U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-456/89030

Docket No. 50-456 License No. NPF-72

Licensee: Commonwealth Edison Company
Post Office Box 767
Chicago, IL 60690

Facility Name: Braidwood Station, Unit 1

Inspection At: Braidwood Site, Braidwood, Illinois

Inspection Conducted: December 2-4, 1989

Team Members: J. Hinds, Team Leader, DRP, RIII
T. Tongue, Senior Resident Inspector, Braidwood
J. Hopkins, Operator Licensing Examiner, DRS, RIII
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Approved By: W. D. Shafer, Chief
Reactor Projects Branch 1

DEC 22 1989
Date

Inspection Summary

Inspection on December 2-4, 1989 (Report No. 50-456/89030(DRP)

Areas Inspected: Special Augmented Inspection Team (AIT) inspection conducted in response to the Unit 1, Residual Heat Removal System, Train B, suction relief valve event of December 1, 1989. The review included validation of the sequence of events, evaluation of the licensee's response and notifications, determination of the root cause for the opening of the suction relief valve, validation of the flow paths and volumes of the water discharged, review of the operator's performance, and evaluation of the adequacy of the operating procedures.

Results: No violations or deviations were identified; however, the licensee has committed to conducting a Preliminary Significant Event evaluation and will make the results of that evaluation available to the NRC. In addition, the licensee has committed to removing the 1A RHR train suction relief valve, performing bench lift tests, and reporting the results as soon as possible. Information received by the AIT subsequent to the exit meeting on December 4, 1989, will be included in this report as it becomes available.

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DETAILS

I. Introduction

A. Synopsis of Event

On December 1, 1989, Braidwood Unit 1 was in cold shutdown and in the process of taking steps to return the unit to operation following its first refueling outage. At about 1:30 a.m., the plant was in the process of drawing a bubble in the PZR. Reactor coolant temperature was about 175°F and pressure was about 350 psig with two reactor coolant pumps and one train of residual heat removal (RHR) in operation. At about the time that letdown flow was increased to allow the PZR bubble to form, the 1B RHR pump suction relief valve lifted, allowing the PZR to rapidly drain to the radwaste system. This situation could have caused reactor vessel voiding and possible eventual uncovering of the core. The situation was promptly recognized by operations personnel who responded by concurrently providing additional charging water and selectively isolating the RHR system to secure the loss of reactor coolant inventory. The licensee declared the appropriate GSEP "ALERT" and notified the State and NRC as required. The source of the loss of inventory was identified and isolated by 3:50 a.m.; recovery was completed by 4:15 a.m. and the ALERT was secured at 4:35 a.m. Followup evaluations by the licensee and the NRC revealed that the 1B RHR pump suction relief valve setting was found to be about 411 psig vice 450 psig as designed which resulted in the unplanned opening when the RCS pressure reached an indicated 416 psig.

B. AIT Formation

On December 1, 1989, based on the review of the information available, as related to the RHR suction relief event at Braidwood Unit 1, by Senior NRC managers, both in Headquarters and in Region III, it was determined that sufficient basis existed, due to significant system interactions, the complexity of the event, an unknown probable cause, the difficulty in understanding the evolution, and the questions pertaining to operator performance and procedure adequacy, to form an Augmented Inspection Team (AIT). The selections included J. M. Hinds, Chief, Reactor Projects Section 1A, Team Leader, T. M. Tongue, Senior Resident Inspector, Braidwood, J. A. Hopkins, Operator Licensing Examiner, S. P. Sands, Licensing Project Manager, and S. Israel, Reactor Operations Analysis Branch. The team selection was based on providing the appropriate expertise to adequately address the areas of concern identified in the AIT Charter, Attachment 1, in the areas of plant operations, equipment damage, and restart preparation activities.

C. AIT Charter

On December 1, 1989, Region III formulated and provided to the AIT a charter for implementation. The approved charter was provided to the Team Leader for distribution to the Team members at the site. The AIT Charter is Attachment 1 to this report. The general areas to be investigated were:

- Develop and validate the sequence of events.
- Evaluate the adequacy of the licensee's response.
- Root cause of the suction relief valve failure.
- Determine and validate flow paths and volumes.
- Assess operator performance and procedure adequacy.

D. Persons Contacted

Commonwealth Edison Company (CECo)

*R. E. Querio, Station Manager, Braidwood
*K. L. Kofron, Production Superintendent, Braidwood
*D. E. O'Brien, Technical Superintendent, Braidwood
*D. E. Cooper, Technical Staff Supervisor, Braidwood
*G. R. Masters, Assistant Superintendent, Operations, Braidwood
*M. E. Lohmann, Assistant Superintendent, Maintenance, Braidwood
*D. Miller, Regulatory Assurance Supervisor, Braidwood
J. Chojnicki, Shift Engineer, Braidwood
*P. G. Holland, Regulatory Assurance, Braidwood
*J. D. Wagner, Regulatory Assurance, Braidwood
P. A. Lau, Regulatory Assurance, Braidwood
B. Goering, INPO Coordinator, Braidwood
*L. W. Raney, Nuclear Safety, Braidwood
*S. T. Shields, Quality Assurance, Braidwood
*L. D. Guthrie, Master Mechanic, Braidwood
*J. Giuffre, Mechanical Maintenance Senior Work Analyst, Braidwood
*P. Smith, Operating Engineer, Unit 1, Braidwood

*Denotes those attending the exit interview conducted on December 4.

In addition, other members of the Braidwood staff were contacted by the AIT members.

II. Description - Unit 1, Train B, RHR Suction Relief Valve Inadvertently Opens and Fails to Close on December 1, 1989

A. Narrative Description

On December 1, 1989, between 1:42 a.m. and 3:50 a.m., Braidwood Unit 1 experienced a loss of reactor coolant inventory beyond the capability of the operating makeup system. This resulted from a premature opening and excessive blowdown of a relief valve on the suction side of an RHR pump. At the time, the unit was in cold shutdown and drawing a bubble in the PZR as part of the activities associated with returning the unit to operation following a refueling outage.

This resulted in about 67-68 thousand gallons of reactor coolant and makeup water being released to the radwaste holdup tanks. There is no evidence of an unplanned release to the environment. In addition, evidence indicated that there was no voiding in the reactor vessel or reactor coolant system. Recovery and corrective actions were such that there was little or no impact on the startup activities.

B. Sequence of Events

On December 1, 1989, Braidwood Unit 1 was in cold shutdown (Mode 5) and was in the final stages of the first refueling outage. The reactor was being prepared to enter Mode 4 (hot shutdown) by performing required surveillances and procedural check off steps in order to return to operation. At midnight, the reactor coolant system (RCS) average temperature was 170°F and pressure was about 350 psig. Two reactor coolant pumps (RCPs) were running and the 1A train of Residual Heat Removal (RH) was in operation in the shutdown cooling mode providing letdown with the 1B RHR train idle and available. The 1A charging (CV) pump was in normal operation, taking its suction from the volume control tank (VCT). The PZR power operated relief valves (PORVs) were in the "cold over pressure protection" mode, as well as the 1B CV pump and both safety injection (SI) pumps, tagged out of service (OOS) with power supplies removed as required by Technical Specifications (TS) and appropriate procedures. The 1A RHR pump suction valve 1RH 8701B was also tagged OOS open for protection against a pressure switch failure.

At about 12:55 a.m., the operators commenced drawing a bubble in the PZR by increasing letdown flow and energizing the PZR heaters. This was done by approved procedure BwOP RY-5, "Drawing a Bubble in the Pressurizer."

RCS pressure showed an increase to about 395 psig, at about 1:28 a.m., and was compensated for by increasing letdown flow.

RCS pressure continued to increase to about 404 psig on the wide range pressure instrument and the operator maximized letdown flow and minimized charging flow to about 70 gpm to compensate for the change. This was recorded in the logs as 1:30 a.m.; however, computer records indicated the actual time was 1:42 a.m. This is apparently, where at least, the 1B RHR pump suction relief valve opened, but was unconfirmed at that time. Logs showed that the operators assumed that one or both of these relief valves opened based on their training and industry experience.

Two minutes later, PZR level indicated on-scale from off-scale high and was decreasing rapidly. Letdown flow was promptly reduced in an effort to stabilize the PZR level. A report was received from the radwaste operator, at 1:45 a.m., that the holdup tanks (HUTs) were showing a significant level increase. At 1:49 a.m., charging flow was increased to help compensate for the PZR level drop and CV pump suction was manually transferred from the VCT to the reactor water storage tank (RWST) in anticipation of the need for additional makeup water.

At 1:52 a.m., the PZR level went to off-scale low and at 1:53 a.m., charging flow was increased to maximum and letdown flow was decreased to minimum.

In an effort to locate and isolate the loss of RCS coolant, at 1:55 a.m., the operators placed the 1B RHR train in service by starting the 1B RHR pump. They then secured the 1A RHR train by stopping the 1A RHR pump and commencing to isolate the 1A RHR train. This technique was used based on field reports of a problem in the vicinity of the 1A RHR pump suction relief valve and an accepted engineering practice to assume a fault in the operating train. It was later determined that the problem in the vicinity of the 1A RHR pump suction relief valve was associated with another nearby relief valve.

In their response, the operators secured the 1B RCP at 1:59 a.m., when primary system pressure dropped to less than 325 psig and the shaft seal indicated the lowest differential pressure. RCS pressure dropped to its lowest value at this point and was later determined to have been 272 psig by computer data. RCP 1D would continue to operate throughout the event to provide PZR spray and flow through core and idle loops.

The 1B CV pump was returned to service and placed in operation at 2:15 a.m., as an additional source of charging flow. This also resulted in an associated RCS pressure increase.

At 2:27 a.m., the licensee declared a GSEP "ALERT" based on a loss of coolant inventory beyond the capability of the makeup system.

At 2:35 a.m., the 1A RHR pump suction isolation valve was returned to service (OOS lifted) and shut, which completed isolation of the 1A RHR train and the suspected leak location.

The State of Illinois was notified of the "ALERT" at 2:37 a.m., via the Nuclear Accident Reporting System (NARS).

By 2:45 a.m., the PZR level was reported as increasing on the cold calibration channel LI462 and there was a corresponding RCS pressure increase to 310 psig. With that information in mind, the operators secured the 1B CV pump; however, the radwaste operator reported that the HUT levels were still increasing.

At 2:54 a.m., the PZR level was observed to be decreasing again; therefore, the 1B CV pump was restarted. This resulted in PZR level increasing, which was observed at 3:02 a.m. The charging flow was reduced to slow the rate of increase of the PZR level which was also done to minimize thermal shock to the PZR.

A report from an operator in the auxiliary building at 3:1[illegible] a.m., indicated evidence of flow through the 1B RHR pump suction relief valve due to noise level and pipe temperatures (warm to the touch).

The 1A RHR train was depressurized in an effort to assure that the 1A RHR pump suction relief valve was seated. This was done by opening the associated letdown valve, 1RH 8734A, at 3:22 a.m.

At 3:24 a.m., the resident inspector was contacted and informed of the event.

At 3:26 a.m., the NRC Headquarters duty officer was notified via the ENS phone.

The Unit 1 shift foreman reported water leakage from relief Valve 0AB-8634. This valve has its inlet supply common to the RHR pump suction relief valves, which leads to the HUTs. The leakage was later determined to be from a weep hole in the side of that valve and was the source of an estimated 30-50 gallons of water released to a limited area of the auxiliary building during the event.

At 3:42 a.m., charging flow was increased to adjust and maintain PZR level.

At 3:45 a.m., an operator was stationed in the vicinity of the 1A RHR pump suction relief valve to monitor for any evidence of flow through the valve. The 1A RHR train isolation valves were opened at 3:46 a.m., and the operator reported no evidence of flow through the pump suction relief valve.

Between 3:49 a.m. and 3:50 a.m., the 1A RHR train was placed in operation by starting the pump and the 1B RHR train was secured by stopping the pump and shutting the isolation valves. This resulted in a significant increase in PZR level identified at 3:52 a.m., and operators responded by securing the 1B CV pump about one minute later.

The operator in the field reported no evidence of flow or leakage through the 1A and 1B RHR pump suction relief valves at 3:54 a.m. and 3:56 a.m., respectively.

At 4:00 a.m., letdown was recommenced from the 1A RHR loop and at 4:02 a.m., the radwaste operator reported that the HUT levels had stabilized.

At about 4:15 a.m., CV pump suction was manually transferred back to the VCT from the RWST.

The control of the event was transferred from the control room to the Technical Support Center upon it's being activated at 4:27 a.m.

At 4:35 a.m., the GSEP "ALERT" was terminated and appropriate notifications were made.

C. Concerns Identified

Concern was raised with regard to the status of the installed relief valve on the 1A RHR pump suction since the data in the most recent Maintenance Nuclear Work Request (NWR) was unclear on the valve adjustment settings. During the determination of the premature opening and excessive blowdown of the 1B RHR pump suction relief valve, the inspector identified several concerns related to maintenance activities. There were significant discrepancies in nozzle ring settings recorded in work packages at the site, combining different methodologies for nozzle ring settings in the same work package may be confusing and mechanics signing off steps in work packages that deviated from directions without appropriate explanations. This evidence shows a weakness in the maintenance of the safety valves with respect to the nozzle ring settings. An agreeable resolution was achieved through telephone conference calls between the licensee, Region III, and NRR, on December 5, 1989. The licensee committed to administratively restrict the pressure in the suction of the 1A RHR train to 375 psig until Valve 51, in the 1A RHR train can be removed, replaced, and bench tested. Although normal plant procedures will prevent excessive pressure in that line, the licensee provided additional guidance to operations personnel by Special Operating Order, SO-ST-0036 (see Attachment 7) which stated:

> "The maximum pressure allowed at the RH suction valves is set at 375#. This is an administrative limit due to the setpoint problems on these relief valves.
>
> Ensure the pressure is maintained within this limit whenever the RH system is aligned for shutdown cooling."

This is to prevent challenging the 1A RHR pump suction relief valve. In addition, the licensee committed to determine the root cause of the 1B RHR pump suction relief valve premature opening and excessive blowdown as soon as possible and provide that knowledge and report to the senior resident inspector promptly. In addition, the licensee committed to evaluate the status of the 1A RHR pump suction relief valve after replacing it with an acceptable duplicate valve.

Some disparity was identified between the logs developed during the event when compared to computer data and graphic plots. In the past, log keeping has been an enforcement issue at Braidwood (reference Inspection Reports No. 50-456/88008(DRP); 50-457/88009(DRP)) and with issues of lesser significance noted subsequently. The strongest example during this event was a discrepancy of as much as twelve minutes noted between when the PZR level was observed dropping by the log entry and the licensee's scenario or computer data. Other variations were also noted. The licensee should evaluate this as part of the event followup and as an example for re-inforcement of the importance of keeping accurate logs.

In evaluation of the adequacy of licensee action and timeliness of the actions and notifications, it was noted that the GSEP "ALERT" declaration was made appropriately. In addition, the State was notified within ten minutes, meeting the fifteen minute criteria. Record review shows that the NRC was notified via ENS phone at 59 minutes within the 60 minute criteria. It appears that since the condition existed and the knowledge was available, the NRC could have been informed sooner. The issue of prompt notification to the NRC is the subject of the NRC Information Notice (IN) 88-32 "Prompt Reporting to NRC of Significant Incidents Involving Radioactive Materials" dated May 25, 1988. Although this Information Notice does not directly apply to this event, the concept applies in order for the NRC personnel to activate their resources as described in Information Notice (IN) 86-18 "NRC On Scene Response During a Major Emergency" dated March 26, 1986. Additionally, although the notification met the regulation of 10 CFR 50.72, timely and accurate ENS notifications have also been an enforcement issue in the past at Braidwood (reference Inspection Reports No. 50-456/89022(DRP); 50-457/89022(DRP)). The licensee should evaluate this observation.

Based on the measured pressurizer level changes during the first 12 minutes of the event and allowing for differences in measured letdown flows and charging flows, the estimated system loss rate appears to be significantly greater than the estimated capacity of valve 33 under the prevailing pressure conditions. In addition, the measured increase in hot leg temperatures for the first 18 minutes following the event appear to be lower than expected from the pressurizer discharge flows and appear to extend for a longer period of time than would be suggested by the pressurizer discharge. The licensee should evaluate this observation as part of this event analysis.

D. Observations

The team reconstructed and validated the sequence of events associated with this loss of reactor coolant inventory. The team used various logs, the licensee's summary, computer data printouts and graphs of various parameters, interviews with various personnel, observing installed instrumentation and equipment, comparison of findings with other team members and corporate knowledge of the plant. The team found general agreement between the data and their perception of the event except for the apparent discrepancy between the estimated blowdown flow rate during the initial portion of the event and the expected capacity of an RHR suction relief valve with improper settings under the prevailing system conditions and hot leg temperature measurements during the this period compared to the apparent discharge from the pressurizer. The team was able to clarify or answer a number of questions that came from senior agency management personnel including inquiries about availability and use of other ECCS equipment, operating personnel resourcefulness in terms of information available during the event, i.e., use of reactor vessel level indication system (RVLIS), RHR flows, pump motor currents, use of a volt meter to read PZR level, etc.

Some disparity was evident between logs developed during the event and verifiable data, such as computer data and graphs. Refer to concerns and recommendations.

The team found that operations personnel knew of the Technical Specification and procedural requirements to have one CV pump and both SI pumps removed from service under these conditions to assure cold over pressure protection of the RCS boundary. Their actions were found to be cautious and deliberate when returning the 1B CV pump to service. Throughout the event, there was always evidence of RCS pressure, RVLIS never dropped below 100% and, the running reactor coolant pump and RH pump motor currents showed no fluctuations indicating voiding. It was also noted that when the 1B CV pump was started, pressurizer level began to return indicating the leak flow rate was greater than the capacity of one CV pump but less than the capacity of two. With this information in mind, although an option, placing more ECCS equipment in service was avoided and efforts were concentrated on isolating the source of RCS loss.

The team also reviewed the adequacy of the licensee's response to the event and whether actions and notifications were timely. Through the review, the team found that the response was acceptable and timely, considering the conditions at the time and availability of equipment.

Notifications to the State and NRC, declarations of the GSEP conditions, and activation of the Technical Support Center (TSC) all appeared to be within the prescribed time frame. Questions or concerns about the promptness of notification of the NRC are addressed under concerns.

III. Investigative Efforts

A. Synopsis of AIT Activities

The AIT members arrived onsite at 8:00 a.m. (CST) on December 1, 1989, with the exception of S. Israel who arrived in the afternoon. At 11:00 a.m. (CST), the AIT members present met with the licensee to review the purpose and agenda of the AIT, introduce the team members, schedule interviews with selected personnel, establish quarantine requirements of the affected valve, and receive a detailed briefing on the event by the licensee. The team was provided with pertinent records and documentation, including a sequence of events, which the licensee had compiled at that point in time. In response to a request from the team, the licensee provided additional records, documentation, computer printouts, charts, graphs, and work package information as it became available.

The AIT had various meetings with various elements of the Braidwood Station Management and staff throughout the assessment of the event including meetings and teleconferences which included the Crosby Valve Company representative serving as an onsite consultant. Subsequent to teleconferences with Headquarters and Region III, the plant was allowed to continue operations to allow restart of the unit which was scheduled to occur on December 11, 1989. Operation was to continue under operating procedures which had been re-inforced by a Special Operating Order requiring that the RHR system not be placed in operation at RCS pressures above 375 psi. The system has an interlock designed to prevent automatic operation of the suction isolation valves above an RCS pressure of 360 psi.

The AIT investigation also included interviews of control room operating personnel and an analysis of the event conducted on the Braidwood simulator. The AIT documented several issues and concerns which were presented to the licensee at the exit meeting held on December 4, 1989 at the conclusion of the onsite activities.

B. <u>Licensee Actions</u>

At the conclusion of the event, the licensee initiated Nuclear Work Requests (NWRs) to remove the 1B RHR Train suction relief valve for evaluation. The 1B RHR Train was isolated and drained down to remove the 1B RHR Train suction relief valve referred to as "Valve 33" (the station assigned valve serial number to identify this valve). Valve 33 was removed from the system and placed on the test bench on December 2. With the Crosby valve consultant and representatives from INPO and the NRC in attendance, the Valve 33 was bench tested on December 2, 1989. The "as found" lift pressures, on three consecutive tests, were determined to be 411, 407, and 405 psig, respectively.

In addition, the licensee immediately began a Potentially Significant Event investigation on December 1, 1989. The findings of this investigation have been documented in the licensee's PSE Preliminary Report dated December 5, 1989, and provided to the NRC. This PSE Preliminary Report is Attachment 6 to this report. The conclusions as to the apparent cause and the facts related to this event closely parallel the findings of the AIT.

On December 13, 1989, the licensee conducted a bench lift test on Valve 51, removed from the 1A RHR train suction relief valve position. Preliminary results of that lift test indicated that Valve 51 lifted at 465, 463, and 462 psig on three consecutive lifts. The Valve 51 nozzle ring setting appeared to be correct and normal prior to the lift test. An official record of these results will be provided later.

C. Operator Performance and Procedure Adequacy

On December 2, 1989, as part of the AIT charter, a team inspector interviewed five (5) licensed operators who were assigned to the Unit 1 control room during the December 1, 1989 event. The operators were interviewed to assess the adequacy of their performance during the event and to determine the adequacy of the procedures used. Additionally, the team inspector asked what types of training the operators received concerning a Mode 4/5 LOCA.

The operators were interviewed singly by one team inspector for approximately 45 minutes and followed an interview protocol (see Attachment 4) drafted to serve as a rough guide. The team inspector used his judgement and discretion to pursue relevant topics beyond the protocol. Followup questions were asked in an attempt to determine the reasons for the operating crew's actions.

The operators were fully cooperative and appeared to respond with candor to the questions. (It should be noted here that the operators had been interviewed by both Braidwood Station Management and an INPO team prior to the AIT interviews.) The team inspector felt confident that valid conclusions could be drawn based on the interviews.

Followup questions focused on the following specific topics: (1) the diagnosis of the event and (2) charging injection flowpath. The descriptions below are a composite of the operators interviewed and are not intended to be an exact narrative. (For a sequence of events, see Attachment 3.)

1. Diagnosis of the Event

The operating crew was asked to describe their thought process during the identification and isolation of the leak. The operators made their initial identification of the leak from rapidly decreasing pressurizer (PZR) level and reactor coolant system (RCS) pressure. After isolating letdown and maximizing normal charging, the crew monitored various containment parameters (pressure, humidity, temperature, radiation, sump levels) and determined that the leak was not in containment. The crew next sent a team of in plant operators into the Auxiliary (Aux) Building to identify the source of the leak. Concurrently, the Radiation Waste Control Station was called to check for any abnormal tank level increases. The Radiation Waste Control Station reported that the Recycle Hold Up Tanks (HUTs) levels were increasing. (The HUTs are the collection point for the Residual Heat Removal (RHR) pump suction relief valves.) The operating crew isolated the operating 1A RHR train and started the 1B train.

The operators were asked why they did not isolate both trains of RHR when the loss of coolant was determined not to be in containment. They stated a reluctance to isolate the remaining high volume injection path with both Safety Injection (SI) pumps tagged out. (The SI pumps are the intermediate head injection pumps.)

When the 1A RHR train was isolated, the Radiation Waste Control Station reported that the HUT level increase had slowed. Additionally, an operator searching the Aux. Building reported water dripping in the vicinity of the 1A RHR suction line. (The water was from a mechanical failure of relief valve 0AB-8634.) These reports, coupled with a lack of indications of a LOCA in containment or the Aux. Buildings (radiation monitors were normal), raised the operators confidence in their preliminary diagnosis that the RHR pump suction relief valve(s) had lifted. However, the control board indicated that RCS pressure was still dropping with no observable PZR level. (The operator used all channels of PZR level, water space temperature, and surge line temperature to confirm this.) The operating crew sent an in plant operator to locally check the 1B RHR pump suction relief valve. He reported it open.

Based on the reports of leakage in the vicinity of the 1A RHR pump suction and a decline in the HUT level increase when 1A RHR train was isolated and a lack of evidence of a leak in containment or the Aux. Building, the operators depressurized the 1A RHR train as a precautionary measure. This was done to assure the 1A RHR relief valve was shut and the leak not reinitiated when the 1A train was returned to service. After the RHR trains were switched and 1B RHR train was isolated, RCS pressure and PZR level were rapidly restored. The operators then stabilized the plant.

Based on the above description, the operating crew appeared to have systematically and deliberately isolated the leak. They appeared to have monitored control room indication and solicited field reports to identify and isolate the leak in a reasonable time.

2. Charging Injection Flowpath

The operating crew was asked to describe their rationale for restoring the "B" Centrifugal Charging Pump (1B CV), which included untagging and "racking in" its circuit breaker, vice injecting the running CV pump through the ECCS high head flowpath (SI-8801A and B). The operating crew gave several reasons for their decision:

- Overpressurizing the RCS

  The operators were concerned that using the ECCS injection path would rapidly fill the PZR solid and overpressurize the RCS. By using the normal charging flowpath, the operators felt they could control injection rate, PZR level and limit the PZR pressure temperature transient. When the 1B CV pump was started, the operators monitored PZR surge line and water space temperature to verify that PZR level was being restored. When these two temperatures rapidly dropped, the operators throttled charging flow to maintain PZR level low on the cold calibrations level channel.

- Avoid "thermally shocking" the PZR

  While drawing a PZR bubble, the operators are required to maintain PZR heatup rate less than 50 deg.F/hr. With the PZR heatup/cooldown limits recently reviewed, the operators were very concerned about exceeding either the 100 deg.F/hr procedural cooldown limit or the 200 deg.F/hr Technical Specification (TS) limit.

- RVLIS Indication

  The operators were monitoring the reactor upper head and reactor vessel plenum level indications throughout the event. Both indicators were at 100 percent during the event. This helped increase the operators confidence that the reactor vessel had not voided.

- Adequate Decay Heat Removal

  The operators monitored RCS and RHR flows and pump currents (ampheres) for indications of steam bubbles in the loops or pump cavitations during the event. All flows and pump currents were normal and did not give any indication of a loss of decay heat removal capability.

  Based on the above discussion, the operator appeared to have clear, well founded reasons for deciding to start a second CV pump and inject through the normal charging path vice the ECCS path. When the second CV pump was started, the operators had increasing RCS pressure, observable PZR inventory and adequate decay heat removal. Even though the leak was still in progress, they felt the plant was stable enough to continue to search for the leak.

D. Procedures

The inspector reviewed procedures used during the event, control room logs, various records and personnel interviews to assess the adequacy of the procedures used to mitigate this event. Prior to the event, the operators were drawing a PZR bubble per BwOP RY-5 (Rev. 2), "Drawing a Bubble in the Pressurizer." When PZR level and RCS pressure decreased rapidly, the operators entered two abnormal operating procedures, PRI-1 (Rev. 53), "Excessive Primary Plant Leakage" and PRI-10 (Rev. 54), "Loss of RH Cooling."

The operators stated that these procedures had the initial actions for maintaining RCS inventory and leak isolation. The procedures had charging flow increased, start a second CV pump and isolate letdown flow. However, the operator stated that PRI-1 and PRI-10 did not address a Mode 5 intersystem LOCA. The operators felt that the procedure gave them a good starting point to mitigate the event. The operators used piping diagrams, field reports, system knowledge (based on experience and training), and good engineering practices to isolate the leak and restore PZR level.

The inspector determined that the Westinghouse (W) Emergency Operating Procedures (EOPs) were not written to address a Mode 5 LOCA. This is a previously identified weakness in the W EOPs and not specific to the Braidwood Station.

E. Training

The operating crew was asked to describe any training they had received concerning a Mode 4/5 LOCA. The operators stated that at least one requalification simulator training session, during the past year, almost duplicated the condition of the event. The simulator scenario had an RHR pump suction relief valve stuck open while the plant was on RHR cooling.

The operators stated that the major focus of the training was to raise their awareness level of the unique problems a Mode 4/5 LOCA presents. ECCS equipment normally immediately available is out of service. Overpressurizing the RCS while in Mode 4/5 is an immediate concern. (PZR Power Operated Relief Valves (PORV) have a reduced lift setpoint with the PZR Cold Overpressure Protection circuit armed.) Abnormal operating procedures (AOPs) and EOPs are not written to address Mode 4/5 LOCAs. Additionally, the operators were trained to contact the Radiation Waste Control Station to check for tank and sump levels increasing in order to locate the leak. The operators also received classroom training and self-study (required reading) on a Mode 4/5 LOCA.

Based on the above discussion, the inspector concluded that the training the operating crew received was adequate to mitigate the event.

During the course of the AIT, questions were raised concerning the fact that the operating crew apparently abandoned the AOPs (PRI-1 and PRI-10) and did not use the W EOPs. The general sense of the issue focused on how management addressed procedural compliance when procedures did not address the specific event.

Section C.1 of Braidwood Administrative Procedure (BwAP) 340-1 (Rev. 6), "Use of Procedures for the Operating Department," directly addresses this concern.

It is recognized that procedures cannot (and should not) be specifically written for every contingency. If an approved plant procedure does not exist which applies to the current situation, personnel are instructed to take action so as to minimize personnel injury, damage to the facility, and to protect the health and safety of the general public and the personnel onsite.

This same statement is in Section 5.4.1.3 of Nuclear Operations Directives (NOD) NOP-OP.1 (Rev. 1), "Conduction of Operations." This corporate level procedure was authorized by the General Managers and Vice Presidents of both PWR and BWR Operations. Additionally, the Procedure Compliance section of BwAP-300-1 (Rev. 3), "Conduct of Operations," Section C.4.a.3) states:

In cases of emergency, plan operations personnel are directed to take such action as is necessary to minimize personnel injury and damage to the plant; to return the plant to a stable, safe condition; and to protect the health and safety of the general public and the personnel onsite. These actions shall be documented and, if appropriate, incorporated into a revision of the affected procedure.

As previously identified, the W EOPs were not written to address a Mode 4/5 LOCA. The training the operators received already identified the weakness.

Based on the procedures cited above and the rationale the operating crew gave for their actions, the inspectors concluded that procedures were correctly followed within their limited scope and any additional actions were based on protecting the health and safety of the general public and personnel onsite.

IV. RHR Suction Relief Valves

The suction relief valves in the RHR system are spring loaded (style JB-35-TD-WR) with a nominal set pressure of 450 psig. Five valves are used at the Braidwood station with serial designations 32, 33, 35, 49, and 51. Four valves are used in the two units and the fifth is a spare. The set pressure is adjusted by an adjusting bolt above the valve spring and is locked into place by a locknut.

Valve characteristics are set by a nozzle ring threaded on the nozzle. The nozzle ring is set by a predelivery test performed by the manufacturer. Each valve has a unique nozzle setting and it is locked in place by a set screw.

A. RHR Suction Relief Valve Maintenance

Based on vendor test data for these valves, the subcooled blowdown capacity of a properly configured valve is about 900 gpm at full accumulation. Discussions with the manufacturer elicited an opinion that the capacity with a misconfigured valve at smaller pressure differentials would probably be less. The estimated blowdown flowrate based on the hot calibrated pressurizer level sensors for the first few minutes of the event is higher than a best estimate of the capacity.

In Section 5.4.7.2.3 of the updated FSAR, these relief valves are designed for a relief capacity of 475 gpm at 375 F to prevent RHR system overpressurization during plant heatup. This design condition would result in a two-phase flow discharge from the valves. The supporting basis for the as built capacity of the valves under two-phase flow conditions was not pursued during this inspection.

Twenty-two work requests for all five valves were reviewed for the period 1986 to the present. Five of these requests were to repair damaged parts of the valve. Each valve was bench tested to verify the set pressures at least twice during this period. These tests were performed according to procedure BwMP 3305-043, Rev. 1, that was approved in 1986, using a certified pressure gauge. Prior to the event of December 1, 1989, the "as found" set pressures ranged from 420 to 605 psig. The low value was obtained for Valve 32 on Unit 1 while still in the construction stage. The high value was obtained for Valve 35 on Unit 1 prior to repair of a broken disc insert pin.

Generally, the "as found" set pressures were from 448 to 468 psig. Three readings of the apparent "as left set pressures" wer ll within a 3 psi and met the test requirement of 455 plus or minus 1% psig. Valve 33, which was in the "B" RHR train of Unit 1 at the time of the event, was tested April 15, 1988, and had apparent measured values of 453, 454, and 455 psig. In addition, it met a "no leak" test of 405 psig. Valve 51, which was in the "A" train of Unit 1, was last tested on October 17, 1989, and had measured pressures of 455, 455, and 455 psig. Valve 51 met a "no leak" test at 420 psig.

The compliance of the pressure test procedure with existing code requirements was not examined during this inspection.

Blowdown of these valves is not performed after purchase from the vendor. The formal procedure used to verify the settings is BwMP 3305-072, Disassembly-Reconditioning-Reassembly of Crosby Style JB-TD-WR ("L" orifice Relief Valve) which was approved on October 17, 1988. This procedure is used for the disassembly-reconditioning-reassembly of these safety valves. The procedure directs "ROTATE nozzle ring (3) to the right (counterclockwise) COUNTING notches moved until ring (3) stops." The number of notches (located around the circumference of the bottom of the ring) is counted and recorded. When the valve is reassembled, the process is reversed by running the ring up against the stop and then "RETURN nozzle ring (3) to its original position by counting the number of notches downward as obtained in step F.2.c.1."

This direction is consistent with the available vendor's technical manual. This ring setting movement was intended to be implemented for valve repairs.

In late March 1988, Valve 35 was found leaking excessively and subsequent examination attributed the problem to a broken disc insert pin. A low nozzle ring setting was identified as the cause of the pin failure because it did not provide adequate cushioning when the valve reseated after lift.

In April 1988, four of the RHR suction relief valves received maintenance on almost weekly intervals. Three of the valves required specific repairs (replaced broken disc insert pin, replaced nozzle, and lapped valve seat). Three of the four work requests contained a station traveler, in addition to the formal procedures in the work requests, that specified a nozzle ring setting verification test that was distinctly different from that used in BwMP 3305-072.

The test description in the travelers were essentially the same except for detail. The nozzle ring setting was supposed to read "Using a screw driver or similar tool turn the nozzle ring to the right (counter clockwise) counting notches until the bottom edge of the nozzle ring hole is level with the leading edge of the nozzle. Record notches turned." The next direction is "Now turn the nozzle ring to the left (clockwise) -XXX notches. This is considered the manufacturers original nozzle ring setting". Each valve has a unique setting based on the vendor's predelivery test. The specific values recorded and specified were:

	As Found	Required
Valve 33	263	110
Valve 49	89	110
Valve 32	198	105

The reason for the discrepancies between the "as found" and "required" was not resolved based on discussions with the maintenance staff during the inspection. Valve 33 had several evolutions as part of the work request (NWR-17190) that were performed (April 12, 1988 to April 15, 1988) in the following sequence:

a. The valve was pressure tested per BwMP 3305-043, "Bench Setting/ Setting of Residual Heat Removal Suction Relief Valves."

b. Checked nozzle ring setting per a station traveler similar to the aforementioned.

c. Replaced the nozzle and lapped the seat per BwMP 3305-072.

d. Retested the pressure setting per BwMP 3305-043.

The ring setting step, Item b. above, had a correction noted on the traveler changing counter clockwise to clockwise and changing clockwise to counter clockwise. It is unclear what clockwise or counter clockwise mean when you are looking edgewise at a nozzle ring.

As previously noted, this repair procedure, BwMP 3305-072, includes a step to record the number of notches of rotation to move the nozzle ring to its stop. This value was recorded as 343 in Step F.2.C.1. After the repair and reassembly, F.5.0.2b, directs the mechanic to return the nozzle from the stop position to the original position by backing off the number in Step F.2.C.1 (343). The number recorded and initialed is 110 and no explanation was noted on the procedure sheet.

There is no independent verification of these actions noted on the work request.

This valve was installed in the B RHR suction line of Unit 1 on April 18, 1988. After the December 1, 1989 event, Valve 33 was removed from Unit 1 and bench tested on December 2, 1989. After the aforementioned set pressure test, a nozzle ring setting verification was performed using instructions in a station traveler. The mechanic was directed at Step MM 2C to turn the nozzle to the right counting the notches until the bottom edge of the nozzle ring hole is level with the leading edge of the nozzle. The notation at this step in the work request is "107, backed off 3, turned forward 3, backed off 5, engaged pin," and the step was initialed. According to discussions with the maintenance personnel, no lineup occurred and the nozzle ring jammed. This was not noted on the traveler at Step MM 2C. Subsequent review by the Braidwood staff of photographs of the "as found" nozzle ring and the view through the holes in the ring seemed to indicate that the nozzle ring was higher than it should be. All work on the valve was halted until a plan was developed for troubleshooting the valve to explain the premature valve lift and excessive blowdown during the December 1, 1989 event.

Examination of NWR 21291 for Valve 35, which was the original cause of the concern about improper nozzle ring settings, indicates a similar occurrence when setting the nozzle ring was adjusted under Procedure BwMP 3305-072. In this instance the notches recorded before the valve repair was 429 and the "as left" notches was "112 per vendor rep." The step was initialed without any further explanation. Verification of the nozzle ring setting at a later date (NWR 37284) indicated that it was the proper position.

Examination of the work packages for the four valves installed in both units as of December 4, 1989, indicated that the latest nozzle ring settings were performed by a station traveler for each valve. The instructions in the travelers were slightly different. Three of the travelers had the nozzle ring setting explicitly stated within the traveler direction. The fourth traveler, for Valve 51, did not explicitly identify the nozzle ring setting, but referred the mechanic to a table with four valve settings in it. In addition, the language for establishing the reference point was inconsistent with the attached diagrams.

The action was initialed as complete.

Examination of vendor test data for each of the valves indicates a different method was used to define the nozzle ring setting than either of those described above. These documents were apparently received after the event on December 1, 1989.

Discussions with the valve vendor representative, on December 4, 1989, indicated that the method identified in the test report and that described by the station traveler were essentially the same, so that the traveler methodology, if implemented properly, provides assurance that the setting is essentially correct.

Observations

Measurements of the valve set pressures from 1986 up to the event give no indication that there was any concern about early valve lift, such as indicated by the December 1, 1989 event.

The actual process of setting the nozzle ring by moving a ring with a screw driven notch by notch and counting notches up to 400 appears to be prone to error, however, this seems to be the accepted industry practice.

The latest nozzle ring setting on Valve 51, presently installed in the 1A RHR system, was performed by an instruction that did not explicitly include the setting value. Discrepancies in mechanics actions were noted in two instances under similar conditions. This is addressed in Section II.C., "Concerns Identified."

The procedures for setting the nozzle ring position as articulated in the station travelers are not well documented and may be ambiguous to perform.

The nozzle ring setting on Valve 33 prior to the event may be in error because of mechanic error.

B. Early Valve Lift and Excessive Blowdown

On December 1, 1989, pressure in the 1B RHR pump suction line reached a maximum of 416 psig based on a computer printout of sensor P0602. The relief valve is located about 10 feet above the sensor point so the actual pressure at the valve would be about 5 psi less (411 psig). The valve was designed and tested so that the actual lift pressure should have been in excess of 450 psig accounting for accumulation, which could be as much as 10%. Valve 33 was removed from Unit 1 and bench tested on December 2, 1989. The "as found" set pressures were 411, 407, and 405 psig.

There have been three previous opportunities reported for the RHR suction relief valves to be challenged. These occurred as part of a surveillance test of the low temperature overpressurization system. These tests occurred in 1987 and 1988 on both units and pressures exceeded 450 psi. It was believed that the RHR suction relief valves were challenged at that time without any unexpected transients occurring, i.e., excessive blowdown.

Review of the past history from the work requests on this valve and the other RHR safety valves at the plant do not give any indication that this deviation could be anticipated. The previous set pressure tests did not show any unexpected results. The pressures tests performed on these valves were in accordance with the vendor's manual; however, after discussions with the valve representative, it appears that the vendor's manual is not sufficiently explicit to define the actual determination of the set pressure.

The vendor's manual states:

> "Set pressure is the value of increasing inlet static pressure of a pressure relief valve of which there is a measurable lift, or at which the discharge becomes continuous as determined by seeing, feeling, or hearing.
>
> 1.3.1.1 In a relief valve on liquid service the set pressure is considered to be the inlet pressure at which the valve starts to discharge under service conditions."

As tested, per BwMP 3305-043, the set pressure is defined by a sudden drop in pressure as the valve inlet is being pressurized slowly at less than 2 psi per second.

The vendor representative at the site opined that observation of a continuous discharge is the appropriate criterion. Subsequent discussions with the manufacturer indicated that the pop test, if performed correctly, should produce similar results. No premature valve lift was noted in the aforementioned plant tests and the valves had their pressures set in this manner. Compliance with specific code requirements was not determined during this inspection.

Each RHR suction relief valve discharges into a 4-inch diameter pipe that goes to a header that collects discharge from other relief valves and goes into the recycle holdup tanks. The holdup tanks are nominally kept at about 1 psig. Based on discussions with the Braidwood technical staff, no other valves were discharging into the common header prior to the event. Thus, the back pressure at the valves was nominally atmospheric and in any event, the safety valves have bellows surrounding the lower portion of the valve spindle to make them insensitive to back pressure. These safety valves are ASME Class 2 and require set pressure testing every five years. The Braidwood test program for these valves would involve staggered testing every other refueling cycle (assumed 18 months).

The external appearance of the valve after the event did not indicate any external damage according to discussions with the maintenance staff.

A similar excessive RHR suction blowdown event including loss of pressurizer level occurred at a foreign reactor in 1985 and was reported in an industry information report in 1986. In this instance, the valve, same model as Valve 33, was broken. The vendor submitted a report to the foreign utility in 1986. A related event was reported in LER 86-001 for Byron, Unit 2. In this instance a relief valve in the component cooling water system didn't reseat because of improper nozzle ring settings.

Discussions with the valve vendor representative, at the time of the site inspection, provided a spectrum of causes that could disturb the set pressure of the valve. These hypotheses were not necessarily specific to the prior history of the valve. In conjunction with the vendor representative, the licensee developed a plan to examine the valve on the test bench and identify the root cause of the early valve lift.

The valve did not reseat until the 1B RHR loop was isolated and the pressure was about 300 psig. This valve should have reseated at a much higher pressure if the valve was properly configured and no adverse condition acting upon it. The reseat pressure is strongly influenced by the dynamic forces created by the nozzle ring just above the nozzle. This area was designed to have some exposed flow area based on a proper position of the nozzle ring.

If the nozzle ring is located too high on the nozzle, it may result in an inadequate ventilation area just above the nozzle. Such a configuration may cause a much lower reseat pressure because of the reactive forces developed. There is also the possibility of hangup of the valve because of misalignment or foreign material. The licensee will examine the cause of the lower reseat pressure that occurred during the event.

Subsequent to the AIT exit on December 4, 1989, the team was informed on December 11, 1989, of findings in the disassembly of the 1B RHR pump suction relief valve, that may have contributed to its adverse performance. A small groove was found in the spindle of the valve where the spindle interfaces with the spindle guide. This was apparently caused by an unknown foreign particle(s) when the valve was assembled which affected the adjustment or lift setting. The nozzle ring setting was also found out of proper position which affected the blowdown setting. This was apparently due to improper adherence to work request instructions which will be evaluated upon review of the Licensee Event Report (LER).

The estimated blowdown flowrates for the first few minutes of the event appear to be higher than the estimated capacity of Valve 33 under the prevailing pressure conditions.

V. System and Equipment Performance

A. RHR Train Operation

The system configuration prior to the event is as follows:

The reactor was in Mode 5 in preparation to go to Mode 4 and drawing a bubble in the PZR. The 1A Train RHR cooling was in operation with letdown flow to the Chemical Volume Control System (CVCS) established. The RCS, at 0140, was at 400 psig and increasing, letdown flow to the letdown heat exchanger was at maximum and charging flow through the 1A CV pump was throttled.

At 0142 the RCS pressure started to drop and the PZR level (Hot cal.) indication dropped from off-scale high at 0143 to on-scale then to off-scale low at 0152. At the same time PZR level indicated off-scale low, letdown flow was isolated and charging flow was increased to maximum. At 0158, the 1B reactor coolant pump (RCP) was stopped and RCS pressure decreased to 272 psig. At 0154 RHR was switched from 1A RHR train to 1B RHR train in order to isolate the 1A train which was suspected to be the loss of RCS inventory.

The Unit 1 RVLIS was in operation with the upper head sensors reading 130°F Delta T. Ideal Delta T values should be less than 150° to avoid nuisance alarms and provide proper water level indication. According to the computer printout of RVLIS, level indication in the upper head was constant at 100% and Delta T for the sensors was steady at 130°F. This provided positive indication that there was no voiding in the upper head of the reactor. The overall effect of RCP operation will be negligible due to low flow in the upper head region of the reactor vessel. Therefore, RVLIS response in the upper head region is expected to be the same with or without the RCPs in operation.

Prior to the event the 1A RHR train was in operation providing shutdown cooling with letdown to the CVCS throttled to maintain RCS pressure and level as a bubble was being established in the PZR. At approximately 0142, the 1B RHR train suction line relief valve (1RH 8708B) lifted causing a rapid drop in RCS pressure (from 404 psig to 272 psig at 0200). The operations personnel began evolutions to secure letdown flow and increase flow through the 1A CV pump to maximum (approximately 400 gpm). The system reached an equilibrium point with respect to RCS pressure at approximately 0200 with the RCS pressure leveling at 272 psig. Pressure in the RCS began to increase slightly at 0206 and made a steady but slow rise as charging flow exceeded the loss of RCS inventory from the relief valve. At this point, it was assumed that the 1A RHR train relief was the valve that had lifted. During this period the PZR volume (14,000 gal.), decreased to zero and continued to drop down the surge line as indicated by the drop in PZR surge line temperature. In order to regain level in the PZR and isolate the relief valves the 1A RHR train was secured and the 1B RHR train was put in operation. In addition, the charging pump suction was switched from the VCT to the RWST. This was accomplished at approximately 0156.

Charging flow was being maintained through the 1A CV pump. At approximately 0159, the 1B RCP was shut down due to concern about reactor pump seal differential pressure caused by the drop in RCS pressure. At approximately 0215 an operator was dispatched to align the 1B CV pump and shut valve 1 RH 8734A from 1A RHR train to the CVCS. In addition, the 1A RHR train suction valve (1RH 8701B) was shut in order to isolate the leak which was suspected to be in the 1A RHR train at that time. At about 0235 the 1B CV pump was started to increase inventory in the RCS and recover PZR level. During this time, the RCS pressure began to increase and there was a rapid decrease in PZR surge line temperature indicating level was increasing in the PZR. The 1B CV pump was then secured at 0245 in the expectation that the leak had been isolated from the 1A RHR train and RCS pressure was on the rise. At approximately 0254, the 1B CV pump was restarted and the leak was determined to be in 1B RHR train suction header line. This was verified by sending an operator to the "curved wall area" to the location of the 1B RHR suction relief valve. The operator verified the 1B suction relief valve as the source of the leak by identifying noise coming from the valve. At this point, the 1B RHR train was isolated and the 1A RHR train was placed in service.

B. RWST Flow Paths and Volumes

In evaluating this event, one area of investigation was the flow rates and volumes associated with the various evolutions. The PZR, including the surge line, contains approximately 1906 ft^3 of water when the system is solid. At the onset of the event (0143) the level in the PZR, using the computer data points, fell from off-scale high to off-scale low (0152) in about 9 minutes.

The surge line temperature began to drop approximately one minute after this. This corresponds to an initial flow rate from the RCS of approximately 1400 gpm. The charging pump was making up inventory at the rate of about 400 gpm.

The volume of water from the RWST through the charging pumps is estimated to be 54,000 gallons. This data was taken from the strip-chart recording or RWST level which indicated a drop in inventory of approximately 12%. RWST water volume is calculated using the relationship of 4500 gal/%. Additionally, the recycle HUTS indicated an increase in level of approximately 56% due to the loss in RCS inventory from the RWST and the PZR. The HUTS volume increase corresponds to approximately 70,000 gallons. This corresponds well with the volume of water calculated as leaving the RCS.

An RCS mass/flow balance indicated the volume lost from the PZR and the RWST was accounted for in the HUTS.

C. Radiation Protection

During the event, an estimated 30 to 50 gallons of water were released from the affected systems through a weep hole in a relief valve with its inlet supply common to the RHR pump suction relief valves. The area of the release was limited to a portion of the 364' elevation of the Unit 1 auxiliary building. Through review of surveys, radiation monitor data, stack monitor data, and interviews with Health Physics personnel, the inspector was able to determine that there was no evidence of an unplanned release of radioactive material to the environment as a result of this event. In addition, there was no evidence of an airborne release to the affected area of the auxiliary building. Removable surface contamination detected in the wetted area was cleaned up to levels acceptable for normal access with appropriate protective clothing.

VI. Safety Significance of the Event

A. Immediate

The unit was in the 101st day of a refueling outage which involved a complete core off load. The potential for a temperature increase from decay heat did not exist. Based on actions taken by the operating crew by increasing the charging flow into the RCS using up to two CV pumps, the level in the PZR retreated down into the surgeline, however, based on surgeline temperatures it is believed that the prompt increase in CV flow maintained the RCS inventory in the surgeline. Additional indication to support this position is based on the control board indication of RVLIS which the operators observed to indicate 100% throughout the event. Another piece of information to support this position is that when operators vented the RCS through the head vents, no gases were found to be present in the effluent. Therefore, in view of the indications that the RCS inventory remained in the surgeline, RVLIS remained at 100% throughout and no gases were present when venting, it is concluded that no voiding of the head occurred, no loss of RHR cooling occurred and no potential for heating of the core occurred. The immediate safety significance of the event was minimal.

B. Under Operational Conditions

Under the current operational configuration, Valve 51 is installed in the 1A RHR train suction relief valve position, had a satisfactory lift test on October 17, 1989 and had measured lift pressures of 455, 455, and 455 psig on these consecutive tests. Valve 35 was used to replace Valve 33 in the 1B RHR train suction relief valve position and had a satisfactory lift test prior to installation on December 2, 1989. The RHR system is now lined up for Engineered Safety Features Operation, the RCS suction isolation valves are shut and lined up for automatic operation. Under the current lineup, the RHR system will not experience RCS pressure in excess of 360 psig until and unless called upon to provide RHR cooling flow in the event of a LOCA. Even in the event the system is called upon to function, the maximum pressure the system will see is 360 psig due to an interlock designed into the system to keep the suction isolation valves shut until pressure decreases to 360 psig. As a worst case condition, in the event that the unit is forced to go to Mode 5 (cold shutdown) and the PZR taken solid, the potential exist to expose the RHR suction piping to RCS pressures above 360 psig. This pressure transient can occur during the process of drawing a PZR bubble. To prevent this occurrence, the licensee has located an additional replacement valve and plans to install this new valve following lift pressure testing or install the Valve 33 prior to the unit coming out of the normal operating temperature and pressure lineup for power operations. Therefore, the safety significance of the event under operational conditions is of little concern. During Mode 5, with the relief valves properly tested, the over pressure protection will be in place and proper setting of these valves should minimize the previously described event.

VII. AIT Concerns and Recommendations

A. Concerns

1. Why did the 1B RHR pump suction relief valve lift prematurely and fail to reseal properly.

2. The status of the 1A RHR pump suction relief valve.

3. The maintenance of safety relief valves and how their lift points and nozzle rings are set, i.e., following procedures.

4. Disparity between times shown in written logs and verifiable computer data or records.

5. Notification of the NRC could have been made sooner.

6. Lack of a Mode 4/5 Loss of Coolant emergency operating procedure (EOP).

B. Recommendations

1. The licensee should pursue the root cause of the premature lift and failure to reseat properly of the 1B RHR pump suction relief valve and report those findings to the NRC as part of their Licensee Event Report (LER).

2. Remove and replace the 1A RHR pump suction relief valve and determine its status, i.e., lift point and blowdown ring setting and report it to the NRC.

3. Maintenance of safety relief valves should be the subject of a more intensive maintenance inspection by the NRC.

4. The licensee should review the log disparities and provide guidance to operations personnel.

5. The licensee should evaluate the criteria for NRC notifications as described in Information Notice IN 88-32 and instruct operations personnel that this should be done as soon as possible during an event.

6. The licensee should provide this experience to the Westinghouse Owners Group (WOG) with the urgency on the need for a Mode 4/5 Loss of Coolant Emergency Operating Procedure (EOP).

7. The licensee should review, consolidate, and improve the procedures for setting the nozzle ring position on the RHR suction relief valves and all similar valves used in the plant.

8. The licensee should resolve any discrepancies between the apparent blowdown flowrate during the initial phase of the event and the capacity of Valve 33 under prevailing pressure conditions and also evaluate the hot leg temperature measurements in light of the pressurizer discharge characteristics.

VIII. AIT Conclusions

The AIT finds that the sequence of events, as developed by the licensee using available records and documents including logs, computer printouts, charts, graphs, and nuclear work packages, coupled with interviews of the individual involved in the event, is a true and valid description of the incident.

As noted in Paragraphs II.D and III.C, it was concluded that the licensee's actions, in response to the incident were both acceptable and timely in view of the plant conditions and indications available to the operators.

Pending review of the licensee's root cause analysis and results of the valve failure investigation, it appears that the premature lift of the valve may be caused by foreign material caught between the valve spindle and guide.

As discussed in Section V, it was concluded that the flow paths and volumes transferred during the incident are probably valid pending the resolution of uncertainty associated with the capacity of misconfigured Valve 33 and that the mass/flow balance of the volumes lost from the PZR and the RWST are closely accounted for and correspond well in the calculations.

Based on interviews with operators; review of procedures and training; and discussions with the licensee, as described in Paragraphs III.C, D, and E, it was concluded that the performance of the operators throughout the incident was reasonable and appropriate to the circumstances and plant conditions. The inspectors concluded that procedures were correctly followed within their limited scope and additional actions taken beyond the scope of the procedures was accomplished within the guidelines of the Braidwood Administrative Procedures and Corporate Directives. It was also concluded that a weakness exists in the EOPs in Mode 5, however, this is a previously identified issue not unique to Braidwood. Based on the training review, it was concluded that the training of the operators was adequate and did give a good basis to aid in mitigating this incident.

In addition, based on information provided by the licensee on December 13, 1989, that the lift pressures of Valve 51, removed from the 1A RHR train, were 465, 463, and 462 psig, respectively, it is therefore concluded that Valve 51 did not lift during the incident.

IX. <u>Exit Interview (30703)</u>

The inspectors met with the licensee representatives (denoted in Paragraph I.D.) throughout the inspection and evaluation of the events and at the conclusion on December 4, 1989, and summarized the scope and findings of the augmented inspection team's activities. The licensee acknowledged these findings. The inspectors also discussed the likely informational contents of the inspection report with regard to documents or processes reviewed by the inspector during the inspection. The licensee did not identify any such documents or processes as proprietary.

Attachment 1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

DEC 1 1989

MEMORANDUM FOR: J. M. Hinds, Jr., Chief, Reactor Projects Section 1A (Braidwood AIT Team Leader)

FROM: W. D. Shafer, Chief, Reactor Projects Branch 1

SUBJECT: AUGMENTED INSPECTION TEAM (AIT) CHARTER

Enclosed for your implementation is the Charter for the inspection of the events associated with the Braidwood Unit 1 Loss of RCS inventory which occurred on December 1, 1989. This charter is prepared in accordance with the NRC Inspection Manual Chapter 0325. The objectives of the AIT are to communicate the facts surrounding this event to Regional and Headquarters management, as well as to identify and communicate any generic safety concerns related to findings and conclusions of the onsite inspection.

If you have any questions regarding implementation of the enclosed Charter, please contact me directly.

W D Shafer
W. D. Shafer, Chief
Reactor Projects Branch 1

Enclosure: AIT Charter

cc w/enclosure:
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C. J. Paperiello, RIII
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SRI, Braidwood

Augmented Inspection Team (AIT) Charter
Braidwood Unit 1 Loss of Reactor Coolant Inventory

You and your team are to perform an inspection to accomplish the following:

1. Develop and validate the sequence of events associated with the loss of reactor coolant inventory that occurred on Unit 1 on December 1, 1989.
2. Determine the adequacy of the licensee's response to this event and whether the actions and notifications were timely.
3. Determine the root cause for the opening of the RHR Suction Relief Valves on the B RHR system.
4. Determine and validate the flow path and volume of water discharged from the RCS and RWST to the Holdup Tank.
5. Assess the adequacy of operator performance during the incident by review of records and logs; and through interviews with the personnel on duty.
6. Identify and determine the adequacy of the applicable operating procedures.

ATTACHMENT 2

TABLE OF ACRONYMS

AEOD	Office for Analysis and Evaluation of Operational Data
AIT	Augmented Inspection Team
AOP	Abnormal Operating Procedure
ASME	American Society of Mechanical Engineering
AUX	Auxiliary (Building)
BwMP	Braidwood Maintenance Procedure
BwOP	Braidwood Operating Procedure
CAL	Calibration
CECo	Commonwealth Edison Company
CV	Coolant Charging
CVCS	Chemical Volume Control System
CST	Central Standard Time
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
ECCS	Emergency Core Cooling System
ENS	Emergency Notification System
EOP	Emergency Operating Procedure
GSEP	General Site Emergency Plan
HUT	Recycle Holdup Tanks
IN	NRC Information Notice
INPO	Institute of Nuclear Power Operations
LOCA	Loss of Coolant Accident
MODE 5	Cold Shutdown
NARS	Nuclear Accident Reporting System
NOD	Nuclear Operations Directives
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulations
NWR	Nuclear Work Request
OSS	Out-of-Service
PZR	Pressurizer
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
PORV	Power Operated Relief Valve
RH	RHR System Designators
RHR	Residual Heat Removal
RIII	Region III NRC
RVLIS	Reactor Vessel Level Indicating System
RWST	Reactor Water Storage Tank
SRI	Senior Resident Inspector
SI	Safety Injection
T	Temperature
TS	Technical Specifications
TSC	Technical Support Center
VCT	Volume Control Tank
W	Westinghouse

ATTACHMENT 3

SEQUENCE OF EVENTS

DECEMBER 1, 1989

CENTRAL STANDARD TIME

NOTE: The following sequence times are based on a collection of the best information available during the inspection. Therefore, there may be some variances with other information provided.

0000 Initial Conditions: At the beginning of Shift 1, Unit 1 was in cold shutdown (Mode 5), reactor coolant system (RCS) was solid with the temperature at 175°F and pressure was 350 psig. Operations personnel were in the process of drawing a bubble in the PZR. Reactor coolant pumps (RCP) B and D were in operation with the PZR power operated relief valves in "cold over pressure protection" condition. 1A residual heat removal (RHR) pump (train) was in operation in the shutdown cooling mode with 1B RHR train idle and available for operation. The 1A charging (CV) pump was in normal operation with letdown coming from the RHR system. 1B RHR pump and both safety injection pumps were secured and tagged out-of-service (OOS) as required by Technical Specifications and procedures for RCS cold over pressure protection. In addition, 1A RHR pump suction valve 1RHR 8701B was tagged OOS open with power removed by procedure to assure RHR would be maintained in the event of a pressure switch malfunction.

0055 Commenced drawing a bubble in the PZR by increasing letdown flow and energizing PZR heaters per BwOP RY-5, "Drawing a Bubble in the Pressurizer."

0128 RCS pressure had increased to about 395 psig. Letdown flow was increased to stabilize pressure.

0142 Letdown flow was maximized and charging flow was minimized (to about 70 gpm) to accommodate the RCS pressure increase to 404 psig as indicated on the wide range pressure instrument. Later it was found that the 1B RHR pump suction pressure had reached 416 psig. Although unknown at the time, this is where the 1B RHR pump suction relief valve is believed to have lifted.

0144 Pressurizer level reached on scale from off scale high and was decreasing rapidly. Letdown flow was reduced to stabilize PZR level.

0145 The radwaste operator informed the control room of a significant increase in holdup tanks (HUTs) levels.

0149 Charging flow was increased to correct for the rapid drop in PZR level. Operations personnel manually swapped CV pump suction from the volume control tank (VCT) to the reactor water storage tank (RWST).

0152 Pressurizer level went off-scale low.

0153 Charging flow was increased to maximum and letdown was reduced to minimum.

0155 1B RHR train cooling was started and 1A RHR train was secured and isolation started. This is based on field reports of a relief problem in the vicinity of the 1A RHR pump suction relief valve and accepted engineering practice to assume a fault is on the operating train.

0159 Secured 1B RCP due to primary pressure dropping to less than 325 psig and the lowest pump shaft seal differential pressure. 1D RCP continued to operate throughout the event. Primary system pressure was noted to be 272 psig and later verified by computer data to be the lowest RCS pressure throughout the event.

0215 1B CV pump OOS was lifted and was placed in operation to provide additional charging flow. This resulted in an associated RCS pressure increase.

0227 A GSEP "ALERT" was declared for loss of coolant inventory beyond the capability of the makeup system.

0235 1A RHR pump suction valve OOS was lifted and the valve shut to complete isolation of the 1A RHR train and suspected leak.

0237 Nuclear Accident Report System (NARS) notification made to State of Illinois.

0245 Pressurizer level was identified as increasing on Channel LI462 and RCS pressure reached 310 psig. 1B CV pump was secured. Radwaste reported HUT levels still increasing.

0254 Pressurizer level was identified as decreasing. 1B CV pump was restarted.

0302 Pressurizer level was increasing. Charging flow was reduced to slow the rate of PZR level increase and possible thermal shock to the PZR.

0319 An operator in the auxiliary building reported evidence of flow through the 1B RHR pump suction relief valve due to noise level and associated pipe temperatures (touch).

0322 Opened and closed 1RH 8734A (1A RHR cross connect to letdown) to reduce 1A RHR train pressure for assurance that the 1A RHR pump suction relief valve was shut.

0324 Resident Inspectors were notified.

0326 ENS notification to the NRC.

0335 Unit 1 Shift Foreman reported leakage, from the vicinity of relief valve 0AB 8634 (discharge common to RHR pump suction reliefs to the HUTs). This was later determined to be from a weep hole in the side of the valve and was the source of the 30 to 50 gallons of water released to a limited area of the auxiliary building.

0342 Charging flow was increased for adjustment to maintain PZR level.

0345 An operator was stationed near the 1A RHR pump suction relief valve.

0346 1A RHR train isolation valves were opened and locally verified that there was no evidence of flow through the 1A RHR pump suction relief valve.

0349 Placed the 1A RHR train in operation by starting the 1A RHR pump.

0350 Secured the 1B RHR pump and isolated the 1B RHR train.

0352 Pressurizer level showed significant increase.

0353 Secured the 1B CV pump.

0354 A field operator reported no evidence of leakage from the 1A RHR pump suction relief valve.

0356 A field operator reported no evidence of leakage from 1B RHR pump suction relief valve.

0400 Placed the 1A RHR letdown in service.

0402 Radwaste reported HUT levels had stabilized.

0415 Manually transferred CV pump suction from RWST back to the volume control tank.

0427 GSEP control transferred to Technical Support Center (TSC).

0435 GSEP "ALERT" terminated.

ATTACHMENT 4

INTERVIEW PROTOCOL

On December 2, 1989, as part of the AIT charter, five (5) licensed operators assigned to the Unit 1 control room during the December 1, 1989 event were interviewed. The operators were interviewed singly by one team inspector for approximately 45 minutes. The interview began at noon. (As mentioned above (Section III.C), the operators had been interviewed by both Braidwood Station Management and a team from INPO prior to the AIT interview.)

General Description of the Interview

a. The team inspector and operator were introduced by a member of the licensee management. The team inspector then told the operator that the AIT was a fact finding investigation. The interview was not intended to be a formal deposition.

b. The operator was asked to give his title position during the event, and a brief narrative of his experience.

c. The operator then stated, in his own words, the sequence of events beginning with the shift turnover and ending when the plant had been stabilized.

d. The operators were asked to describe the procedures used during the event and characterize their usefulness in mitigating the event.

e. The operators were asked to describe any training they had received relevant to the event and to characterize its usefulness in mitigating the event.

f. Finally, the operators were asked if they had anything else to add or any questions for the inspector.

The description above is not intended to be a recreation of the interview. The team inspector used his judgement and discretion to pursue related topics.

ATTACHMENT 5

PERSONNEL INTERVIEWED BY CREW POSITIONS

1. Unit 1 Nuclear Station Operator (NSO) (Reactor Operator)
2. Shift Extra NSO (Reactor Operator)
3. Unit 1 Shift Foreman (Senior Reactor Operator)
4. Shift Control Room Engineer (Senior Reactor Operator and Shift Technical Advisor)
5. Shift Engineer (Senior Reactor Operator)

December 5, 1989
BW/89-3122

Messrs: T. Maiman
D. Galle
T. Kovach
J. Bitel

Subject: Potentially Significant Event Report on Premature Lifting of the 1B Residual Heat Removal Pump Suction Relief Valve

Attached is the report of a Potentially Significant Event at Braidwood Station which occurred on Friday, December 1, 1989.

If you have any questions, please contact Jerry Wagner on extension 2497.

R. E. Querio
Station Manager
Braidwood Nuclear Station

REQ/JDW/jfe
(B4332)

DVR 20-1-89-193

cc: D. O'Brien
D. Miller
L. Literski
S. Hunsader

8920z

PSE PRELIMINARY REPORT

STATION: Braidwood UNIT: 1

EVENT DATE: December 1, 1989 EVENT TIME: 0142

TITLE: Premature Lifting of the 1B Residual Heat Removal Pump Suction Relief Valve due to Apparent Incorrect Lift and Reset Settings

Initial Conditions:

MODE 5 - Cold Shutdown; Reactor Coolant System (RCS) Temperature/Pressure: 170 degrees F/350 psig; and Pressurizer was water solid. Preparations were in progress to draw a bubble in the Pressurizer. The B and D Reactor Coolant Pumps (RCP) were in operation and both trains of Residual Heat Removal (RHR) were aligned to the RCS for Shutdown Cooling with the A loop in operation. Letdown to the Chemical Volume and Control System (CV) was being supplied from the 1A RHR system.

Description:

At 0055 on December 1, 1989 in preparation for drawing a bubble, the Nuclear Station Operator (NSO) increased letdown flow from 97 to 110 gpm and energized all 3 Pressurizer Backup Heaters in accordance with the procedure. Pressurizer pressure and temperature were approximately 350 psig and 404 degrees F at the time.

At 0103 a Process Control Instrumentation Panel was removed from service for planned maintenance. This deenergized 1 of the 2 wide range pressure channels that provide indication in the Control Room.

At 0126 the NSO increased letdown flow to stablize RCS pressure which had risen to 395 psig. The NSO continued to increase letdown flow and decreased CVCS flow to a minmum. The NSO deenergized 2 of the 3 backup heaters.

At 0139 letdown flow was maximized.

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 40 psig below the specified setpoint of 450 psig +/- 1%.

At 0144 the NSO observed that the pressurizer level had returned to scale from an off scale high condition. The level was decreasing at a rapid rate. Letdown flow was reduced in an attempt to stablize pressurizer level.

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

1. Shift Supervision was notified and immediately reported to the Control Room.

2. Control room indicators that would be indicative of a leak inside the containment were monitored by the NSOs. There were no indications of leakage inside the containment.

3. The Radwaste Control Room Operator was contacted and requested to check indications for any increase in the input to Radwaste. He reported that both Recycle Hold Up Tanks (HUT) 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 pressurizer level indicated 0%.

At 0153 charging flow was increased to a maximum for the 1A CV Pump and letdown flow was isolated. Control Room Personnel 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 1A RHR train was shutdown and isolated and the 1B train was placed in operation. The NSO opened the CV pump suction isolation valve from the Refueling Water Storage Tank (RWST) and closed the outlet valves from the Volume Control Tank (VCT). An Equipment Attendant (EA) identified that a relief valve was spraying water. This was perceived to be in the same area as the 1A RHR Pump suction relief valve.

At 0200 the 1B RCP was shutdown. RCS Pressure reached 272 psig at about this time and stablized. This was the lowest pressure achieved in the RCS during the event.

At 0215 the SRO Licensed Shift Supervisors decided to return the 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 Licensed Supervisor 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 Pressurizer level had been indicating 0% for 24 minutes. There were no other pumps readily available to perform the required function.

At 0227 a GSEP was classified as an Alert based on EAL 2.n. - Primary System Leakage is beyond the makeup capabilities of available charging pumps.

At 0235 the second in-series RHR suction isolation valve for the A train was closed. This was performed due to suspected leakage of the downstream isolation valve. This suspicion was generated by reports that the rate of increase in the HUTs was decreasing. The 1B CV Pump was started.

At 0237 the NARS notification was made to declare an Alert.

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

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

At 0255 the GSEP callout tree was initiated. Due to the nature of this event the SE had to spend a significant amount of time briefing the Operating Engineer and the Assistant Superintendent of Operating prior to initiating the callout tree. As a result the 60 minute staff augmentation goal could not be met.

At 0302 the Pressurizer Cold Cal Level indication was above 0%. The NSO reduced CV pump flow by throttling a flow control valve. HUT level was still increasing.

At 0312 an Equipment Attendant (EA) was dispatched to determine if the 1B RHR Pump suction relief was leaking.

At 0319 the EA reported that the 1B RHR Pump suction relief was lifting. Operations personnel verified that the 1A RHR suction relief was not leaking. The relief valve that was spraying water was identified as the 0AB8634, a relief valve on RHR suction relief valve discharge header.

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

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

At 0350 the 1B RHR Train was shutdown and isolated. Pressurizer Level started to increase significantly.

At 0353 the 1B CV Pump was shutdown. Stable plant conditions were established.

At 0427 Comand and Control was transfered to the TSC.

At 0435 the event was terminated.

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 water level attained in the system is beleived to be the lower portion of the Pressurizer Surge Line, Reactor Vessel Level Indication read 100% throughout the entire event.

5. The Pressurizer Water Space and Surge Line Temperature Indicators experienced an indicated cool down in excess of 200 degrees F in a one hour period. This occured when pressurizer level was re-established during the event. This would be a normally expected occurance for an event of this nature. Pressurizer temperature was approximately 440 degrees F and RCS temperature was approximately 170 degrees F at the start of the event.

Apparent Cause:

The apparent cause of this event was the premature lifting of the relief valve combined with an incorrectly set blowdown ring. The improperly set blowdown ring caused the valve to remain open below the proper blowdown setting of approximately 10%. The cause of the premature lifting of the valve is still under investigation. Three lifts have been performed in the shop since its removal. The valve lifted at 411 psig , 407 psig, and 405 psig.

Safety Significance:

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.

This event was investigated by an INPO evaluation team and an NRC Augmented Inspection Team.

Corrective Actions:

Immediate corrective actions:
- Isolate letdown and increase charging to maximum
- Identify the source of the leakage
- Isolate the the RHR trains in a systematic manner
- Establish stable plant conditions
- Remove the faulty valve and replace it with one from stores
- Clean up and decon the area that was sprayed with water from the the AB relief valve
- An Engineering evaluation was conducted on the effects of the indicated cooldown rates on the Pressurizer. Based on the results of this evaluation it has been concluded that the effects were insignificant and continued operation is acceptable.
- The documentation for valve lift and reset settings for 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.

Actions to prevent recurrance:

- Actions will be taken based on the results of the investigation which is in progress. This investigation will include a root cause analysis to determine failure mode of the 1B RHR Pump suction relief valve.

Submitted by: K C Kofron
Production Superintendant

8920z

Attachment -7-

SPECIAL OPERATING ORDER

Special Operating Order No. SO-ST-0036

Effective Date 12-5-89

TITLE: RH Shutdown Cooling Pressure Limit

The maximum pressure allowed at the RH suction valves is set at 375#. This is an administrative limit, due to the set point problems on these relief valves.

Ensure the pressure is maintained within this limit whenever the RH system is aligned for shutdown cooling.

(Final)

APPROVED
JUL 24 1986
BRAIDWOOD
ON-SITE REVIEW