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

November 17, 1992
BW/92-0583

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

Dear Sir:

The enclosed voluntary Licensee Event Report from Braidwood Generating Station is being transmitted to you with the requirement of 10CFR50.73(a)(2)(iv), which requires a 30-day written report.

This report is number 92-012-00, Docket No. 50-456.

K. L. Kofron
Station Manager
Braidwood Station

KLK/AJS/dla
667ZD85G

Encl: Licensee Event Report No. 92-012-00

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

190043

IE22 1/1

LICENSEE EVENT REPORT (LER)

Facility Name (1)	Docket Number (2)	Page (3)
Braidwood 1	0 5 0 0 0 4 5 6	1 of 0 3

Title (4)

1A Auxiliary Feedwater Pump Start Due to Low-2 Steam Generator Level As A Result of Leakage Past 1FW079A

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)
									None	0 5 0 0 0
1 0	1 8	9 2	9 2	0 1 2	0 0	1 1	1 3	9 2		0 5 0 0 0

OPERATING MODE (9)	5	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)			
POWER LEVEL (10)	0 0 0	___ 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)	___ 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)	___ 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
K. Boyle, Operating Ext. 2202	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

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 October 18, 1992 at approximately 1000 hours, Steam Generator water levels on Unit 1 were raised to approximately 45%. This is slightly above the 40.8% Steam Generator LO-2 setpoint. Concurrent with these activities, maintenance was being performed on the feedwater isolation valves. Earlier the same day, the 1FW009B had been stroked with no problems encountered. When the 1FW009A was opened at 1557 CDST, a rapid decrease in 1A Steam Generator level occurred. The 1A Auxiliary Feedwater (AF) pump automatically started. The 1A Auxiliary Feedwater (AF) pump was then placed in pull-to-lock, which stopped the pump, and valve 1FW009A was manually closed. After approximately fifteen minutes the 1A AF pump was restarted and all S/G levels increased to approximately 50%. The root cause of the event was the leakby of a feedwater check valve. Leakage past check valve 1FW079A allowed upstream feedwater piping to be filled from the steam generator, which resulted in 1A Steam Generator level lowering past the Lo-2 setpoint, and subsequent auto-starting of the 1A AF Pump. Since the feedwater check valve was being utilized for an isolation boundary at a pressure significantly lower than it is normally used, it was determined that no additional maintenance was necessary to this valve. There have been previous occurrences of Auxiliary Feedwater Pump auto-starts due to Lo-2 S/G levels. The previous root causes and corrective actions are not applicable to this event.

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: Braidwood 1; Event Date: October 18, 1992; Event Time: 1557 CDST
Mode: 5 - Cold Shutdown; Rx Power: 000%;
RCS [AB] Temperature/Pressure: 98 Degrees F / 0 psig;

B. DESCRIPTION OF EVENT:

There were no systems or components that were degraded at the beginning of this event that contributed to the severity of the event.

On October 18, 1992 at approximately 1000 hours, Steam Generator water levels on Unit 1 were raised to approximately 45%. This is slightly above the 40.8% Steam Generator LO-2 setpoint, and was decided upon to support diesel generator testing and to subsequently provide a "flush" for chemistry. The levels were maintained relatively low because the S/G's were to be drained and refilled per chemistry following the Diesel Generator testing. Since draining is a time consuming process, it was decided to only put as much water as necessary into the Steam Generators to support the diesel test so as to accelerate the subsequent "flush".

Concurrent with these activities, maintenance was being performed on the feedwater isolation valves. Earlier the same day, the 1FW009B had been stroked with no problems encountered. When the Electrical Maintenance (EM)technician [non-licensed] requested the 1FW009A be stroked open, the Unit 1 Nuclear Station Operator (NSO) [licensed operator] and Unit 1 Unit Supervisor [licensed operator] discussed the evolutions, and decided that since the D/G lineup had been signed off and the previous shift had no problems, the 1A pump would be left in "after-trip". When the 1FW009A was opened at 1557 CDST, the Unit NSO monitored the 1A S/G level. When he saw a rapid decrease, he told an extra NSO [licensed operator] to place the 1A AF PP in pull-to-lock (PTL). As the extra NSO's hand reached for the control switch, the 1A Auxiliary Feedwater pump started. The pump was placed in PTL, which stopped the pump, and the 1FW009A was manually closed. After approximately fifteen minutes the 1A AF pump was restarted and all S/G levels increased to approximately 50%. The appropriate Emergency Notification System (ENS) notification was made at 1707 CDST pursuant to 10CFR50.72(b)(2)(ii).

This event is being reported pursuant to the requirements of 10CFR50.73(a)(2)(iv), which requires the reporting of actuations of any Engineered Safety Feature (ESF).

C. CAUSE OF THE EVENT:

The root cause of the event was the leakby of a feedwater check valve. Leakage past check valve 1FW079A allowed upstream feedwater piping to be filled from the steam generator, which resulted in 1A Steam Generator level lowering past the Lo-2 setpoint, and subsequent auto-starting of the 1A AF Pump.

D. SAFETY ANALYSIS:

This event had no effect on plant or public safety since the engineered safety feature operated as designed.

Under worst case conditions of the 1A AF pump failing with a Lo Lo steam generator level present, would have no effect since the RHR system was providing core cooling. The Auxiliary Feedwater System is not required to be operable in Mode 5. Therefore, no impact on plant safety would result.

E. CORRECTIVE ACTIONS:

The 1A AF pump control switch was placed in PTL, which stopped the pump, and valve 1FW009A closed. After approximately fifteen minutes the 1A AF pump was restarted and all S/G levels were increased to approximately 50% to provide a larger margin to the Lo-2 setpoint. Since the feedwater check valve was being utilized for an isolation boundary at a pressure significantly lower then it is normally used, it was determined that no additional maintenance was necessary to this valve.

F. PREVIOUS OCCURRENCES:

There have been previous occurrences of Auxiliary Feedwater Pump auto-starts due to Lo-2 S/G levels. The previous root causes and corrective actions are not applicable to this event.

G. COMPONENT FAILURE DATA:

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
NUCLEAR VALVE DIVISION	Check Valve	FW079A	465QBB1-00