Commonwealth Edison
Braidwood Nuclear Power Station
Route #1, Box 84
Braceville, Illinois 60407
Telephone 815/458-2801

June 7 1991
BW/91-0516

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

Dear Sir:

The enclosed Licensee Event Report from Braidwood Generating Station is being transmitted to you in accordance with the requirements of 10CFR50.73(a)(2)(iv), 10CFR50.73(a)(2)(i)(A), and 10CFR50.73(a)(2)(ii) which require a 30-day written report.

This report is number 91-002-00; Docket No. 50-457.

Very truly yours,

K. L. Kofron
Station Manager
Braidwood Nuclear Station

KLK/DN/clf
(226/ZD8FG)

Enclosure: Licensee Event Report No. 91-002-00

cc: NRC Region III Administrator
NRC Resident Inspector
INPO Record Center
CECo Distribution List

IE22

LICENSEE EVENT REPORT (LER)

Form Rev 2.0

Facility Name (1)	Docket Number (2)	Page (3)
Braidwood 2	0| 5| 0| 0| 0| 4| 5| 7	1 of 0 4

Title (4) Reactor Shutdown caused by Excessive Containment Purse Valve Leakage Discovered during Local Leakage Rate Testing

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)
0| 5	1| 0	9| 1	9| 1	0 | 0 | 2	0 | 0	0 | 6	0| 7	9| 1	None	0| 5| 0| 0| 0| |
										0| 5| 0| 0| 0| |

OPERATING MODE (9): 1

POWER LEVEL (10): 0 9 7

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

20.402(b)	20.405(c)	X 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)	X 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 AREA CODE	
T. Eliakis, System Test Engineer	Ext. 2994	8 | 1 | 5	4| 5| 8| -| 2| 8| 0| 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
X	V | A	I| S| V| *	J| 0| 1| 0	Yes					

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)

On May 10, 1991 local leakage rate test surveillance BwVS 6.1.7.3-1 was started by a System Test Engineer (STE) to verify that the measured leakage rate of the 48 inch containment purge supply containment penetration (P-97) was less than 23.15 Standard Cubic Feet per Hour (SCFH) when pressurized to at least 44.4 psig. P-97 consists of two containment isolation valves in series; 2VQ001A and 2VQ001B. A Leak Rate Monitor (LRM) test rig was attached to the P-97 test connection. The STE started P-97 pressurization by opening an air supply valve. At 2130, it was noted that the highest pressure achieved for P-97 using the LRM test rig was only 33 psig. The LRM was bypassed to directly pressurize P-97 to 46 psig. When the air supply was isolated, the pressure in P-97 decreased to 21 psig in 14 minutes and confirmed that valve leakage was higher than the Technical Specification limit. At 2200 on May 11, 1991 a unit shutdown commenced. Since this shutdown was required by Technical Specifications, an Unusual Event was declared. At 0322 on May 12, 1991 a reactor trip was caused by a spike on Source Range Channel N-31. At 0530 on May 13, 1991 the Unusual Event was terminated when the unit entered Mode 5 (Cold Shutdown). The cause of the seal leakage was attributed to seal degradation. The resilient material seals were replaced for both valves. BwVS 6.1.7.3-1 was satisfactorily performed with a leakage rate of 5.11 SCFH. There have been no previous occurrences.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) Year	Sequential Number	Revision Number	Page (3)
Braidwood 2	0 5 0 0 0 4 5 7	9 1	- 0 0 2 -	0 0	0 3 OF 0 4

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

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: Braidwood 2; Event Date: May 10, 1991; Event Time: 2215;

Mode: 1 - Power Operation; Rx Power: 97%;

RCS [AB] Temperature/Pressure: NOT/NOP

B. DESCRIPTION OF EVENT:

There were no systems or components inoperable at the beginning of the event which contributed to the severity of the event.

At 1334 on May 10, 1991 surveillance BwVS 6.1.7.3-1 "Primary Containment Type C Local Leakage Rate Tests of Containment Purge Supply Isolation Valves" was started by a System Test Engineer (STE) (non-licensed). The purpose of the surveillance is to verify that the measured leakage rate of the 48 inch containment purge (VQ) [VA] supply containment penetration (P-97) is less than 23.15 Standard Cubic Feet per Hour (SCFH) when pressurized to at least 44.4 psig. Testing is performed at least once every six months. P-97 consists of two containment isolation valves in series, 2VQ001A (located inside containment) and 2VQ001B (outside containment). In accordance with Technical Specification requirements, each valve was closed and deenergized.

A Leak Rate Monitor (LRM) test rig was attached to the P-97 test connection. The STE started P-97 pressurization at 1400 by opening an air supply valve. After expecting P-97 to be completely pressurized, the STE checked the LRM pressure indicator and observed a decrease in pressure. Initially, the STE felt that the LRM was defective and obtained a replacement. The spare LRM was connected and checked. The STE again observed a decrease in pressure. At this time, the STE was contacted for an unrelated incident. When the STE returned to monitor the LRM pressure, it had stabilized and was slowly increasing. Since P-97 is a large penetration, the STE expected that it would take some more time to pressurize.

At 2130, it was noted that the highest pressure achieved for P-97 using the LRM test rig was only 33 psig (expected pressure was 46 psig). To expedite pressurization, the LRM was bypassed and the STE was able to directly pressurize P-97 to 46 psig. When the air supply was isolated, the pressure in P-97 decreased to 21 psig in 14 minutes. The large pressure drop confirmed that valve leakage was higher than the Technical Specification limit.

At 2215, Limiting Condition for Operation Action Requirement (LCOAR) 2BwOS 6.1.1-1a was entered. The action required by Technical Specifications was to restore the P-97 containment isolation valves to an operable status within 24 hours. If P-97 could not be declared operable then the unit would need to be in Mode 3 (Hot Standby) in the next 6 hours, and in Mode 5 (Cold Shutdown) within the following 30 hours.

The appropriate NRC notification via the ENS phone system was made at 2247 pursuant to 10CFR50.72(b)(1)(ii).

Since the valves are tested simultaneously, it could not be determined which valve was leaking. Nuclear Work Request (NWR) A48651 was written to repair P-97. The ductwork downstream of valve 2VQ001B was removed. P-97 was pressurized and valve 2VQ001B showed signs of leakage. Repair efforts were started on valve 2VQ001B. The resilient material seal was cleaned in an effort to reduce leakage.

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

At 1100 on May 11, 1991 initial contact was made with NRC Region III personnel to discuss the feasibility of obtaining a temporary waiver of compliance to extend the allowed outage time for 2VQ001B from 24 hours to 72 hours in order to allow sufficient time for repairing the valve. The basis for the relief was to be the integrity of the valve inside containment (2VQ001A). This valve would be "snoop" checked to provide some level of confidence that valve leakage would be within 10CFR50 Appendix J limits.

At 1430 a decision was made to replace the resilient material seal for valve 2VQ001B. P-97 was pressurized and a containment entry was required to check for leakage past valve 2VQ001A. The results of this inspection revealed that this valve also exhibited signs of significant leakage. This condition was considered unacceptable and efforts to obtain temporary relief were terminated.

At 2200 a unit shutdown commenced. Since this shutdown was required by Technical Specifications, an Unusual Event was declared in accordance with the Braidwood Station Emergency Plan.

The appropriate Illinois State and local agencies notification via the Nuclear Accident Reporting System was made at 2205.

The appropriate NRC notification via the ENS phone system was made at 2210 pursuant to 10CFR50.72(a)(1)(i), 10CFR50.72(a)(3) and 10CFR50.72(b)(1)(i)(A).

At 0322 on May 12, 1991 a reactor trip occurred. Reactor power had been reduced to approximately 5E-11 amps on the intermediate range nuclear instrumentation. At this time, the source range detectors had just automatically reenergized and the unexpected trip was caused by a spike on Source Range Channel N-31. All control rods fully inserted and a Feedwater Isolation (FWI) signal was generated. With the reactor subcritical, the unit entered Mode 3 (Hot Shutdown) at the time of the reactor trip. At 0326 stable plant conditions were verified and the FWI signal was reset.

The appropriate NRC notification via the ENS phone system was made at 0553 pursuant to 10CFR50.72(b)(2)(ii).

At 1000, a plant cooldown was started. At 0530 on May 13, 1991 the Unusual Event was terminated when the unit entered Mode 5 (Cold Shutdown)

The appropriate NRC notification via the ENS phone system was made at 0543 pursuant to 10CFR50.72(c)(1)(iii).

This event is being reported pursuant to:

10CFR50.73(a)(2)(iv) - any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature, including the Reactor Protection System.

10CFR50.73(a)(2)(i)(A) - the completion of any nuclear plant shutdown required by the plant's Technical Specifications.

10CFR50.73(a)(2)(ii) - any event or condition during operation that results in the condition of the plant, including its principal safety barriers, being seriously degraded.

C. CAUSE OF EVENT:

The root cause of this event was component failure. The excessive leakrate for P-97 was caused by seal leakage from valves 2VQ001A and 2VQ001B. Th leakage was attributed to seal degradation.

The reactor trip was caused by a spike of the Source Range channel N-31. The channel trip setpoint (10E5 Counts per Second) was checked and found to be correct. The unit had been at power for 338 consecutive days and N-31 had been in a deenergized condition during this time. When reactor power decreased to 5E-11 amps on the intermediate range, the source range detector automatically energized. With the high voltage now present, a spike on N-31 occurred. The spike exceeded the setpoint and caused a reactor trip. After the spike, N-31 indication returned to normal (approximately 10E3 Counts per Second). The redundant source range channel N-32 energized approximately 2 seconds after the trip and responded normally throughout the shutdown.

D. SAFETY ANALYSIS:

This event had no effect on the safety of the plant or the public. The surveillance had been performed on Nov. 7, 1990 with leakage of 10.72 SCFH. This leakage rate value is then added to the overall containment leakage rate for all penetrations. The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure. As an added conservatism, the measured overall integrated leakage rate is further limited to account for possible degradation of the containment leakage barriers between leakage tests. The six months between surveillances is consistent with the requirements of Appendix J of 10CFR50 and based on an NRC initiative, Generic Issue B-20, "Containment Leakage Due to Seal Deterioration." The testing performed provided indication of resilient material seal degradation to allow for repair of the valves.

E. CORRECTIVE ACTIONS:

When P-97 was declared inoperable, a plant shutdown commenced to comply with LCOAR 6.1.7-1a. The shutdown was completed within the allowed time interval specified in the Technical Specifications.

The resilient material seals were replaced for both valves. BwVS 6.1.7.3-1 was satisfactorily performed with a leakage rate of 5.11 SCFH. The combined integrated leakage rate is currently 106.42 SCFH which is well below the limit of 463 SCFH.

F. PREVIOUS OCCURRENCES:

There have been no previous occurrences.

G. COMPONENT FAILURE DATA:

Manufacturer	Nomenclature	MFG Part Number
Nyles-Jamesbury	Tefzel Valve Seat	102-0077-7B

DEVIATION REPORT

DVR NO.
20 - 02 - 92 - 010
STA UNIT YEAR NO.

Form Rev 2.0

PART 1 | TITLE OF DEVIATION
2VQ001A/B Leakage in Excess of Surveillance Requirements.

OCCURRED 5-10-91 2215
DATE TIME

SYSTEM AFFECTED	PLANT STATUS AT TIME OF EVENT	WORK REQUEST NO.	TESTING
VQ	MODE 1 POWER(%) 97	A48651/A48652	[X] YES [ ] NO

DESCRIPTION OF EVENT

At 2215 while doing surveillance test of 48" CNMT Isol Valves the leakage of the 2VQ001A&B exceed the surv acceptance criteria. The action Statement C. for Tech Spec 3.6.1.7 was entered. Test Rig was not able to achieve a pressure greater than 33 psig. Normal is 46 PSIG. Pressurizing without Rig to 46 psig, observed pressure decrease to 21 psig in 14 minutes.

POTENTIALLY SIGNIFICANT EVENT PER NSD DIRECTIVE A-07 [ ] YES [X] NO

10CFR50.72 NRC RED PHONE NOTIFICATION MODE [X] 1 HOUR [ ] 4 HOUR 2247 TIME [ ] NO

Jerald D. Wagner 05/10/91
RESPONSIBLE SUPERVISOR DATE

PART 2 | OPERATING ENGINEER'S COMMENTS

Unit shutdown started 2200 5/11/91 when attempts to repair were unsuccessful. Unit went to Mode 5 where LCO was no longer applicable and will remain until repairs are effective.

[ ] NON REPORTABLE EVENT

[X] 30 DAY REPORTABLE/10CFR (a)(2)(ii)

[ ] 5 DAY REPORT PER 10CFR21

[ ] ANNUAL/SPECIAL REPORT REQUIRED

A.I.R. #

L.E.R. # 91-002

NOTIFICATION ______ REGION III ______ DATE ______ TIME

OFC of VP PWR OPS NSD ______ DATE ______ TIME

[ ] CECO CORPORATE NOTIFICATION MADE IF ABOVE NOTIFICATION IS PER 10CFR21

TELECOPY ______ CECO CORPORATE OFFICER ______ DATE ______ TIME

PRELIMINARY REPORT COMPLETED AND REVIEWED J. A. Chojnicki
OPERATING ENGINEER DATE

INVESTIGATION REPORT & RESOLUTION ACCEPTED BY STATION REVIEW
6-5-91 Dale Nelson 6-5-91
6-5-91

RESOLUTION APPROVED AND AUTHORIZED FOR DISTRIBUTION
STATION MANAGER 6/10/91 DATE

86-5176 (Form 15-52-1) 11-20-85

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