UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
475 ALLENDALE ROAD
KING OF PRUSSIA, PENNSYLVANIA 19406-1415

August 3, 1995

MEMORANDUM TO: James T. Wiggins, Director
Division of Reactor Safety

FROM: Thomas T. Martin
Region Administrator
Region I

for TTM

SUBJECT: SPECIAL TEAM INSPECTION CHARTER FOR REVIEW OF A JULY 1995 HOPE CREEK SHUTDOWN COOLING FLOW BYPASS EVENT

On July 8, 1995, Hope Creek operators aligned reactor recirculation system valves in a manner that was contrary to procedures and resulted in significantly reduced shutdown cooling to the core due to diversion of flow from the Residual Heat Removal (RHR) system. Since the potential safety significance of this event was not immediately identified by licensee management, and weaknesses in both management and operator performance are evident, I have determined that a Special Team Inspection (STI) should be conducted to review and evaluate the circumstances, safety significance, and generic implications that are associated with this event.

Accordingly, the Division of Reactor Safety (DRS) is assigned the responsibility for the overall conduct of this Special Team Inspection. Jim Trapp, DRS, is appointed as Special Team Inspection Leader (Other STI members are identified in Attachment 2). The Division of Reactor Safety (DRS) is assigned the responsibility for coordinating with other NRC offices, as appropriate. Further, the Division of Reactor Safety is responsible for the timely issuance of the inspection report, and the identification and processing of potentially generic issues. The Division of Reactor Projects is responsible for processing any enforcement action that results from the findings of this STI.

Attachment 1 represents the charter for the Special Team Inspection and details the scope of the inspection. The inspection shall be conducted in accordance with Attachment 1. This charter may be amended with my verbal approval, if necessary, to further assess the licensee's actions on this matter.

Docket Nos. 50-354

Attachments:
1. Special Team Inspection Charter
2. Team Membership

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PDR ADOCK 05000354
Q PDR

Distribution w/atts:
J. Taylor, EDO
J. Milhoan, OEDO
W. Russell, NRR
R. Zimmerman, NRR
S. Varga, NRR
J. Stolz, NRR
B. Dean, OEDO
W. Kane, DRA, RI
R. Cooper, DRP, RI
W. Lanning, DRP, RI
J. White, DRP, RI
W. Hehl, DRSS, RI
S. Shankman, DRSS, RI
S. Barber, PE, RI
D. Jaffe, PD I-2, NRR
V. Driggs, PAO, RI
D. Chawaga, SLO, RI

ATTACHMENT 1

HOPE CREEK SHUTDOWN COOLING BYPASS EVENT

SPECIAL TEAM INSPECTION (STI) CHARTER

The general objectives of this STI are to:

1. Verify and validate that the licensee conducted a thorough and systematic review of the circumstances surrounding the July 8, 1995 Shutdown Cooling Flow Bypass Event, and ensured a detailed sequence of events exists for this event.

2. Review and evaluate the adequacy of the licensee's root cause and corrective actions for the July 8, 1995 Shutdown Cooling Flow Bypass Event.

3. Assess the operators' actions preceding, during, and subsequent to the event, including the evaluation made by the Safety Review group that was presented to senior Hope Creek management on July 12, 1995. Consider licensee management decision-making relative to this evaluation, especially the delay in initiating their internal investigation until July 20, 1995. Further, assess the quality and timeliness of the licensee's communication of this matter to the NRC.

4. Assess and evaluate whether this event should have been formally reported to the NRC.

5. Review and evaluate the adequacy of both initial and operator requalification training for abnormal conditions and transients while in the SDC mode of operation. Consider both classroom and simulator training effectiveness.

6. Review and evaluate the licensee's methods for monitoring relevant parameters and assessing conditions while in cold shutdown. Consider whether an unintentional mode change resulted from this event, and evaluate the appropriateness of the temperature indications used to determine mode change.

7. Evaluate operator knowledge, understanding, and implementation of routine, abnormal, administrative, and other relevant procedures. Consider written and verbally communicated management expectations and policies that may have contributed to this event.

8. Assess the safety significance of this event. Consider the worst case postulated scenario for this event assuming no operator action or intervention. With the worst case scenario underway, determine the additional failures (and the existing barriers, if any, to those failures) that would have to occur before a significant offsite release would result.

ATTACHMENT 2

HOPE CREEK STI MEMBERSHIP

Jim Trapp, STI Leader, Team Leader, Division of Reactor Projects, Region I

Robert Summers, Senior Resident Inspector, Hope Creek, DRP, RI

Tracy Walker, Senior Operations Engineer, Division of Reactor Safety, RI

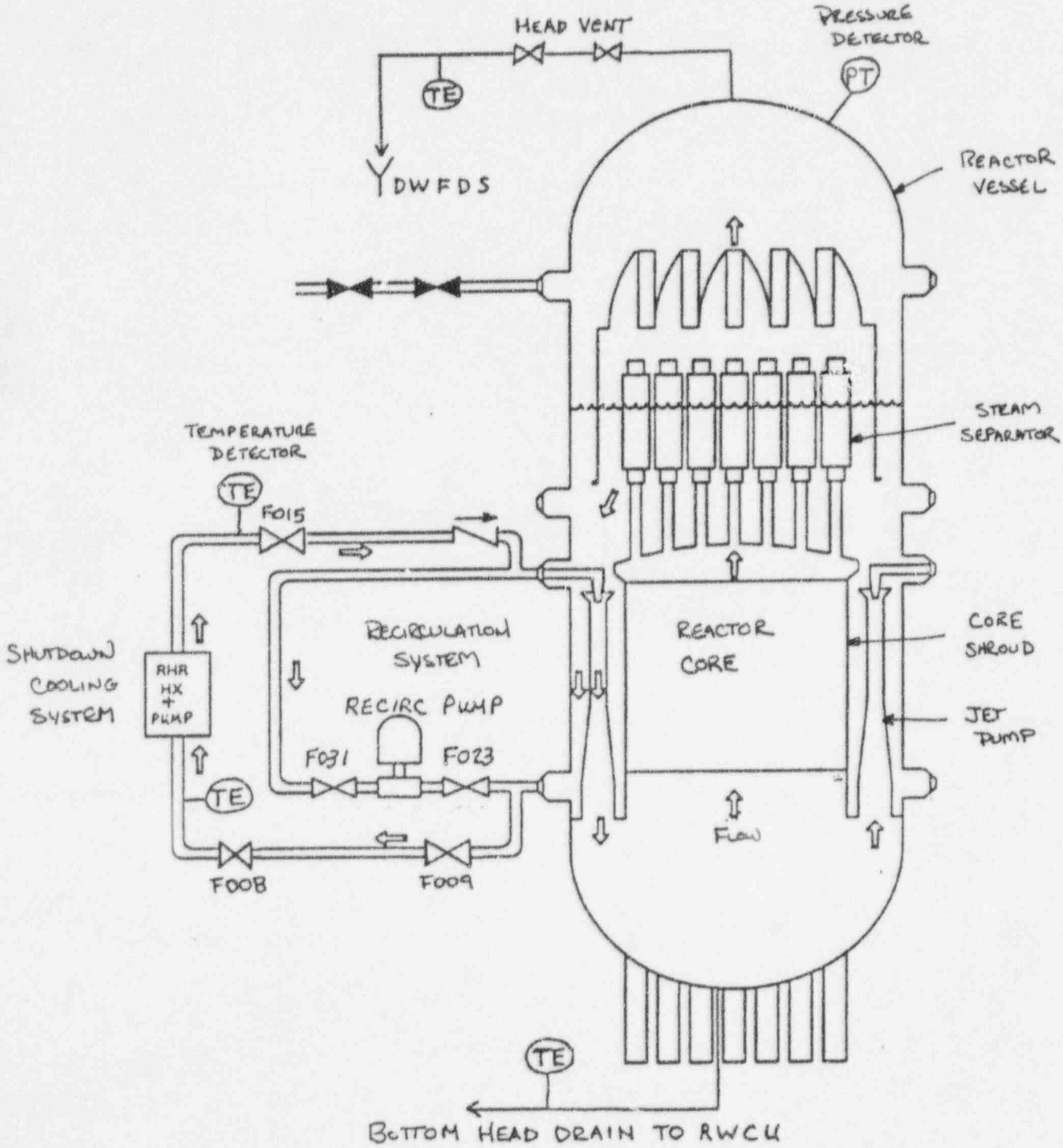
Dave Jaffe, Hope Creek project manager, PD1-2, NRR

George Thomas, Reactor Systems Branch, NRR

Other NRC personnel, consultants, or contractors will be engaged in this STI, as needed.

ATTACHMENT 2

SHUTDOWN COOLING - RECIRCULATION SYSTEM INTERFACE



ATTACHMENT 3

HOPE CREEK

AUGUST 24, 1995

EXIT MEETING

SLIDES

HOPE CREEK
SPECIAL TEAM INSPECTION

NRC EXIT MEETING

August 24, 1995

J. T. Wiggins, Director
J. M. Trapp, Senior Reactor Engineer

AGENDA

1.	Introduction	James T. Wiggins
2.	Exit	James M. Trapp
3.	Licensee Comments	Mark Reddemann
4.	Future Staff Actions	James T. Wiggins
5.	Closing Remarks	James T. Wiggins

SPECIAL TEAM INSPECTION
HOPE CREEK

NRC EXIT MEETING

AUGUST 24, 1995

NRC INSPECTION REPORT 50-354/95-81

TEAM MEMBERS

Jim Trapp, Region I, Inspection Team Leader
Tracy Walker, Region I, Operation Engineer
Ray Lorson, Region I, Resident Inspector - Peach Bottom
Dave Jaffe, HQ, Sr. Project Manager, Hope Creek
Dave Desaulniers, HQ, Human Factors Branch
George Thomas, HQ, Reactor Systems Branch

INSPECTION DATES

August 7-11, 1995 and August 16, 1995

INSPECTION SCOPE

Conduct an independent evaluation of the circumstances surrounding the July 7-9, 1995 partial bypass of shutdown cooling flow from the Hope Creek reactor vessel.

TEAM FINDINGS

1. BACKGROUND

2. SEQUENCE OF EVENTS

3. SIGNIFICANT TEAM FINDINGS

   - OPERATOR PERFORMANCE

   - PROCEDURES AND TRAINING

   - POST EVENT EVALUATION

   - SAFETY SIGNIFICANCE

BACKGROUND

Brief Description of the Operation of the Shutdown Cooling and Recirculation Systems During Cold Shutdown Conditions at Hope Creek

- In cold shutdown (Mode 4), the reactor is in a shutdown condition and the average reactor coolant temperature is required by Technical Specifications to be maintained below 200°F. One of two loops of the residual heat removal (RHR) system is in service to remove decay heat from the core. The recirculation pump suction and discharge valves are normally closed. At Hope Creek, reactor vessel water level is maintained near the normal operating level and the recirculation system valves are periodically stroked (alternately opened/closed) to prevent thermal binding of those valves. See the attached drawing for the system flowpaths during the July 7-9 timeframes.

BACKGROUND (Continued)

Past Performance Associated With Losses of Shutdown Cooling (SDC) was Poor

- Shutdown Cooling lost 10 times since 1987.
- 3 of last 5 losses of SDC had procedure adherence issues.
- PSE&G's 1995 Safety System Functional Inspection concluded that the loss of SDC has become so common that operators expect it to occur.
- Last event in March 1995 was caused by lack of procedure adherence.
- An opportunity was missed to resolve this issue
- Earlier opportunities to assess the decay heat removal issues were not adequate. In March 1995 valve FO31B thermally bound and in 1994 a control room observation noted operators leaving the recirculation pump discharge valves cracked open. Appropriate procedure enhancements were not made at that time which could have prevented this event.

SEQUENCE OF EVENTS

July 7, 1995

6:30 p.m.	Plant shutdown due to technical specification requirements.

July 8, 1995

00:18 a.m.	Mode switch placed in the "shutdown" position, plant in hot shutdown condition (Mode 3).
7:54 a.m.	The B residual heat removal (RHR) pump placed inservice to establish shutdown cooling (SDC).
7:54 a.m. - 9:40 a.m.	The A and B recirculation pump discharge valves (F031A and F031B) were stroked to prevent thermally binding.
9:40 a.m.	Valve FO31A would not open. (Valve thermally bound closed.)
10:57 a.m.	The plant is placed in the cold shutdown condition (Mode 4). (Average reactor coolant temperature less than 200°F.)

SEQUENCE OF EVENTS (Continued)

11:00 a.m.	Valve F031B is partially opened to prevent thermally binding. (Not in accordance with station procedures. Diversion of SDC flow begins.)
11:52 a.m.	Reactor head vent opened.
4:35 p.m.	The SDC system is secured in accordance with plant procedures to test RHR system valves. Securing the SDC system resulted in vessel pressurization and the first of 2 mode changes. (This mode change was not identified by the plant operators or during the subsequent evaluation of the event by PSE&G.)
5:09 p.m.	SDC system returned to service in accordance with plant procedures. (This action cools the reactor vessel and the plant returns to the Mode 4 cold shutdown condition.)
5:30 p.m	Operators enter the drywell to perform outage activities and to investigate the reason for the FO31A valve failure. (Refer to the 9:40 a.m. event.)

SEQUENCE OF EVENTS (Continued)

6:45 p.m.	Operators manually "crack" open the FO31A valve. (This valve had thermally bound closed at 09:40 a.m.)
	Upon exiting the drywell, plant operators reported condensation on drywell surfaces and also that their glasses were "fogging" while inside the drywell.
8:30 p.m.	The night shift senior reactor operators (the SNSS and NSS) decided to close the FO31A and FO31B valves.
9:00 p.m.	Valve FO31A was closed.
	Valve FO31B would not close. (This was later determined to be caused by a failure of a component in the valve operator.)
	Valve FO31B was opened for an additional two to three seconds and then operators again attempted unsuccessfully to shut the valve. (Opened because of a misconception of the control logic for this valve. This action increase bypass flow and starts the plant heatup.)

SEQUENCE OF EVENTS (Continued)

10:00 p.m.	Reactor pressure was above O psig and increasing, which indicated that a second mode change from Mode 4 to Mode 3 had occurred.

July 9, 1995

1:30 a.m.	The operating crew developed a plan to enter the drywell and to manually shut Valve FO31B.
2:30 a.m.	SNSS cancels the plan to enter the drywell due to personnel safety concerns associated with "footing" in the drywell.
4:54 a.m.	SDC was removed from service to test RHR system valves.
	Valve FO31B was opened fully. **(Opened because of a misconception of the control logic for this valve.)**
	Valve FO31B would not close.
5:08 a.m.	SDC place back in service with FO31B fully open.
5:50 a.m.	FO31B manually shut. **(Shutdown cooling restored and the event was terminated.)**

SIGNIFICANT TEAM FINDINGS

- OPERATOR PERFORMANCE
- PROCEDURES AND TRAINING
- POST EVENT EVALUATION
- SAFETY SIGNIFICANCE

OPERATOR PERFORMANCE

Onshift Communications And Command And Control During This Event Were Less Than Adequate

- Day shift supervision was unaware that valves FO31A & B were cracked open.
- Night shift supervision was not aware that in an attempt to close FO31B that the valve was opened more.
- Onshift supervision failed to clearly communicate the importance of closing the valves.
- The operators did not use technical support or maintenance staff to troubleshoot valve problems.
- The narrative logs did not adequately document the bypass of SDC.

Procedure Usage By Operators Was Inadequate

- The procedural instructions provided for stroking the valves were not implemented.
- Some operators did not review the limitation in the recirculation procedure that directed stroking the valves.

OPERATOR PERFORMANCE (Continued)

Monitoring and Assessment of Plant Indications By Plant Operators Were Inadequate

- Operators did not monitor recirculation flow following leaving the recirculation discharge valves open.

- Operators did not adequately assess the available indications to determine plant conditions.

Plant Operators Should Have Concluded a Mode Change Had Occurred

- The operators had adequate plant indications to conclude that a mode change had occurred.

PROCEDURES AND TRAINING

Procedure Quality Was Less Than Adequate

- The guidance for stroking valves was inconsistent.
- The procedure step provides inadequate guidance and is open to interpretation.
- Implementing this step as a limitation is inappropriate.
- Inadequate procedural guidance for determining average reactor temperature.

Operator Training Was Less Than Adequate

- Training on operating concerns associated with bypass flow was ineffective. After opening the recirculation valves the operators did not adequately monitor key plant indications to verify decay heat removal.
- Some relevant operating experience applicable to aspects of this event had not been incorporated into training.

PROCEDURES AND TRAINING (Continued)

- Training on operational characteristics during shutdown cooling was inadequate. Operators were not provided adequate training on monitoring reactor coolant system temperature or other indications with the RHR system out of service or degraded.

- Operators were not knowledgeable of the control logic for valves FO31A & Bs.

- Operators had different understandings of thermal binding mitigation strategies.

POST EVENT EVALUATION

Plant Management Did Not Correctly Assess the Significance of this Event

- Plant management's initial response to the event was not consistent with the significance of this event.

- The improper assessment of this event led to a 10 day delay in initiating a comprehensive root cause evaluation.

- The final root cause evaluation initiated on July 20, 1995, identified most of the significant performance issues. However, some noteworthy elements were not addressed. For example:

    - The evaluation did not identify or evaluate the first mode change.

    - The evaluation did not identify weak operator communications as a contributing cause.

    - The evaluation did not identify procedure inadequacy as a root cause.

- Regardless, the corrective actions stated in the LER did address the significant performance deficiencies.

POST EVENT EVALUATION (Continued)

Independent Oversight Organization Performance was Mixed

- Good initial findings by Quality Assurance and the Safety Review Group.
- The follow-up to convince plant management of the importance of this event was not adequate.

Event Reporting to the NRC was Inadequate

- This event required a 4-hour notification (50.72) to the NRC that was not made.
- An Licensee Event Report (50.73) was appropriately provided to the NRC.

SAFETY SIGNIFICANCE

Plant Instrumentation and Procedures Did Not Provide Adequate Guidance For Determining and Maintaining Average Reactor Coolant System (RCS) Temperature

- During normal shutdown conditions (no bypass flow) RHR heat exchanger inlet temperature is a valid indication of average RCS temperature.

- The bypassing of SDC flow seriously degraded the licensee's ability to monitor average reactor coolant temperature.

- Adequate plant indication of average RCS temperature is not available when the residual heat removal pumps are secured.

SAFETY SIGNIFICANCE (Continued)

Several Technical Specifications Were Not Complied With Due to the Inadvertent Mode Change

- Both trains of residual heat removal shutdown cooling mode of operation were not operable during cold shutdown.

- The residual heat removal system overpressure protection was blocked for greater than one hour with the reactor above cold shutdown.

- Main steam isolation valve sealing steam was not inservice with the reactor above cold shutdown.

- A RHR containment isolation valve was inoperable for more than 4 hours with the reactor above cold shutdown.

SAFETY SIGNIFICANCE (Continued)

This Event Was Safety Significant

- The safety consequences of this event were minimal and this event had no direct adverse effect on the health and safety of the public.

- This event was safety significant.

- Two of 3 primary fission product barriers were not in place or were degraded with the reactor conditions above the CSD condition. There were adequate backup systems available to protect the third fission product barrier (fuel cladding).

- The third fission product barrier (Fuel Cladding) appeared adequately protected, based on no unusual indications of fuel performance. However, further review by PSE&G and NRC continues into the Hope Creek-specific and generic implications of the event. Analyses are needed to confirm our understanding and to determine the extent of safety margin that remained.

SAFETY SIGNIFICANCE (Continued)

- The identified human performance issues, such as the poor shift communication and the failure to follow procedures are also safety significant.
- The team also identified that securing RHR to test the manual isolation function of the RHR suction and discharge valves shortly after plant shutdown may present an unnecessary challenge to the operators' ability to maintain the plant in the cold shutdown condition.

LICENSEE COMMENTS

<u>FUTURE STAFF ACTIONS</u>

1. **Issue Report**

2. **Review Report For Potential Enforcement Actions**

3. **Identify and Respond to Generic Issues**