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

December 29, 1989
BW/89-3230

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)(i) which requires a 30-day written report.

This report is number 89-016-00; Docket No. 50-456.

Very truly yours,

R. E. Querio

R. E. Querio
Station Manager
Braidwood Nuclear Station

REQ/JDW/jfe
(7126z)

Enclosure: Licensee Event Report No. 89-016-00

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

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LICENSEE EVENT REPORT (LER)

Form Rev 2.0

Facility Name (1)	Docket Number (2)	Page (3)
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Title (4) Residual Heat Removal Pump Suction Relief Valve Premature Actuation and Failure to Reseat Due to Deficient Work Practices and Personnel Error.

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 2	0 1	8 9	8 9	0 1 6	0 0	1 2	2 9	8 9		0 5 0 0 0

OPERATING MODE (9): 5

POWER LEVEL (10): 0 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	
	AREA CODE	
Howard James, Technical Staff Engineer Ext. 2482	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	B P	R V	C 7 1 0	YES					
A	B P	R V	C 7 1 0	YES					

SUPPLEMENTAL REPORT EXPECTED (14)

__ Yes (If yes, complete EXPECTED SUBMISSION DATE) X NO

Expected Submission Date (15): Month | Day | Year

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

At 0142 on December 1, 1989 while drawing a bubble in the Pressurizer (PZR), Reactor Coolant System (RCS) pressure slowly increased from 375 psig to 404 psig. At this time the 1B Residual Heat Removal (RH) pump suction relief valve, which had a setpoint of 450 psig, actuated and remained open. Charging flow was increased but PZR level indicated 0% by 0151. Reactor Operators concluded that an RH pump suction relief valve had lifted because Hold Up Tank levels were increasing rapidly. The operating train of RH, 1A, was isolated at 0155. At 0200 RCS pressure reached 272 psig and stabilized. RCS level was at the lower portion of the PZR surge line and flow into the RCS was equal to the flow exiting the RCS. At 0215 the Licensed supervisors decided to return the second Charging pump to service per 10CFR50.54(x). At 0235 the second charging pump was started. At 0245 PZR level was above 0%. At 0319 field reports identified that the 1B RH pump suction relief had actuated. At 0350 the B RH Train was isolated which terminated the event. Approximately 64,000 gallons had relieved through the valve. The cause of the early lift was dirt between the valve spindle and guide sleeve which affected valve lift setpoint adjustment. The cause for the valve remaining open was an incorrect nozzle ring setting due to a personnel error. Maintenance procedures will be reviewed. Training will be conducted. No previous occurrences.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) Year		Sequential Number		Revision Number	Page (3)
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TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: Braidwood 1; Event Date: December 1, 1989; Event Time: 0142;

Mode: 5 - Cold Shutdown; Rx Power: 0%;

RCS [AB] Temperature/Pressure: 170 degrees F/350 psig

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.

The Pressurizer (PZR) was water solid. The B and D Reactor Coolant Pumps were in operation and both trains of Residual Heat Removal (RHR) [BP] were aligned to the RCS for Shutdown Cooling with the A loop in operation. Letdown to the Chemical Volume and Control System (CV) [CB] was being supplied from the 1A RHR system.

At 0055 on December 1, 1989 in preparation for drawing a bubble in the PZR, the Nuclear Station Operator (NSO) (Licensed Reactor Operator) increased letdown flow from 97 to 110 gpm and energized all 3 PRZ Backup Heaters. This was in accordance with Operating Procedure BwOP RY-5, Drawing a Bubble in the Pressurizer. PZR pressure and temperature were approximately 350 psig and 404 degrees F at the time.

At 0126 the NSO observed that RCS pressure had risen to 395 psig. To stabilize RCS pressure the NSO performed the following:

1. Increased Letdown flow.
2. Decreased CV charging flow to minimum.
3. Deenergized 2 of the PZR Backup Heaters.

By 0139 the NSO had increased letdown flow to the maximum value, however, RCS pressure was still slowly increasing.

At 0142 with the RCS pressure at 404 psig, the 1B RHR Pump suction relief valve actuated. This corresponded to approximately 410 psig at the elevation of the 1B RHR Pump suction relief valve. This was 46 psig below its setpoint of 450 psig +/- 1%.

At 0144 the NSO observed that the PZR level had returned to scale from an off scale high condition. The level was decreasing at a rapid rate. The NSO began reducing letdown flow in an attempt to stabilize PZR level.

At 0149 the NSO increased charging in an attempt to maintain pressurizer level. The following actions were also taken at approximately the same time:

The Shift Engineer (SE) (Senior Reactor Operator) (SRO) and the Shift Foreman(SF)(SRO) were notified and immediately reported to the Control Room. The SF assumed the SRO direct supervisory responsibilities upon being briefed of the situation by the Station Control Room Engineer(SCRE) (SRO) and the NSO.

The NSOs monitored control room indicators that would be indicative of a leak inside the containment. There was no indication of leakage inside the containment.

The NSO directed the available Equipment Attendants (EA) (non-licensed operators) to check the areas of the Auxiliary Building for signs of Unit 1 CV or RHR system leakage.

The NSO contacted the Radwaste Control Room Operator (RWCRO) (non-licensed operator) and requested a check of Radwaste indications for signs of an increase in the input to Radwaste. The RWCRO reported that both Recycle Hold Up Tanks (HUT) [GA] were increasing at a rapid rate.

Based on this information, the Control Room Personnel concluded that one or both of the RHR Pump suction relief valves had lifted and not reset.

At 0151 the NSO observed that PZR level indicated 0%.

At 0153 the NSO increased CV charging to maximum for the 1A CV Pump and isolated letdown flow. The NSOs and the SF decided to isolate the 1A RHR Train in an attempt to isolate the leak. This was based on the assumption that a system perturbation would most probably have occurred in the operating loop.

At 0155 the NSO shutdown and isolated the 1A RHR train and placed the 1B RHR train in operation. The NSO opened the CV pump suction isolation valve from the Refueling Water Storage Tank (RWST) [BR/BQ] and closed the outlet valves from the Volume Control Tank (VCT).

At 0200 the NSO that had been monitoring the RCS pressure reduction since the start of the event, observed that the No. 1 seal differential pressure for the 1B Reactor Coolant Pump (RCP) was approaching the minimum permissible value of 200 psid. The NSO shutdown the 1B RCP at this time. Shortly after the pump was shut down RCS Pressure reached 272 psig. The NSOs and SF believed that the increase in the RCS pressure reduction rate that occurred about this time was the result of the RCP shutdown. The NSOs and SF observed that the No 1 Seal differential pressure for the 1D RCP was above 200 psid. They decided to keep the 1D RCP in operation as long as the No. 1 Seal differential pressure was 200 psid or greater. They believed that this would prevent another increase in RCS pressure reduction rate. It would also provide PZR spray flow capability when PZR level was recovered.

The RCS pressure stabilized at 272 psig. This was the lowest pressure achieved in the RCS during the event. Flow into the RCS from the 1A CV pump was equal to the amount of flow exiting the RCS. The recovery portion of the event started at this point.

At approximately 0202, an NSO directed an EA who had been checking for signs of leakage in the Auxiliary Building to isolate the 1RH8734A, Manual Isolation Valve from the Discharge of the 1A RH Pump to the CV Letdown Header. This valve was located in the Containment Penetration area of the Auxiliary Building on the 364' elevation.

At approximately 0205 the EA isolated the 1RH8734A. The EA identified a puddle on the floor and water dripping down from a cable pan. The EA determined that this water came from a relief valve located above the cable pan that was dripping a small amount of water at this time. The EA was not aware of the valve's actual identity or function. The EA conveyed the information to the Unit 2 SF who was on the 364' elevation outside the penetration area. The Unit 2 SF had been assisting with the in plant leakage search for Unit 1. The Unit 2 SF conveyed this information to the Control Room. During the information transfer from the EA to the Unit 2 SF, and the Unit 2 SF to the NSOs and SF in the control room, this relief valve was perceived to be the 1RH8708A, the 1A RHR Pump suction relief valve. This perception created the mind set that the 1RH8708A was the valve that had actuated.

At 0215 the SRO Licensed Shift Supervisors, the SF and SE, decided to return the 1B CV pump to service. This decision was made pursuant to the requirements of 10CFR50.54(x). The Technical Specifications specify that a maximum of 1 Centrifugal Charging Pump shall be operable in Mode 5. The SRO Supervisors viewed this as a necessary action to protect the health and safety of the public. The 1B CV pump was readily available by racking in its breaker. The PRZ level had been indicating 0% for 24 minutes. There were no other pumps readily available to perform the required function.

At 0227 an Alert was was classified as per Emergency Action Level 2.n. of Braidwood Emergency/Implementing Procedures (BwZP), "Primary System Leakage is beyond the makeup capabilities of available charging pumps".

At 0235 the NSO isolated the 1 RH8701B, the second of the in-series RHR suction isolation valves for the 1A RHR pump. This was performed due to suspected leakage of the downstream isolation valve, 1RH8701A. This suspicion was generated by the mindset created by the misidentification of the leaking relief valve as the 1RH8708A combined with reports from the RWCRO that the rate of increase in the HUTs was decreasing. The NSO started 1B CV Pump at this time.

At 0237 the Nuclear Accident Reporting System(NARS) notification was made to Illinois State agencies to declare the Alert.

At 0245 the NSO observed that the PZR Cold Calibrated (Cold Cal) Level Channel had increased above 0%. The NSO shutdown the 1B CV Pump. This was due to a concern about thermal shock to the PZR. RCS pressure was 310 psig. The RWCRO reported that HUT levels were still increasing.

At 0254 the NSO restarted the 1B CV pump. This was due to the PZR Cold Cal Level decreasing below 0% and the RCS pressure decreasing to 301 psig.

At 0302 the NSO observed that PZR Cold Cal Level indication was above 0%. The NSO reduced CV pump flow by throttling the 1CV121, CV Charging Header Flow Control Valve.

At 0312 an NSO instructed an EA who was on the 364' elevation, to proceed to the 1RH8708B, 1B RHR Pump Suction Relief Valve, and determine if it had actuated.

At 0319 the EA and the Unit 2 SF identified that the 1RH8708A did not appear to be open based on the absence of noise at the valve. They also identified that the 1RH8708B was open. This was based on the loud screaming noise that could be heard.

At approximately 0322 the NSO directed an EA,who had been dispatched to the 364' area, to reopen and then close the 1RH8734A. This was performed to provide a flowpath to depressurize the isolated 1A RH train. Control Room personnel believed that this would ensure that the 1RH8708A had reseated.

At 0326 the appropriate ENS notification was made pursuant to 10CFR50.72(a)(i) and 50.72(b)(1)(i)(B).

At 0335 the Unit 2 SF identified the relief valve that was spraying water as the 0AB8634, a relief valve connected to the common header where the RHR suction relief valves discharged.

At 0345 an EA was stationed by the 1RH8708A to observe for signs of valve actuation when the 1A RHR train was realigned to the RCS.

At 0346 the NSO opened the 1A RHR train suction isolation valves, 1RH8701A and B.

At 0348 the EA verified that the 1RH8708A had not actuated.

At 0349 the NSO placed the 1A RHR train in operation. The EA by the suction relief valve verified that it was not lifting.

At 0350 the NSO shut down and isolated the 1B RHR Train. The NSO observed that PRZ Level had started to increase significantly.

At 0353 the NSO shutdown the 1B CV Pump and stable plant conditions were established.

At 0427 SE transferred Command and Control to the Technical Support Center.

At 0435 the event was terminated.

At 0445 the appropriate NARS notification was made to terminate the event.

At 0512 the appropriate NRC notification via the ENS phone system was made pursuant to 10CFR50.72(c)(1)(iii) to terminate the event.

A review of the data collected during the event has identified the following:

1. The total water volume passed through the 1B RHR Pump suction relief valve was approximately 64,000 gallons.

2. The total volume pumped to the RCS from the RWST was approximately 54,000 gallons.

3. The relief valve remained open approximately 111 minutes.

4. The lowest level the system attained is believed to be the lower portion of the Pressurizer Surge Line, Reactor Vessel Level indicated 100% throughout the entire event.

5. The PZR Liquid Space and Surge Line Temperature Indicators experienced an indicated cool down in excess of 200 degrees F in a one hour period. This occurred when PZR level was re-established during the event. The PZR Surge Line temperature indicator decreased from 374 degrees F at 0254 to 164 degrees F at 0354. The PZR Liquid Space temperature indicator decreased from 444 degrees F at 0143 to 174 degrees F at 0243. These were normally expected occurrences for an event of this nature. PZR temperature was approximately 440 degrees F and RCS temperature was approximately 170 degrees F at the start of the event.

On December 02, 1989 the Onsite Review of the engineering evaluation to determine the effects of the cooldown on the PZR was completed. The evaluation concluded that the structural integrity of the PZR was acceptable for continued operation.

This event was evaluated by an NRC Augmented Inspection Team and an event investigation team from INPO.

This event is being reported pursuant to 10CFR50.73(a)(2)(i):

- Any operation or condition prohibited by the plants Technical Specifications.
- Any deviation from the plants Technical Specifications authorized pursuant to 10CFR50.54(x).

C. Cause of Event:

The root cause of this event was the premature actuation of the relief valve combined with an incorrectly set nozzle ring.

The 1RH8708B valve was systematically removed, tested, and disassembled for root cause analysis to determine failure mode. A representative of the Manufacturer, Crosby Valve and Gauge Company, participated in the disassembly and analysis at the request of Braidwood Station. A summary of the investigation results is detailed below.

The Valve was bench tested upon removal. The valve lifted at 411 psig, 407 psig and 405 psig on three successive lift tests.

The nozzle ring setting was checked. The nozzle ring was found to be out of adjustment high by 233 notches. With the Nozzle ring at this setting the valve would not reseat until pressure was significantly below the lift setpoint. This mis-setting occurred when the Nozzle ring was set during maintenance prior to installation in April of 1988. The cause of this mis-setting was a failure to follow procedure. The Mechanical Maintenance personnel (MM) (non-licensed maintenance personnel) who adjusted the nozzle ring set the ring 110 notches from the "upper locked" position instead of 343 notches as required by procedure BwMP 3305-072, Disassembly-Reconditioning-Reassembly of Crosby Style JB-TD-WR ("L" Orifice) Relief Valve. The procedure requires the nozzle ring to be taken from the "as found" position to the upper locked position during disassembly. The number of notches are counted and document in step F.2.c.1). This step was performed and the number of notches was documented as 343. During reassembly the MM was directed to take the ring to the upper locked position and lower it the number of notches recorded in step F.2.c.1). For this event the MM took the final position setting from a Station Traveler which provided the Manufacturers setting of 110 notches below "zero position". Zero position was not the upper locked position but the position where the nozzle ring was level with the valve seat. This caused the "as left" nozzle ring setting to be 110 notches below the upper locked position instead of 110 notches below the zero position which was 233 notches lower.

The nozzle ring setting error was corrected and the valve was bench tested to determine if the improper setting had affected the lift setpoint. The valve lifted at 400 psi, essentially the same lift value as the incorrect setting.

The valve lift setpoint was systematically adjusted with bench tests performed after each adjustment. The valve responded in the correct manner after each adjustment for both increasing and decreasing setpoint adjustments.

The valve was disassembled, decontaminated and inspected. Grooves were found on both the spindle and the spindle guide. It is suspected that a piece of sand grit or metal grinding became lodged between the spindle and the spindle guide during maintenance performed on the valve in April of 1988. As the adjusting bolt was turned to set the lift setpoint the spindle rotated. With the foreign material in the gap between the spindle and spindle guide the adjusting process made the groove circumferential. During a valve lift, presumably the lift setpoint verification prior to installation, the foreign material became dislodged. The vendor representative stated that the increased friction from a piece of foreign material between the spindle and guide would account for the 45 psi disparity between the as left setpoint value from the maintenance test performed in April of 1988 and the actual valve lift setpoint experienced on December 1,1989.

D. Safety Analysis:

This event had no effect on the Safety of the Plant or the public. After the initial level decrease, the water level appears to have remained in the lower portion of the surge line until it was recovered. The reactor had been in an extended refuel outage. A significant potential for a temperature increase from decay heat did not exist.

Under the worst case condition of a failed open RH relief valve occurring during initial system alignment after extended Unit Operation there would still be no effect as this is enveloped in Section 5.4 of the Updated Final Safety Analysis Report. The design of the RH system provides for complete isolation of an RH loop using redundant isolation valves. Adequate leakage dection capability is provided by Radwaste input instrumentation. And physical train separation provides the capability to operate indefinitely with one train isolated.

E. Corrective Actions:

Immediate corrective actions:

1. Letdown was isolated and charging was increased to maximum.

2. The RH trains were isolated in a systematic manner to stop the leakage.

3. Stable plant conditions were established.

4. The faulty valve was removed and replaced with one from stores.

5. The area that was sprayed with water from the the AB relief valve was cleaned up and decontaminated.

6. An Engineering evaluation was conducted on the effects of the indicated cooldown rates on the PZR. Based on the results of this evaluation it has been concluded that the effects were insignificant and continued operation is acceptable.

7. The documentation for valve lift and reset settings for the the 1A , 2A, and 2B RHR Pump suction relief valves has been reviewed. Base on this review, it has been concluded that these three relief valves are set correctly.

8. The valve that was removed from the 1B RH pump suction line was rebuilt, tested and installed on the 1A RH pump suction line. This was performed to facilitate bench testing of the 1A RH pump suction relief valve. This was done in response to a Recommendation from the NRC Augmented Inspection Team. The nozzle ring setting was correct. The valve actuated at 465 psig on the first test, 5 psi above the acceptance criteria of 455 psig +/- 1%. The valve setting was adjusted and the valve was returned to stores.

Additional Corrective Actions:

1. Maintenance procedures for Crosby relief valves will be reviewed to ensure that instructions are clear and adequate. This will be tracked to completion by action item 456-200-89-19301.

2. This event will be covered with Licensed personnel as part of the requalification program. This action will be tracked to completion by action item 456-200-89-19302.

3. This event will be covered with appropriate Maintenance personnel as part of a Training Tailgate session. This session will stress the following:

   a. Proper verification of reference point when adjusting blowdown rings on relief valves.

   b. Ensuring a clean environment is maintained during assembly/disassembly of components.

   c. Proper techniques for determination of lift setpoints for relief valves.

   This action will be tracked to completion by action item 456-200-89-19303.

4. Braidwood Station will request that the Westinghouse Owners Group develop a Mode 4/5 LOCA Procedure citing this event as an example. This action will be tracked to completion by action item 456-200-89-19304.

F. Previous Occurrences:

There have been no similar previous occurrences.

G. Component Failure Data:

Manufacturer	Nomenclature	Model Number
Crosby Valve and Gauge Co	Relief valve	JB-35-TD-WR