuration being tested consisted of two feedwater lines, bounded by valves 1B21-F065A and 1B21-F065B, and a reactor water cleanup line bounded by valve 1G33-F040. Commonwealth Edison has maintained that the vent path for 1G33-F040, although separated by a check valve (1G33-F039) from the leak rate boundary valve, was adequate because check valves cannot be considered leak-tight. In addition, based upon verbal conversations during past Appendix J inspections, CECo believed that this test configuration had been acceptable to the NRC staff. After discussions with NRC Region III personnel and the Senior Resident Inspector, CECo believes that the nonconservative testing verified that the combination of the valves had acceptable leakage, not the individual valve.

B. DESCRIPTION OF EVENT (CONTINUED)

The as-found leakage past valve 1G33-F040, when tested with the existing vent path, was determined to be 0.0 scfh. As a result of the concern over the vent location, the valve was leak tested in the reverse direction, with measured leakage of 0.0 scfh.

The comparable Unit 2 piping configurations was reviewed, and found to be identical to Unit 1. As Unit 2 was in Operational Condition 1, it was not possible to test valve 2G33-F040 in the reverse direction. As a result, valve 2G33-F040 was declared inoperable, and Technical Specification 3.6.3 action a. was consulted. The Primary Containment Isolation Valves involved in this Type C test boundary consist of non-slam check valves 2B21-F010A and 2B21-F010B located inside the primary containment, and air operated testable check valves 2B21-F032A and 2B21-F032B located outside the primary containment. In addition, motor operated gate valves 2G33-F040, 2B21-F065A and 2B21-F065B, located in the containment are provided to ensure long term isolation capability. With 2G33-F040 inoperable, using check valves to satisfy Technical Specification 3.6.3 action a.1. is not possible, it was necessary to comply with action a.2., and be in Hot Shutdown within the next 12 hours. A temporary waiver of compliance was granted to allow continued Unit 2 operation.

C. APPARENT CAUSE OF EVENT

It was determined that the Local Leak Rate Test performed on the RWCU Return Containment Isolation Valve, 2G33-F040, per LTS-100-11 was invalid because the upstream side of the valve was not properly vented. This discrepancy was previously identified during review of the procedure. The concern was addressed by Technical Staff supervision, and it was determined that this method was acceptable. The cause of the improper test was the belief that a check valve could not be considered leak tight especially at low differential pressures and therefore would not be a barrier to a vent path. This assumption was erroneous. The 2G33-F040 was tested in the normal direction from the containment side. A vent path was established by opening test tap valves 2G33-F037 and 2G33-F038. However, a check valve, 2G33-F039, is installed between the 2G33-F040 containment isolation valve and the open vents. If check valve 2G33-F039 is leak tight, then any leakage through the 2G33-F040 containment isolation valve would not been seen and would go unnoticed.

D. SAFETY ANALYSIS OF EVENT

Although the 2G33-F040 valve is not considered as a containment isolation valve during initial Loss of Coolant Accident (LOCA) conditions, it is utilized for long term leakage control purposes for the Feedwater (FW) [SJ] lines along with the 2B21-F065A/B valves. During initial LOCA conditions when the primary containment would be subjected to peak accident pressure (Pa) the inboard and outboard feedwater check valves 2B21-F010A/B and 2B21-F032A/B would be acting as the containment barriers as the 2G33-F040 and 2B21-F065A/B valves do not receive an automatic isolation signal.

D. SAFETY ANALYSIS OF EVENT (CONTINUED)

Prior to Unit 2 startup from its last refuel outage L2R04 in March of 1992, the primary containment feedwater penetrations were tested successfully with the following test results:

2B21-F010A Inboard Feedwater Check......................3.04 scfh
2B21-F032A Outboard Feedwater Check....................12.08 scfh

2B21-F010B Inboard Feedwater Check......................1.03 scfh
2B21-F032B Outboard Feedwater Check....................0.0 scfh

2B21-F065A Feedwater "A" Outboard Stop.................0.65 scfh
2B21-F065B Feedwater "B" Outboard Stop.................0.56 scfh
2G33-F040 Reactor Water Cleanup Return Stop...........0.83 scfh

Although it has been determined that the LLRT test results for the 2G33-F040 valve are invalid, it can be reasonably concluded that under LOCA conditions, the actual leakage from the containment through the feedwater lines would be maximum of 12.08 scfh for the "A" Feedwater line and 1.03 scfh for the "B" Feedwater line. The minimum-pathway leakage is 3.04 scfh and 0.0 scfh for the "A" and "B" Feedwater lines respectively.

Prior to Unit 2 startup from L2R04, the total 0.6 La maximum allowable containment leakage (max-path) was determined to be 96.93 scfh, well below the Technical Specification limit of 231.4 scfh. The 0.6 La total included the worst leaking components in each Feedwater line, 12.08 scfh (32A) and 1.03 scfh (10B), for a total of 13.11 scfh for the Feedwater penetrations.

The worst case scenario for this case is that the maximum allowable 0.6 La containment maximum-path leak rate would be exceed if the 2G33-F040 was leaking at a high rate. Although this valve is in the 0.6 La administrative program, the actual penetration leakage by itself would not pose any significant tasks or hazards to the public since the total maximum-pathway leakage does not definitively represent the probable leakage from the containment under accident conditions. In this case where two containment barriers (inboard/outboard check valves) have been proven to be reliable, exceeding the limit solely due to 2G33-F040, would contribute little or nothing to a radiological release under accident conditions.

E. CORRECTIVE ACTIONS

A temporary Waiver of Compliance was requested by LaSalle Station and granted by the Nuclear Regulatory Commission to allow LaSalle Station Unit 2 to continue operation. Reference letter from Mr. Richard Barrett to Mr. Thomas J. Kovach dated October 30, 1992.

LaSalle Station has committed to performing a satisfactory Type C Local Leak Rate Test on the 2G33-F040 valve at the first available outage in which Unit 2 is in Cold Shutdown for a duration of two weeks or greater and no later than the next refuel outage L2R05 presently scheduled for Fall 1993.

E. CORRECTIVE ACTIONS (CONTINUED)

A special test procedure (LST) was written to perform a Local Leak Rate Test (LLRT) on the Unit 1 Reactor Water Cleanup (RT, [CE] Return Containment Isolation Valve, 1G33-F040 valve was successfully tested in the reverse direction on 10/24/92. A proper vent path was established via the "B" Feedwater (FW) [SJ] line.

A Local Leak Rate Test Program review was conducted to verify and ensure that other discrepancies of this nature do not exist at LaSalle Station. The review concluded that the discrepancy identified with the Local Leak Rate Test of the 1(2)G33-F040 valve was an isolated case and no other discrepancies were found.

F. PREVIOUS EVENTS

None.

G. COMPONENT FAILURE DATA

None.

SHEET 7

EVENT SUMMARY AND CAUSE CODES

DVR Number
01-2-92-076

- [ ] Lost generation
- [ ] Cost > $25,000
- [ ] Hazard or Spill
- [ ] Personnel injury
- [ ] Reactor trip
- [ ] ESF actuation
- [ ] NRC reportable
- [x] LER
- [ ] PSE
- [ ] NRC violation, level__
- [ ] GSEP event, class____
- [ ] Tech Spec LCO
- [ ] Potential or future los
- [ ] SALP functional area__

Component type | Failure mode | Department

X
X
X

Licensed? L or blank | Level | Department | Type | Detail code

A
A
A

Type | Detail Code | Department

B
B
B

Type | Detail code

C

Type of deficiency | Detail code | Procedure type

D E
D
D

Type | Detail code | Department

E
E
E
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