PSEG

Public Service Electric and Gas Company P.O. Box 236 Hancocks Bridge, New Jersey 08038-0236

Nuclear Business Unit

OCT 02 1995

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

Dear Sir:

HOPE CREEK GENERATING STATION
DOCKET NO. 50-354
UNIT NO. 1
LICENSEE EVENT REPORT NO. 95-016-01

This Supplemental Licensee Event Report entitled ''Shutdown Cooling Bypass Event-Residual Heat Removal System B Loop Flow Bypass'' is being submitted pursuant to the requirements of 10CFR50.73(a)(2)(i)(B).

Sincerely,

Mark E. Reddemann
General Manager -
Hope Creek Operations

JPP
SORC Mtg. 95-091

C Distribution
LER File

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The power is in your hands.

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95-2168 REV. 6/94

NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 04/30/98
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

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TITLE (4)
Shutdown Cooling Bypass Event - Residual Heat Removal System B Loop Flow Bypass

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 NAME	DOCKET NUMBER
										05000
07	09	95	95	-- 016 --	01	10	02	95	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	4	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
			20.2201(b)		20.2203(a)(2)(v)	X	50.73(a)(2)(i)(B)		50.73(a)(2)(viii)
POWER LEVEL (10)	0		20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
			20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
			20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
			20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
J. Clancy	(609) 339-3144

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)			EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On July 8, 1995, at approximately 1100 hours, a Shutdown Cooling Bypass Event occurred and continued until approximately 0550 on July 9, 1995. On July 10, 1995 it was determined that this bypass event rendered the shutdown cooling mode of Residual Heat Removal (RHR) inoperable which is a condition prohibited by Technical Specification (TS) 3.4.9.2. The bypass event was initiated when the operating crew left the Reactor Recirculation Pump discharge valve (1BBHV-F031B) in a partially open position to mitigate potential thermal binding. During the shutdown cooling evolution, approximately 2000 GPM of RHR heat exchanger outlet flow was diverted through the open valve and re-directed to the RHR shutdown cooling suction line. About ten hours later, bypass flow increased to approximately 4000 GPM when the valve was further opened in an attempt to re-close the valve. The valve was manually closed on July 9, at 0550 hours, terminating the event. Investigation into this event identified key corrective actions concerning operator training and procedure compliance, valve thermal binding assessment, and management response. It was determined that two Operational Condition changes occurred from Cold Shutdown to Hot Shutdown. This was not known at the time of the event. As a result of these unplanned Operational Condition changes, several TS Limiting Conditions of Operation were not met. This supplement provides an updated description of the event and corrective actions.

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PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor (BWR/4)
Reactor Coolant System, EIIS Identifier - AD

IDENTIFICATION OF OCCURRENCE

Shutdown Cooling Bypass Event - Residual Heat Removal System B Loop Flow Bypass.

Event Date: July 8,9, 1995

This is reportable under 10 CFR 50.73 (a)(2)(i)(B).

CONDITIONS PRIOR TO OCCURRENCE

Plant in OPERATIONAL CONDITION 4, (Cold Shutdown).
Reactor at 0% of rated power.

DESCRIPTION OF OCCURRENCE

On July 8, 1995, Hope Creek Generating Station was removed from service (Ref. LER 95-015) in compliance with Technical Specifications for the inoperability of the AK400 Chiller associated with the Control Room Emergency Filtration System. With the plant in Operational Condition 4, and with Residual Heat Removal(RHR) loop "B" in service, the operators periodically cycled (open and closed) Recirculation Pump discharge valves 1BBHV-F031A and 1BBHV-F031B to prevent thermal binding in accordance with Station Operating Procedure HC.OP-SO.BB-0002(Q), "Reactor Recirculation System Operation".

At 0940 hours and again at 0950 hours, on July 8, 1995, the shift attempted to stroke valve 1BBHV-F031A to avoid potential thermal binding, but the valve would not open. The cooldown proceeded, and at 1057 hours Operational Condition 4 was entered.

At 1100 hours, valve 1BBHV-F031B, which had been successfully cycled twice previously, was cracked and left open to ensure it did not bind, as was being experienced with 1BBHV-F031A. This was not in accordance with Station Operating Procedures which require opening and closing the valve. At this time, cold shutdown conditions were met. At 1152 hours, and in accordance with station procedures, the operators opened the reactor head vent valves.

At 1635 hours, the "B" RHR shutdown cooling loop was removed from service to support testing per station operating procedures. At this time, the "B" RHR recirculation loop flow was bypassing 2000 GPM of shutdown cooling flow from the reactor core. Unknown to the operators, the prior six hours (approximately) of degraded shutdown cooling had allowed temperatures higher than those indicated at the RHR heat exchanger

DESCRIPTION OF OCCURRENCE (Cont'd)

inlet to develop in the vessel, and therefore reduced boiling margin existed. When the RHR shutdown cooling loop was removed from service, the temperatures rose to the boiling point and pressure began to increase within the vessel. Pressures were greater than atmospheric for approximately 30 minutes and peaked at approximately 17 psig. The loop was returned to service at 1709 hours, and the operators observed that the "B" RHR heat exchanger inlet temperature had increased from 163 degrees F to 182 degrees F, which would not be unexpected. However, the indications were lower than the actual temperatures in the vessel due to the RHR shutdown cooling flow bypassing the vessel, which continued to affect the indicated RHR heat exchanger inlet temperatures. The pressures returned to atmospheric and the RHR heat exchanger inlet temperature returned to 163 degrees F after the "B" RHR shutdown cooling loop was restored to service.

When the drywell was determined safe for personnel access, two equipment operators entered to tag out the inboard Main Steam Isolation Valves, inspect the 1AVH212 drywell cooler for leakage, and to manually unseat valve 1BBHV-F031A. When operators attempted to unseat the 1BBHV-F031A valve (approximately 1845 hours) the valve was found to open freely. The valve was then moved electrically from the control room and positioned off the seat, indicating dual position. Upon exiting the drywell (shift turnover time) the operators reported noticing a large amount of condensation (fogging of safety glasses, visible water droplets on equipment and surfaces, etc.).

At shift turnover, the reactor coolant temperature was indicating 163 degrees F, on the RHR Heat Exchanger inlet temperature element as well as the reactor water cleanup bottom head drain temperature indicator. The problems and status associated with the 1BBHV-F031 valves were discussed by the Reactor Operators (RO) during turnover. The Senior Nuclear Shift Supervisor (SNSS) turnover took until 2000 hours due to other shift related activities. The lengthy turnover caused the SNSS to miss the shift turnover briefing. After completing his turnover, the SNSS reviewed the status of the control panels with the NSS at approximately 2030. During the review, the SNSS noticed that the 1BBHV-F031B had dual indication. The SNSS had been told of the problem with 1BBHV-F031A but only now discovered that 1BBHV-F031B was also cracked open. A 2000 GPM "B" recirculation loop flow was also observed by the SNSS. Shift management made a decision to close both recirculating pump discharge valves (1BBHV-F031A/B) at this time.

At 2045, a tagout of the Primary Containment Instrument Gas system was implemented. This removed the air supply to all drywell pneumatic loads and caused the chilled water supply valves for the drywell coolers to fail open. This provided a possible flow path from a known leak in the 1AVH212 Drywell Unit Cooler to the drywell floor drain sump.

At 2100 hours, the operators remotely closed 1BBHV-F031A, but were unsuccessful in attempting to close 1BBHV-F031B. The operators opened 1BBHV-F031B further

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DESCRIPTION OF OCCURRENCE (Cont'd)

in an attempt to close it electrically. This was based on the belief that the valve was not opened enough to make up the close permissive. A third unsuccessful attempt was made to close the valve. At this time the operators did not note that the "B" recirculation loop flow had increased to 4000 GPM from 2000 GPM.

Shortly thereafter, a slow increase in Drywell Leak Detection (DLD) flow was noticed. Previously, DLD had been a steady 0.4 GPM. The increase was attributed to the previously known leak from a cooling coil in the 1AVH212 drywell unit cooler discussed above. This was later determined to be condensate from the head vent steam condensing in the drywell.

At 0100 on July 9, 1995, a high reading on a reactor pressure trip unit (60 psig) prompted the operators to investigate the available margin to the shutdown cooling isolation trip (82 psig). Operators were concerned about an inadvertent actuation of isolation actuation instrumentation and potential loss of shutdown cooling. Following investigations by I&C technicians, voltage readings determined that pressures on all four channels were between nineteen (19) and twenty four (24) psig. The readings were attributed to either elevation head, or "zero" on the 1500 psig scale.

At 0130 hours, a tentative decision was made to enter the drywell to close the 1BBHV-F031B valve. At 0230 the SNSS cancelled that decision due to safety concerns relative to drywell conditions previously detected by the operators during their earlier drywell entry. He also wanted to wait until RHR was again secured so that 1BBHV-F031B could be stroked open fully and then closed. The reasoning was that the differential pressure (DP) across the valve was contributing to its failure to stroke.

At 0454 hours, the "B" RHR loop was secured to perform a surveillance. During this time, the operators attempted to stroke 1BBHV-F031B open fully (expecting to be able to close the valve with no DP across the valve due to RHR pump shutdown). The valve fully opened but would not close.

At 0500 hours, the Operators dispatched an electrician to the breaker, and an equipment operator to the drywell, during which time the SNSS and NSS discussed the possibility of closing the "B" recirculation pump suction valve (1BBHV-F023B) as a contingency plan. They determined that no procedural guidance was available for this and additionally expected 1BBHV-F031B to be closed very soon.

At 0508 hours the "B" RHR pump was restarted. Post event review of the "B" recirculation loop flow recorder strip chart indicated that loop flow had only slightly increased. This indicated that the cracked open 1BBHV-F031B valve was previously passing maximum flow (i.e., 4000 GPM). At 0550 hours, 1BBHV-F031B was manually closed and the RHR heat exchanger inlet temperature increased to 191 degrees F before returning to the previous value of 155 degrees F indicating that insufficient RHR flow had been circulating to the reactor core.

DESCRIPTION OF OCCURRENCE (Cont'd)

PSE&G Nuclear Engineering performed static heat balance calculations following the event. These calculations were performed to ascertain whether the Technical Specification definition of Operational Condition 3 (Hot Shutdown) had been inadvertently entered. Operational Condition 3 is defined as average reactor coolant temperature of greater than 200 degrees F. Initial calculations were based on all known heat inputs to the reactor versus heat outputs. These were determined from parameters measured and recorded during the event.

The initial calculations utilized a value of 4 GPM being discharged as saturated steam from the head vent. Based on this calculation, Operational Condition 3 (Hot Shutdown) was determined not to have been entered. However, later calculations, utilizing a verified value of 2 GPM yielded an average reactor coolant temperature of approximately 207 degrees F which is within the Operational Condition 3 definition. This calculation was finalized on August 4, 1995. Although investigations into this event initially focused on this inadvertent Operational Condition change, it was later determined that, on July 8, 1995, in between 1635 hours and 1709 hours (see previous discussion), Operational Condition 3 (Hot Shutdown) had also been inadvertently entered when the "B" RHR shutdown cooling loop had been initially removed from service.

During the first inadvertent Operational Condition change (1635 hours to 1709 hours on 7/8/95), the LCO for TS 3.6.3, "Primary Containment Isolation Valves," and the LCO for TS 3.3.2, "Isolation Actuation Instrumentation," were not met because of tagging to support the outage. As a result of not meeting the requirements of these LCOs during the inadvertent Operational Condition change, the requirements of TS LCO 3.0.4 were not met. TS 3.0.4 prohibits entry into an Operational Condition when the conditions for the LCO are not met and the associated ACTION requires a shutdown.

During the second inadvertent Operational Condition change (from 2100 hours on 7/8/95 to 0550 hours on 7/9/95), the LCO for TS 3.3.2, "Isolation Actuation Instrumentation," was not met; the LCO for TS 3.6.1.4, "MSIV Sealing System," was not met; the LCO for TS 3.6.3, "Primary Containment Isolation Valves," was not met; and the LCO for TS 3.7.1.2, "Station Service Water System," was not met due to tagging to support the outage. As a result of not meeting the requirements of the LCO for TS 3.3.2, 3.6.1.4 and 3.7.1.2 during this second inadvertent mode change, the requirements of TS LCO 3.0.4 were again not met.

APPARENT CAUSE OF OCCURRENCE

Thermal binding of the 1BBHV-F031 valves and the torque switch failure on 1BBHV-F031B were the initiating condition and the initiating equipment failure, respectively. It was also determined that the effects of these conditions were worsened by subsequent actions. The root causes consisted of procedural non-compliance, a lack of questioning attitude, not believing

APPARENT CAUSE OF OCCURRENCE (Cont'd)

indications, and a lack of follow-up regarding verification and validation of plant indications resulting in a degraded shutdown cooling condition. Contributing causes included inadequate training and industry operating experience (OEF) review.

Additional information concerning the above causes follows.

1. On three occasions operators manipulated valve 1BBHV-F031B open without procedural guidance and without determining the impact of leaving the valve open.

Primary Causal Factor - Procedural Non-compliance

Plant operating procedures HC.OP-SO.BB-0002(Q), "Reactor Recirculation System Operation" and HC.OP-SO.BC-0001(Q), "Residual Heat Removal System Operation'' provide guidance on operating their respective systems. Neither procedure allows the 1BBHV-F031A or the 1BBHV-F031B valve to remain in a mid-position indefinitely while the RHR system is in service. Thermal binding was assumed to have occurred on 1BBHV-F031A. Operators non-conservatively rationalized, that the guidance to stroke the valve allowed them to leave 1BBHV-F031B cracked open in order to meet the intent of a limitation in the recirculating water pump procedure which was put in place to prevent thermal binding.

Contributing Causal Factors - Inadequate Training/Ineffective OEF

Operators have not been trained specifically on the effect of having RHR flow bypass the core and return to the RHR pump suction via the recirculating water pump loop, although there have been several similar industry events. Events at Dresden and Oyster Creek were responded to as not being applicable to Hope Creek because of design and procedural differences.

A deficiency exists in the operators knowledge of the operation of the torque and limit switches on a Limitorque motor operated valve. The lack of full understanding coupled with prior experience (jogging valves open to make up the close permissive) prompted the operator to open the 1BBHV-F031B further in an unsuccessful attempt to enable valve operation in the close direction. This action increased the amount of flow through the "B" recirculation loop to a point where more decay heat was being produced than was being removed.

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APPARENT CAUSE OF OCCURRENCE (Cont'd)

2. Operators failed to recognize the effect core bypass flow had on decay heat removal and on the temperature indications that they were using. Several opportunities were missed to preclude or terminate the event.

Primary Causal Factor - Less than adequate Work Practices (Lack of Questioning Attitude/Not Believing Indications/Lack of Follow up)

When the SNSS noted that the 1BBHV-F031A/B valves were open and that there was flow in the "B" recirculating loop he discussed it with the NSS. The SNSS determined that RHR inlet temperature was steady at 155 degrees F and that the recirculating flow had been steady for many hours. The SNSS and NSS correctly determined that the 1BBHV-F031A/B valves needed to be closed but did not conclude that there was any urgency to close them since they believed RHR heat exchanger inlet temperature and vessel bottom head drain temperature represented reactor coolant system temperature.

When drywell leak detection flow alarmed and was increasing, operators assumed that it was due to a known leak in a drywell cooler cooling coil. When reactor pressure indications were noted and then verified to be higher than expected, operators assumed that the readings were due to either elevation head or were "zero" on the 1500 psig scale. Operators were very focused on avoiding a spurious shutdown cooling isolation due to instrument drift. However, Operators did not pursue pressure readings further once they were confident that a spurious isolation would not occur.

3. Operators failed to initiate prompt corrective action after the failure of a component to operate (1BBHV-F031B) and missed an opportunity to terminate the event.

Primary Causal Factor - Less than adequate Work Practices (Lack of Questioning Attitude/Lack of Follow Through)

After the 1BBHV-F031B failed to close on subsequent attempts there was no immediate action to involve maintenance personnel to determine whether an electrical problem existed that was impacting operation of the valve in the close direction. Involving maintenance at this time would have given the operators more information and eliminated the belief that differential pressure was keeping the valve from going closed.

Operators continued to believe the RHR heat exchanger inlet temperature and bottom head drain temperature were an accurate measure of reactor coolant temperature and did not accurately determine the priority of the valve problem.

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ANALYSIS OF THE EVENT AND SAFETY SIGNIFICANCE

This event is reportable under 10CFR50.73(a)(2)(i)(B) in that the actions taken during this event rendered the shutdown cooling mode of RHR inoperable, which is a condition prohibited by TS 3.4.9.2.

Operators were unaware that core conditions at the time of the event promoted steaming. This knowledge deficiency had minimal impact upon overall plant safety as adequate core cooling was assured by maintaining normal reactor vessel level and by the availability of the Low Pressure Emergency Core Cooling Systems (ECCS). In addition, it was determined that the reactor coolant system reached thermal equilibrium during the event. However, two of the three fission product barriers were not in place.

Primary containment purging was in progress. This containment configuration could have resulted in an inadvertent release of radiological material to the environment since the reactor head vent was open to the drywell atmosphere. In the event of a significant radiological release, the event would have been terminated when the activity exceeded the Reactor Building Ventilation System radiation monitor's setpoint thus isolating the release flow path.

A post event evaluation was performed by Radiation Protection personnel indicating that the release value was less than 10% of normal. Normal is 1/100 of the site allowable release.

Although the consequences of this event were minimal, this event is significant due to a series of Operator errors in that they failed to recognize the heat up, Operational Condition changes or the significance of these events.

As discussed previously, this event involved procedural non-compliance and resulted in a failure to meet the requirements of TS 3.0.4. The causes of the procedural non-compliance, as well as the corrective actions taken to preclude recurrence, are described in this LER. The consequences of not meeting the LCOs during the inadvertent Operational Condition changes were minimal. As discussed previously, adequate core cooling was assured by maintaining normal reactor vessel level and by the availability of Low Pressure ECCS. The inoperable service water pump had minimal safety impact since the remaining three service water pumps were operable to remove heat from the Safety Auxiliaries Cooling System. The effect of the inoperable MSIV sealing system was minimal due to reactor conditions at the time the system was removed from service and that there was no evidence of fuel damage prior to or during the event. The purpose of this system is to mitigate the consequences of a DBA LOCA, which were not the initial conditions of the event. In addition, the consequences of the inoperable Reactor Water Cleanup System containment isolation valve (BGHV-F001) was minimal since the outboard isolation valve (BGHV-F004) was operable.

ANALYSIS OF THE EVENT AND SAFETY SIGNIFICANCE (Cont'd)

The automatic containment isolation signal for the RHR shutdown cooling suction valves were also inoperable, but remote manual isolation capability remained available.

This event also resulted in a failure to meet 10CFR50.72 reportability requirements. Specifically, a four hour report was not issued in accordance with 10CFR50.72 (b)(2)(iii)(B), which requires reporting of any event or condition that alone could have prevented the fulfillment of the safety function of systems that are needed to remove residual heat

This failure was due to the failure of Hope Creek personnel to recognize, in a timely manner, that the event constituted a loss of shutdown cooling. In addition, no subsequent four hour report was generated due to an apparent lack of understanding concerning missed 1 hour/4 hour reports (post event). Poor internal and external communications and poor management followup to this event contributed to an inadequate response. Corrective actions concerning this issue are discussed later.

PREVIOUS OCCURRENCES

There were no LERs similar to this event. However, there have been other loss of Shutdown Cooling events due to system isolations including; LERs: 95-006, 94-003, 94-001, 92-014, 89-022, 89-005, 87-044, 87-043.

CORRECTIVE ACTIONS

The corrective actions for this event are described below. Many of these corrective actions have been identified in documents previously provided to the NRC or in previous communications with the NRC. These documents/communications include the following:

- LER 95-016-00, dated August 9, 1995 (LER CA #)

- Letter from Mr. L. Eliason to Mr. T. Martin dated August 4, 1995 (LRE letter CA #)

- Communications during the NRC inspection activities and the associated public exit meeting

- Discussions during the August 21, 1995 teleconference between the NRC, PSE&G and General Electric (GE)

- PSE&G Shutdown Cooling Bypass Event-Final Report, dated August 7, 1995. (Ind Rpt. CA #)

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CORRECTIVE ACTION (Cont'd)

* Discussions during the September 7, 1995 video conference between the NRC, PSE&G and GE.

References to these documents/communications are included for each corrective action listed below.

1. On July 10, an engineering team was established to determine if a Operational Condition change occurred and if shutdown cooling was operated in a degraded condition. On July 10, 1995 it was determined that shutdown cooling was inoperable due to the valve misalignment. They also concluded that the event was reportable in accordance with 10CFR50.73. (LER CA 1)

2. A primary causal factor for this event was procedure non-compliance. On July 10, Night Order Book (NOB) entries were made requiring the SNSS's to review this event with their shifts ASAP and to re-state department expectations with regard to procedure usage. Operations Management personnel have stressed their expectation of verbatim procedure compliance. In addition, the Operations Department has established a performance indicator to periodically assess the fulfillment of their expectations relating to procedure compliance. (LER CA 2 & 15 and Ind. Rpt. 4.11)

3. During the week of July 10, the acting GM contacted the training center to ensure the event, its root causes and corrective actions would be reinforced with all shift operations personnel during Segment 1 of 1995/96 Licensed Operator Requalification Training as well as with SRO Initial/Upgrade class in training at that time. Special training on this event, its root causes and corrective actions has been completed. In addition, Limitorque training has been provided to Hope Creek Operations personnel. Cycle 1 of licensed operator training will include a comprehensive review of this event including root causes and corrective actions.

   A contributing causal factor for this event was that operators failed to recognize and correctly assess reactor core conditions. Operators have not been trained regarding this type of event. The Nuclear Training Department has reviewed current training materials and revised them accordingly to ensure that shutdown cooling bypass flow events are effectively incorporated into the licensed operator training program. In addition, the Operations Department had been and will continue to measure the effectiveness of their diagnostic training (through evaluation of crew performance) and incorporate their findings into the licensed operator training program. (LER CA 3 & 16, LRE letter CA 3 and Ind. Rpt. 4.12, 4.14 & 4.16)

CORRECTIVE ACTION (Cont'd)

4. On July 15th, stand down meetings were conducted by the SNSSs with each shift to review effective tools for preventing operator errors and to review use of these tools in the context of those operating events that occurred during the forced outage. (LER CA 4)

5. On July 20th, an independent, multi-disciplined root cause team was commissioned to evaluate the event. Their findings have been evaluated and required corrective actions identified. These corrective actions are described in this LER. (LER CA 5 and LRE letter CA 18)

6. On July 31st, an extra SRO was assigned to day shifts (Monday through Friday), as an interim measure to handle some of the shift administrative burden and to make the NSS more accessible to the reactor operators to support diagnostics and oversight. This interim action was discontinued on September 18th and replaced by a Work Control Group which performs administrative tasks. (LER CA 6 and Ind. Rpt. 4.16)

7. Operating procedures have been revised to reflect lessons learned, including: 1) the minimum shutdown cooling flow required to assure adequate cooling; 2) strategies for level control while in shutdown cooling; 3) indications to be used if conflicting information develops regarding shutdown cooling parameters; and 4)the recommendations from Engineering, which included eliminating stroking of the recirculation system suction and discharge valves and guidance associated with thermal binding of the 1BBHV-F031A/B.

   In addition, the basis for maintaining level less than that required for natural circulation while shutdown cooling is in operation has been reassessed. A revision to procedure HC.OP-IO.ZZ-0004(Q) was completed on August 25, 1995, to raise the level to above 80 inches once Operational Condition 4 is entered. (LER CA 7 & 20, LRE letter CA 1, 7, 8 & 13 and Ind. Rpt. 4.17)

8. A common cause analysis team was initiated to review the recent increase in operator errors. Their evaluation has been completed, the findings have been evaluated and required corrective actions have been identified. (LER CA 9 and LRE letter 19)

CORRECTIVE ACTION (Cont'd)

9. An Operational Condition change is not easily determined when RHR heat exchanger inlet temperature is not representative of average reactor coolant temperature. Operators have been provided guidance to determine if an Operational Condition change occurs under these conditions. Guidance for determining average reactor coolant temperature has also been developed. The heat balance calculations for this event have been completed and independently verified by the NSSS vendor. (LRE letter CA 2 &10 and Ind. Rpt. 4.1 & 4.2)

10. Some licensed operators that were interviewed demonstrated a lack of knowledge regarding NUREG 1022 and Draft NUREG 1022 (Event Reporting Guidelines 10CFR50.72 and 10CFR50.73). Additional training on these documents has been provided to licensed operators.

    In addition, the Hope Creek management team experienced uncertainty regarding the company's position on 10CFR50.72 and voluntary reporting. This uncertainty appears to be the lack of understanding regarding the company's position on missed 1 hour/4 hour reports (post event). NBU management has supplied appropriate guidance regarding these issues, and the Salem and Hope Creek Event Classification Guides (ECGs) have been revised to include guidance on voluntary reporting. The revision to the ECGs obviates the need to revise the Corrective Action Program procedure, NC.NA-AP.ZZ-0006(Q). (LER CA 11 & 22 and Ind. Rpt. 4.4, 4.5, 4.6 & 4.19)

11. A formal review of this event was delayed as a result of the failure to recognize the complete significance of the event. Additional guidelines for the investigation of significant events has been developed and procedures revised accordingly. In addition, the Hope Creek General Manager has restated and reinforced his expectations regarding event review. (LER CA 12 and Ind. Rpt. 4.7)

12. Hope Creek Management failed to effectively communicate the details and significance of this event both internally and externally. Hope Creek is providing training on effective communications to station personnel. Additional guidance has also been provided concerning accurate and prompt communications to NRC personnel. (LER CA 13, LRE letter CA 17 and Ind. Rpt. 4.8)

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CORRECTIVE ACTION (Cont'd)

13. The long term solution to the existing work around issue of thermal binding for the recirculation suction and discharge valves has been re-evaluated. The valve vendor has provided specific recommendations regarding this issue and advised against cycling the valves during shutdown cooling. Procedures have been revised accordingly. The option of limit seating these valves has been determined by the NSSS vendor to be feasible. (LRE letter CA 7 & 9, LER CA 14 and Ind. Rpt. 4.9 & 4.10)

14. The event was reviewed with all Hope Creek licensed operators and the specifics of this event have been provided to Nuclear Business Unit personnel. (LER CA 17 and Ind. Rpt. 4.13)

15. It has been determined that ineffective OEF review contributed to this event. Nuclear Reliability and Assessment (NRA) has compiled industry shutdown cooling events for analysis. OEF recommendations for application at Hope Creek have been presented to the Operations Department and will be evaluated by December 1, 1995.

    In addition, the existing process of OEF review for applicability to Hope Creek has been evaluated for its overall effectiveness. Corrective actions to eliminate OEF deficiencies have been developed and are scheduled to be implemented by January 1, 1996. (LER CA 18 &19, LRE letter CA 11 and Ind. Rpt 4.15)

16. The reactor experienced bypass flow for this event for approximately 19 hours. This bypass flow was reverse flow through the "B" recirculation pump. Engineering has evaluated potential concerns with regard to damage induced by reverse flow through the "B" recirculation pump for an extended period of time and determined that there was no damage due to reverse flow. In addition, the accuracy of recirculation flow equipment relative to flow direction has been determined to be accurate in both directions. (LER CA 21, LRE letter CA 12 and Ind. Rpt. 4.18)

17. Investigations into the history of cracking valves open to prevent thermal binding have taken place. At least one past instance was found when this was done. It was determined that the requirement to stroke valves was added to a revision to procedure HC.OP-SO.BB-0002(Q), dated January 12, 1989, in response to SOER 84-7. There is no indication that the valve vendor was contacted as part of the response to SOER 84-7. (LRE letter CA 4, 5 and 6)

18. Communications have taken place with the NSSS vendor and other utilities and an Operating Experience (OE) report for this event was issued on August 15, 1995. Information received from the NSSS vendor and other utilities was used to prepare a revised OE report issued on September 22, 1995. (LRE letter CA 10)

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CORRECTIVE ACTION (Cont'd)

19. Focused control room observations have taken place to evaluate procedural compliance of shift personnel. These observations have been assessed and required corrective actions have been identified. (LRE letter CA 14)

20. The scheduling of electrical protection assembly (EPA) surveillances has been changed to ensure that the decay heat conditions are adequate for performing the test. (NRC Question from Followup Inspection)

21. An evaluation was completed to determine if the shutdown cooling isolation valves need to be cycled to demonstrate that they can be operated by remote manual operation from the control room given that bypassing the overpressure protection function and one channel of level isolation is accomplished using a key lock switch. It has been determined that removing shutdown cooling to demonstrate manual valve control is no longer warranted. (NRC Question from Followup Inspection)

22. A meeting was held with the NSSS vendor and the NRC to discuss thermal hydraulic analyses related to this event to ensure that the NRC's concerns were addressed. Completion of these thermal hydraulic analyses and issuance of a final report is scheduled for February 1, 1996. (8/21/95 telecon)

23. An evaluation of Hope Creek previous shutdown cooling events is being conducted to determine if there are any "common threads" in the causes of these events. This evaluation will be completed by October 6, 1995. (Management Initiative)