NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Hope Creek Generating Station	05000354	1 OF 10

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 --	00	08	09	95	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9): 4
POWER LEVEL (10): 0

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)
	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
X	YES (If yes, complete EXPECTED SUBMISSION DATE).	NO		10	02	95

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 in the areas of operator training, operator procedure compliance, valve thermal binding assessment, and management response to this event. It was further determined on August 4, 1995, that an Operational Condition change occurred from Cold Shutdown to Hot Shutdown. This was not known at the time of the event. As a result of this unplanned operational condition change, several TS Limiting Conditions of Operation were not met.

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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 surveillance testing. The loop was returned to service at 1709 hours, and the operators noticed the "B" RHR heat exchanger inlet temp increased from 163 °F to 182 °F. This temperature increase was expected after temporarily removing the

DESCRIPTION OF OCCURRENCE (Cont'd)

system from service. The temperature returned to 163 °F after the "B" RHR loop was returned to service.

When the drywell was determined safe for personnel access, two 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 °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 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.

DESCRIPTION OF OCCURRENCE (Cont'd)

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 secured so that 1BBHV-F031B could be stroked open fully and then closed.

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. 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 °F indicating that insufficient RHR flow had been provided to the reactor core.

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.

DESCRIPTION OF OCCURRENCE (Cont'd)

Operational Condition 3 is defined as average reactor coolant temperature of greater than 200 ° 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 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 ° F which is within the Operational Condition 3 definition. This calculation was finalized on August 4, 1995. During the inadvertent Operational Condition change, the LCO for TS 3.3.2 "Isolation Actuation Instrumentation", and 3.6.1.4 "MSIV Sealing System" were not met because of previous tagging in preparation for the outage. Specific LCOs which were not met will be addressed in a supplement to this LER.

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. 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 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 OEF review.

Also, 10CFR50.72 reportability requirements were not implemented due to the failure to recognize that the event constituted a loss of shutdown cooling. Furthermore, poor internal and external communications and poor management followup to this event conributed to an inadequate response.

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

APPARENT CAUSE OF OCCURRENCE (Cont'd)

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 then 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.

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°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.

APPARENT CAUSE OF OCCURRENCE (Cont'd)

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(LTA) 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.

SAFETY SIGNIFICANCE

Operators were unaware that core conditions 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 additon, it was determined that the reactor coolant system reached thermal equilibrium during the event.

Primary containment purging was in progress. This containment configuration could have resulted in an inadvertent release of radiological material to the environment since the 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.

SAFETY SIGNIFICANCE (Cont'd.)

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 change or the significance of these events.

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

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.

2. On July 10, Night Order Book (NOB) entries were made requiring the SNSS's to review this event with their shifts ASAP and re-stating department expectations with regard to procedure usage.

3. During the week of July 10, the acting GM contacted the training center to ensure the event, its root causes and corrective actions are reinforced with all shift operations personnel during Segment 1 of 1995/96 Licensed Operator Requalification Training as well as the current SRO Initial/Upgrade class. Additionally, this event will be included in initial Licensed Operator training.

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.

5. On July 20th, an independent, multi-disciplined root cause team was commissioned to evaluate the event.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

CORRECTIVE ACTION (Cont'd)

6. On July 31,1995, an assignment of an extra SRO to day shifts (Monday through Friday) was made, as an interim measure to handle some of the shift administrative burden.

7. On July 30, the system operating procedures were revised based on engineering input regarding manipulation of the recirculation system suction and discharge valves. Other operating procedures will be revised based on Technical Engineering's guidance associated with thermal binding of the 1BBHV-F031A/B.

8. The minimum flow requirements to prevent Reactor Pressure Vessel stratification and to ensure adequate temperature indication will be determined and incorporated into the appropriate procedures.

9. A common cause analysis team has been initiated to review the recent increase in operator errors. Their evaluation will be completed by August 30, 1995. Additional corrective actions will be taken based upon their findings.

10. An Operational Condition change is not easily determined when RHR heat exchanger inlet temperature is not representative of average reactor coolant temperature. Operators will be provided guidance to quickly determine if an Operational Condition change occurrs under these conditions.

11. Some licensed operators 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 will be provided to licensed operators and other appropriate NBU personnel.

12. A formal review of this event was delayed as a result of failure to recognize the complete significance of the event. Additional guidelines for the investigation of significant events will be developed.

13. Hope Creek Management failed to effectively communicate the details and significance of this event both internally and externally. Hope Creek will provide effective communications training to Station personnel.

14. The long term solution to the existing thermal binding issue for the recirculation suction and discharge valves will be re-evaluated.

CORRECTIVE ACTION (Cont'd)

15. A primary causal factor for this event was procedure non-compliance. Operations Management personnel have already stressed their expectation of verbatim procedure compliance. The Operations Department will periodically assess the fulfillment of their expectations relating to procedure compliance.

16. 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 will review current training materials and revise them accordingly to ensure that shutdown cooling bypass flow events are effectively incorporated into the licensed operator training program.

17. The specifics of this event will be provided to Nuclear Business Unit personnel.

18. It has been determined that an ineffective OEF review contributed to this event. Nuclear Reliability and Assessment (NRA) will coordinate with the Operations Department and perform another review of all industry loss of shutdown cooling events. The review will include lessons learned for application at Hope Creek.

19. The existing process for OEF review for applicability to Hope Creek will be evaluated for its overall effectiveness.

20. Current operating procedures will be revised to reflect lessons learned. This will include: 1) the minimum shutdown cooling flow required to assure adequate cooling, 2) strategies for level control while in shutdown cooling, and 3) indications to be used if conflicting information develops regarding shutdown cooling parameters.

21. The reactor experienced bypass flow for this event for approximately 19 hours. This bypass flow was reverse flow through the "B" recirculation Pump. Engineering will evaluate potential concerns with regard to damage induced by reverse flow through the "B" recirculation pump for an extended period of time.

22. 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 will supply appropriate guidance regarding these issues.