UNITED STATES
# NUCLEAR REGULATORY COMMISSION
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SEP 18 1995

MEMORANDUM TO: Richard W. Cooper II, Director
Division of Reactor Projects, Region I

Ellis W. Merschoff, Director
Division of Reactor Projects, Region II

William L. Axelson, Director
Division of Reactor Projects, Region III

James E. Dyer, Director
Division of Reactor Projects, Region IV

Roy P. Zimmerman, Associate Director
for Projects

Ashok C. Thadani, Associate Director
for Technical Review

James Lieberman, Director
Office of Enforcement

Charles E. Rossi, Director
Division of Safety Programs
Office for Analysis and Evaluation
of Operational Data

FROM: R. W. Borchardt, Chief R. W. Borchardt
Inspection Program Branch
Division of Inspection and Support Programs
Office of Nuclear Reactor Regulation

SUBJECT: JANUARY 1996 SENIOR MANAGEMENT MEETING PLANT WRITE-UPS

This memorandum provides guidance for preparing inputs for the individual plant write-ups that will be included in the January 9-10, 1996, Senior Management Meeting (SMM) Executive Summary notebooks. Inputs will be required for the plants that are selected at the upcoming screening meetings for discussion at the SMM.

As in the past, each plant's write-up will consist of a Narrative Summary and a Data Summary, although the formats of these summaries have been modified. The write-ups have been streamlined by the elimination of background information that was of limited value in SMM deliberations, and the revised formats place increased emphasis on plant safety performance. These changes

CONTACT:
Patrick Castleman, NRR
415-3747

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should enhance the write-ups' usefulness to the senior managers in making Watch List determinations and in trending plant performance over a number of SMM cycles.

Attachments 1, 2, and 4 provide the formats for the write-ups. The organizations responsible for inputs to various sections are indicated in the left-hand margins of these attachments, and you should provide PIPB with inputs only for those areas for which you have lead responsibility. Supporting organizations are requested to closely coordinate with the lead organizations to ensure the timely inclusion of complete information.

Attachment 1 provides the outline to be used for Narrative Summary inputs for plants that were discussed at the last SMM and are to be discussed again. Attachment 2 is the outline to be used for Narrative Summary inputs for plants that will be discussed for the first time. The major changes to the Narrative Summaries consist of presenting plant information by SALP functional area, the addition of assessment/analysis paragraphs, and the direct inclusion of pertinent performance oriented data that had previously been included in the Data Summary section. The Narrative Summaries should provide an integrated and objective view of plant performance. Each regional office should base the Narrative Summaries on the output of the most recent PPRs, incorporating any insights from the screening meetings and emergent events and issues. Attachment 3 is a sample Narrative Summary for a hypothetical plant.

Attachment 4 provides a Data Summary outline. This section has been reduced to consist only of PRA information and enforcement history. The other information that previously had been presented in this section (operational performance, control room staffing, requalification program observations, significant MPAs or plant unique issues, and plant physical status) should now be discussed in the Narrative Summary under the appropriate SALP functional area, and only to the extent that they provide insights that relate to plant performance issues.

Attachment 5 is the superior performance evaluation matrix. Regional offices should complete evaluation matrices only for those plants that have had SALP reports issued since the June 1995 SMM and were rated Category 1 in all four functional areas.

As was done for past Executive Summary notebooks, AEOD is requested to provide the latest available performance indicator information for the plants to be discussed at the SMM.

Please prepare the appropriate inputs in accordance with the formats in Attachments 1, 2, 4, and 5. Send the documents electronically to P. Castleman (PIC) of my staff. To allow for the timely assembly of the notebooks, your inputs should be submitted by November 27, 1995.

The revisions to the plant writeups described in this memorandum are intended to enhance the effectiveness of the SMM process. Any comments or suggestions on these revisions should be provided to P. Castleman.

Attachments: 1. Narrative Summary Input for Plants Discussed at the Last SMM
2. Narrative Summary Input for New Plants of Concern
3. Sample Narrative Summary
4. Data Summary Input
5. Superior Performance Evaluation Matrix

cc: W. Russell
F. Miraglia
D. Crutchfield
F. Gillespie
S. Varga
J. Roe
B. Sheron
G. Holahan
B. Boger
T. Martin, RI
S. Ebneter, RII
H. Miller, RIII
J. Callan, RIV

## NARRATIVE SUMMARY INPUT FOR PLANTS DISCUSSED AT THE LAST SMM

| | |
|---|---|
| | **I. HISTORY** |
| REGION | Briefly describe when and why the plant was first discussed by senior managers. Discuss the plant's performance history, including, if applicable, its Watch List status since it was first discussed at a SMM. Refer to the narrative summary prepared for previous SMM and condense where possible. |
| | **II. PERFORMANCE SUMMARY AND ASSESSMENT** |
| REGION*/ PROJECTS/ AEOD as applicable | Briefly describe the changes in licensee performance since the last SMM. Discussions should be presented in "bullet" format by SALP functional area. Objective data for bullets should include such items as plant events, hardware issues that may impact availability and consequently affect risk, future major plant modifications, enforcement actions, latest SALP ratings, DET results, inspection findings, TS/LCO use, and requests for NOEDs. |
| | This section should be derived directly from PPRs and the preparations for the screening meetings, and should include events and issues that have emerged since the screenings. |
| REGION | Conclude each functional area's objective data presentation with a narrative assessment and analysis paragraph. These paragraphs should flow from the objective information, and should: assess and integrate overall performance (including licensee self-assessment and corrective action); discuss root causes of identified issues, and; articulate emerging concerns. |
| | **III. FUTURE ACTIVITIES** |
| REGION/ PROJECTS | Summarize planned or anticipated major inspections, enforcement conferences, and management meetings. Include current outage schedules with major plant modifications and program upgrades. |
| | Summarize significant ongoing, planned or anticipated licensing initiatives (i.e., TS upgrades and changes for plant modifications). |
| | Identify any other licensee initiative or NRC activity that should be considered at the SMM. |

In case of multiple assignment, the organization with lead responsibility appears first.

## NARRATIVE SUMMARY INPUT FOR NEW PLANTS OF CONCERN

### I. BASIS FOR CONCERN

REGION*/ PROJECTS

Briefly describe why the plant is being discussed. Describe events, weaknesses, declining trends or other information that supports the basis for concern. Provide a causal analysis of this data and evaluate the plant's departure from previous performance.

### II. PERFORMANCE SUMMARY AND ASSESSMENT

REGION/ PROJECTS

Briefly describe licensee performance. Discussions should be presented in "bullet" format by SALP functional area. Objective data for bullets should include such items as plant events, hardware issues that may impact availability and consequently affect risk, future major plant modifications, enforcement actions, latest SALP ratings, inspection findings, TS/LCO use, and requests for NOEDs.

This section should be derived directly from PPRs and the preparations for the screening meetings, and should include events and issues that have emerged since the screenings.

REGION

Conclude each functional area's objective data presentation with a narrative assessment and analysis paragraph. These paragraphs should flow from the objective information, and should: assess and integrate overall performance (including licensee self-assessment and corrective action); discuss root causes of identified issues, and; articulate emerging concerns.

### III. FUTURE ACTIVITIES

REGION/ PROJECTS

Summarize planned or anticipated major inspections, enforcement conferences, and management meetings. Include current outage schedules with major plant modifications and program upgrades.

Summarize significant ongoing, planned or anticipated licensing initiatives (i.e., TS upgrades and changes for plant modifications).

Identify any other licensee initiative or NRC activity that should be considered at the SMM.

* In case of multiple assignment, the organization with lead responsibility appears first.

REVISED SMM NARRATIVE SUMMARY

**NUCLEAR POWER PLANT X**

## I. HISTORY

Plant X was first placed on the NRC Watch List as a Category 2 plant in June 1990. A diagnostic evaluation team (DET) assessment followed in October 1990. During the Unit 2 refueling outage in September 1991, a continued performance decline was evident, particularly in the licensee's inability to effectively manage emergent work activities. Plant X remained on the NRC Watch List due to inattention to detail in maintenance and operations activities, ineffective management control, procedural inadequacies, failure to follow procedures, training inadequacies in maintenance, and hardware reliability concerns.

Through the first half of 1992, performance at Plant X improved in most areas, but improvements were uneven and overall progress was slow. In September of 1992, two significant plant events occurred: Units 1 and 2 simultaneously tripped due to multiple operator errors, and the Unit 1 auxiliary feedwater system failed due to poor maintenance. Following these events, management shut down both units while resources were focused to establish new performance standards. Initially, positive results were seen, including a reduction in the number of operator work-arounds, the establishment of clear management expectations of plant staff, a reduction in plant contaminated areas, and an error-free restart of both units. However, the momentum started through these efforts was not sustained, and performance again began to decline just after the January 1994 SMM.

## II. PERFORMANCE SUMMARY AND ASSESSMENT

Operations

- On March 3, the 1A emergency diesel generator was declared inoperable due to air start system problems. On March 10, the licensee commenced a Unit 1 shutdown when the diesel could not be made operable within the technical specification (TS 3.9.B.21) limiting condition for operation. The unit startup was delayed several times for various reasons, including boric acid heat tracing failures, steam leaks in the main steam and feedwater systems, nuclear engineer performance issues, and poor control room operator attentiveness. On March 25, Unit 1 was synchronized to the grid.
- Examples of personnel errors included:
  - Inadequate procedures, poor training on procedure changes, and personnel error resulted in the Unit 1 containment leakage exceeding TS limit since January 6. (1/13/95)
- A severity level III violation and $100,000 Civil Penalty were issued in February 1995 due to procedural and technical specification violations associated with the restart of a reactor coolant pump and breach of containment integrity.

PLANT X

- Several initiatives undertaken to improve personnel performance include:
  - Defining Procedure Adherence: In April, the licensee conducted a stand-down of all departments to discuss management's expectations for procedure adherence. This resulted in numerous procedure changes as workers determined that many procedures lacked information or were incorrect.
- In early March, the licensee was assessed by several outside entities: the Board of Directors, Safety Management Review Board, and an NRC Operational Safety Team Inspection (OSTI). These assessments confirmed that momentum had stagnated and fundamental changes were needed. Some changes in attitude and improved accountability have been noted. Personnel errors connected with procedural adherence, corrective actions, and engineering support activities continued.
- Weaknesses were also noted in operations management's ability to apply various technical specification requirements. Specifically, the licensee failed to enter a concurrent 24 hour shutdown LCO; inadvertently entered TS 3.0.3 when operations staff made the safety related A train 4160Vac distribution system and the B train EDG inoperable simultaneously; and manually tripped the reactor due to a misinterpretation of TS 3.9.B.27.m.

**Assessment/analysis:** Performance at Plant X has been weak, as evidenced by procedural adherence problems, poor system configuration control, and operator inattention to critical plant parameters. Although some improvements in management involvement and effectiveness were noted, they were inconsistent, and the rate of personnel errors by licensed operators remained high. Despite the fact that safety focus was found to be adequate during the OSTI, the effectiveness of improvement initiatives declined, as evidenced by poor decision-making and diminished attention given to the identification and resolution of work-arounds. Procedural adherence was poor and some procedures were difficult to follow; these weaknesses were exacerbated by inadequacies in the training of operators.

Maintenance

- Recent material condition problems:
  - An electrical fault occurred on the 1A SI pump motor. The motor had been scheduled for preventive maintenance in April 1995. (2/8/95)
  - Steam generator 2C atmospheric dump unexpectedly ramped fully open at full power, causing an increase in reactor power to greater than technical specification limits. The root cause has yet to be determined. (4/18/95)
  - Several air operated valves in Unit 1 drifted closed due to air system leaks. (2/27/95)
  - RHR pump 1B was declared inoperable due to a broken room cooler service water return valve. This was a repeat failure. (3/6/95)
- To improve plant material condition, the licensee established a quality team to prioritize material condition deficiencies across all plant departments. This recent effort has not yet yielded tangible results.

PLANT X

**Assessment/analysis:** Some improvements in the overall material condition of the plant were made, but the underlying problems with assessing and correcting plant material condition deficiencies have not been effectively addressed. Equipment failures continue to challenge the reliable operation of the facility, as evidenced by the number of plant events that have resulted from improper maintenance, poor material condition, and personnel errors by maintenance staff. Continuing problems were noted with the work control process, personnel performance, and interdepartmental communications. Some improvement in problem identification was noted, but the resolution of issues remained weak. Management's actions to improve worker standards were considered positive.

Engineering

- A design deficiency resulted in a potential for some containment isolation valves to reopen under certain conditions. (3/13/95)
- Some improvements were implemented, including:
  - Establishing material condition as a site focus area with the site engineering manager as the sponsor.
  - System engineers performing a review of all work requests to assign these requests to a work window.
- Nuclear engineers improperly performed physics calculations for the newly installed Unit 2 core, resulting in a significant delay in returning the unit to service. (4/95)

**Assessment/analysis:** Work quality suffered from procedure inadequacy, poor management reviews, large work backlogs and low engineering expectations and standards. Missing references, unidentified transition points, and incorrect terminology existed. Weaknesses in system engineering, procedure quality and procedure adherence resulted in poor prioritization of work, reduced component reliability, and delays in identifying and resolving equipment problems. Continuing management oversight and emphasis is needed to ensure long-term resolution of technical issues.

Plant Support

- Uncontrolled Contaminated Material: In February, the licensee identified over 100 contaminated tools and materials outside of the radiologically posted areas. The licensee performed a site radiological sweep to address this instance of poor contamination control, and is still developing comprehensive corrective action.
- The procedure upgrade program has been completed and improvement has been noted; however, procedures are still considered to be of average quality.
- Pipes from several systems in Unit 1 have been replaced, significantly reducing the radiation levels during outages. Additionally, decreases in the requirements for ISI and weld overlay activities have contributed to the reduction in radiation exposure.

**PLANT X**

**Assessment/analysis:** Although radiation protection improvements were made in the rate of personnel contaminations, the amount of contaminated areas, and daily non-outage dose, the lack of control over contaminated tools and materials that occurred in February demonstrated the extent of procedural adherence problems and lack of effective corrective actions. Continued emphasis on improving performance in radiation practices is needed. In emergency preparedness and security, a slight decline in performance was seen. In EP, effective corrective actions were implemented for the performance weaknesses that were identified in the 1993 exercise. In security, a performance decline was noted in package searches and some badge control problems.

## III. FUTURE ACTIVITIES

- The staff is currently evaluating the license's proposed Technical Specification Upgrade Program. The first amendment was issued in 1994 and nine more packages are in concurrence. Both the licensee and staff have added resources to the project. The staff review is scheduled to be completed by the end of 1997.

- The Unit 1 refueling outage is scheduled for October 1995. During the last Unit 2 refueling outage, UT inspection revealed the potential of a 135° circumferential non-through-wall crack at the 2B steam generator girth weld. The NRC evaluated the cracking conditions and issued an SER allowing continued operation for 24 months. Plant X is planning to examine and repair the girth welds in all eight SGs during the next round of refueling outages. Regional office nondestructive examination (NDE) inspections are planned, as noted below.

- The following inspections are planned over the next 6 months:

Regional Initiative:

| Procedure | Title |
|---|---|
| 42700 | Plant Procedures |
| 57080 | NDE - Ultrasonic Examination |
| 57090 | NDE - Radiographic Examination |
| 61706 | Core Thermal Power Evaluation |
| 61708 | Isothermal and Moderator Temperature Coefficient Determinations |
| 62705 | Electrical Maintenance: Observation of Work Activities, and Review of Quality Records |
| 83526 | Control of Radioactive Materials and Contamination, Surveys, and Monitoring |

Safety Inspection:

| | |
|---|---|
| 2515/123 | Implement Revised 10 CFR Part 20 |

Others:

Initial Operator Examination
INPO Plant Evaluation

# DATA SUMMARY INPUT

## I. PRA

SPSB

Include the following in the PRA write-up and limit the total discussion to 2 pages.

PRA Insights

Briefly discuss insights from the plant-specific or generic PRA.

PRA Profile

Briefly discuss the PRA profile, including dominant sequences from the PRA.

Core Damage Precursor Events

Briefly describe any ASP events from the last 2 years of published data, including the conditional core damage frequency. For example:

July 7, 1991: During a monthly surveillance test, one of the two primary containment isolation valves, in each of the two independent RHR sub-systems, was found to be inoperable. It was found that the loop "B" inboard isolation gate valve stem threads in the motor operator were worn and broken, causing the valve to lock in a partially open position. The stem of the "A" outboard isolation globe valve was severed. With these valves degraded, it was determined that neither RHR train was able to perform its Low Pressure Coolant Injection (LPCI) safety function and only the "A" train was able to perform the residual heat removal function. -- CCDP: 2E-5

## II. ENFORCEMENT HISTORY

OE

Briefly discuss all escalated enforcement actions that have occurred during the last two years. For example:

4/91 - CIVIL PENALTY: The action was based on a violation of primary containment requirements that lasted an entire operating cycle. A civil penalty was issued to emphasize the importance that the NRC places on ensuring that the containment will be capable of performing its safety function of preventing significant releases of radioactive material to the environment, the need to specify acceptance criteria for flange fit-up including bolt-tightening, and the need to perform appropriate post-maintenance testing. Escalation of the civil penalty was appropriate for duration. ($100,000)

10/92 - SEVERITY LEVEL III VIOLATION: The action was based on the licensee's failure to have the required number of low pressure coolant injection subsystems operable because of an improperly set torque switch on an MOV. The civil penalty was completely mitigated based on the licensee's identification of the problem, corrective actions, and good past performance in the area of MOV testing.

PLANT NAME
EVALUATION FACTORS FOR SENIOR MANAGERS' REVIEW OF LICENSEES
EXHIBITING SUPERIOR PERFORMANCE

| Evaluation Factors | Response | Comments |
|---|---|---|
| I. SALP Assessment | | |
| a. SALP 1 ratings in Operations, Maintenance, Engineering, Plant Support. | | |
| b. A commitment to achieve excellence and address identified weaknesses has been demonstrated. | | |
| II. Self-Assessment and Problem Resolution | | |
| a. Self-assessments of performance by quality assurance and safety oversight groups are timely and effective. | | |
| b. Safety issues are routinely identified to the appropriate management level and corrected in a timely manner. | | |
| c. Corrective actions are thorough and properly prioritized such that problems are effectively resolved. | | |
| III. Management Organization and Oversight | | |
| a. Corporate and plant management is fully committed to achieving superior overall safety performance. | | |

| Evaluation Factors | Response | Comments |
|---|---|---|
| b. Corporate and plant management effectively oversees plant activities and is actively involved in operating the plant and resolving problems. | | |
| c. Corporate and plant management provides strong direction and fosters a nuclear safety work ethic that is understood at all levels in the organization. | | |
| IV. Current Performance Level | | |
| a. Performance indicators reflect superior overall safety performance since the last SALP. | | |
| b. The NRC has not taken any escalated enforcement actions resulting in a civil penalty for events that occurred during the past year. | | |
| c. The NRC does not expect a civil penalty to result from the present consideration for escalated enforcement of any events that occurred during the past two years. | | |
| d. A superior level of safety performance has been maintained since the last SALP, as evidenced by a lack of significant operational problems and operator errors. | | |

| Evaluation Factors | Response | Comments |
|---|---|---|
| e. Significant problems with the quality of and adherence to procedures have not been identified. | | |
| V. Additional Considerations | | |
| a. A reduction of NRC inspection activity is not expected to contribute to complacency within plant management. | | |
| b. The NRC is not conducting any significant inspections or investigations of allegations that, if substantiated, might adversely reflect on overall plant performance. | | |
| c. The NRC does not expect plant performance to be adversely affected by anticipated changes to the rate-making basis for the licensee. | | |
| d. The NRC does not consider there to be any management issues, such as corporate support or recent personnel changes, that might adversely affect plant performance. | | |

# HOPE CREEK

## I. BASIS FOR CONCERN

Hope Creek's performance has been declining, primarily due to weaknesses in the control of plant activities as a result of inadequate communications and ineffective resolution of some plant problems. The latter is primarily attributed to weak root cause determinations. For example, in March 1995, poor control of testing and operation of the Decontamination Solution Evaporator (DSE) contributed to an unmonitored release through the South Plant Vent. This event also revealed operations and communications weaknesses. A July 1995 Partial Loss of Shutdown Cooling event confirmed communications and procedural adherence weaknesses. Enforcement action has been taken for the Partial Loss of Shutdown Cooling event. Senior plant management also failed to recognize the significance of this event.

The Vice President-Nuclear Operations has recently concluded that although the plant material condition is generally in better condition than at Salem, Hope Creek has similar performance problems. These operational problems have led to the transfer of the Operations Manager and other management and organizational changes. The unit entered a November 1995 refueling with an open-ended work plan. Recent NRC inspections were conducted to ensure adequate operations oversight during the shutdown, and to ensure that shutdown risk from emergent work was adequately controlled and managed.

## II. PERFORMANCE SUMMARY AND ASSESSMENT

<u>Operations</u>

- In March 1995, poor control of testing and operation of the Decontamination Solution Evaporator (DSE) contributed to an unmonitored release through the South Plant Vent. Ineffective internal and external communications compounded the event.
- A July 1995 Partial Loss of Shutdown Cooling Flow resulted from inadequate procedural guidance and poor operator training. Poor communications between onshift operators and their supervisors and the failure to follow operating procedures were additional root causes. The failure of senior plant management to properly assess the significance of this event resulted in a delay in initiating a comprehensive event evaluation and also contributed to the failure to properly inform the NRC.
- Operators responded well to two events that occurred in May 1995:
  - The first event involved a loss of a 500 KV transmission line that caused a spurious runback of the "B" reactor recirculation pump.
  - The second event involved an unexpected feed water flow transient which resulted in a reactor vessel level decrease to 32 inches. Especially noteworthy was the coordination among operations,

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system engineering, and maintenance personnel during the second event.

- Operator knowledge, understanding, and implementation of Technical Specifications (TS) was weak:

  - For example, the Reactor Core Isolation Cooling system automatically tripped, and a senior shift supervisor was inappropriately waiting for further engineering input prior to making his operability determination.

  - In another case, a degraded condition was identified in the High Pressure Coolant Injection system (minimum flow check valve weld crack) in which the initial operability call was made based on engineering documentation that did not fully consider either TS requirements or Generic Letter 91-18 guidance.

- Operators committed a number of personnel errors, including mispositioning components which resulted in degraded performance of systems important to safety (e.g. residual heat removal, service water); failure to adequately control activities specifically prohibited by technical specifications (e.g. polar crane operation); and, failure to control safety tagging that could have led to personnel safety concerns. Although these errors did not result in any immediate impact on plant or personnel safety, they demonstrated operator performance deficiencies.

- An NRC review of recent Licensee Event Reports (LERs) indicates an adverse trend in safety system failures. This trend highlighted weakness in the operation and maintenance of key components. This further supports the earlier NRC findings that the licensee has been ineffective at resolving plant problems; primarily due to weak root cause determinations.

- As a result of recent assessments in the operations department following the July 1995 Partial Loss of Shutdown Cooling Flow, the operations manager has been replaced, and three senior shift supervisors have been moved off-shift, with their crews being reconstituted. The licensee has also implemented a multi-faceted performance improvement plan designed to address the widespread programmatic performance weaknesses.

Assessment/analysis: Operations department performance during this period has been generally mixed, with some areas declining. Operators demonstrated good performance when conducting planned evolutions and responding to events. Conversely they demonstrated weak knowledge when implementing technical specifications. Management has continued to provide generally good support for day to day operations; however, their support was ineffective during a partial loss of shutdown cooling flow from the reactor vessel and during an unmonitored release. Internal communications, as well as those with the NRC, were noted as weak. The licensee has recently reorganized the operations department to improve its effectiveness.

Maintenance

- The licensee demonstrated outstanding actions following the identification of a safety related motor operated valve which had been significantly overgreased. These actions led to the timely and effective resolution of the potentially generic concern.

- Routine maintenance activities were generally effective at supporting reliable plant operation. For example, on-line maintenance for the "C" emergency diesel generator was especially well planned and executed. In another example, maintenance and technical department personnel secured an adjustment collar for an air-operated scram discharge volume drain valve to resolve a problem with valve closure.

- The inspectors reviewed the corrective maintenance backlog and concluded that it was adequately prioritized, scheduled and monitored. However, weaknesses were noted in management's ability to assess the overall safety impact of this backlog, adhere to established time standards for work completion, and understand how effectively backlogged work in individual priority levels was being worked off.

- Some recent decline in equipment performance has been noted in process and effluent radiation monitoring equipment, HVAC equipment, and the feedwater system. While not directly related to the primary safety systems, these performance problems have led to an increasing work load and frequent challenges to plant operators and maintenance and chemistry technicians.

- Surveillance activities were problematic. For example:

  - The licensee identified that a surveillance procedure for vital bus degraded voltage relays caused both available channels to be made inoperable.

  - The licensee determined that certain diesel generator surveillances did not completely verify vital bus load shedding circuitry. Specifically, overlap testing between diesel generator undervoltage (UV) relays and individual breaker trip circuits had never been adequately demonstrated since initial plant licensing.

  - An 18-month TS surveillance requirement to actuate a traversing in-core probe (TIP) explosive squib valve had not been completed since February 1991. This was attributed to inadequate surveillance scheduling.

  - There were three examples of missed or incomplete ECCS actuation instrumentation surveillances for the HPCI and ADS systems.

The licensee has implemented the Technical Specification Surveillance Improvement Program with the broad charter to correct longstanding problems with the surveillance program.

Assessment/analysis: Maintenance performed major activities well, including coordination with other departments. Planned process improvements have resulted in improved corrective action for maintenance problems. Additionally, a major overhaul of the "C" emergency diesel generator was effectively accomplished and management demonstrated good safety awareness while performing this on-line maintenance. Increased management attention is needed to address the growth of the corrective and preventive maintenance backlog. Non-safety related equipment degradation has led to more frequent operator and technician challenges. Some weaknesses have been noted with the surveillance program; examples include (1) missed surveillances due to errors in scheduling, (2) incomplete surveillances for a vital bus's load shed circuitry, and (3) missed or incomplete surveillances for HPCI & ADS ECCS actuation circuitry.

Engineering

- In May 1995, the NRC inspected Hope Creek's hardened wetwell vent to determine the licensee's compliance with commitments made in response to NRC Generic Letter (GL) 89-16, "Installation of a Hardened Wetwell Vent." This inspection concluded that the torus vent system was well designed, properly evaluated, and complied with GL 89-16 commitments. Operators were well trained on the modifications and emergency operating procedures were comprehensive.

- Conduct of system engineering support activities has been good. For example, all facets of the new fuel receipt process were conducted in a safe and effective manner. System engineering also coordinated well with other departments throughout the evolution.

- Root cause analysis and resolution of the March 1995 partial loss of offsite power and manual scram events were determined to be comprehensive and in keeping with new initiatives to improve the station's corrective action program. The licensee's root cause analyses, corrective action implementation, and development of long term preventive actions were excellent.

- In April, two Hiller-actuated flex wedge gate valves failed to open on demand. Because of the long standing nature of these problems, these and other related problems have been documented in numerous licensee and NRC inspection reports. Based on these continued failures, NRC inspectors determined that previously implemented corrective actions have been ineffective and that increased management attention is warranted.

- On July 7, operators shut the plant down from approximately 100% power in accordance with plant technical specifications due to a failed control room emergency filtration (CREF) system. The "A" CREF system was declared inoperable on June 30, and troubleshooting efforts were unsuccessful at identifying the cause of its periodic tripping. The licensee requested enforcement discretion on July 7 to continue troubleshooting efforts and process an emergency technical specification change to provide a longer AOT for this system. The NRC did not grant

enforcement discretion since the root cause of the problem was not yet identified.

Assessment/analysis: Corporate engineering support of plant activities was a continued strength. This included thorough and in-depth evaluations of abnormal conditions, and effective resolution of generic issues, such as the hardened wetwell vent modifications. Reactor engineering conducted new fuel receipt and inspection in a safe manner. In contrast, corrective actions were not always effective. Hiller actuated valves did not open on demand and they continue to be problematic. Management oversight was variable and, in one significant instance, weak, since the licensee requested enforcement discretion for a failed control room emergency filtration (CREF) system without defining the root cause for the failure. The request was denied and a forced shutdown was required to adequately troubleshoot and correct the problem.

Plant Support

- Plant support activities were observed to be very good. Notwithstanding the weak corrective actions for unresolved problems with the radiation monitoring system and radioactive waste handling, radiation protection and chemistry personnel performed appropriate samples and other tasks to ensure that releases were appropriately monitored and quantified. Improvements were seen in personnel exposure tracking activities.

- Radiation protection personnel continued efforts to decontaminate the grounds following the unmonitored radioactive release event of April 5, 1995, and these were judged to be timely, comprehensive and effective.

- The radioactive waste system functional review was found to be extremely thorough and provided excellent, critical assessment of radwaste system operations. Response to the unusual high volume of radwaste was considered well coordinated with proper management attention. Condensate chemistry problems due to a breakdown of demineralizer resin contributed to the large volume of radwaste and impacted plant startup activities.

- In May 1995, the Mobile Laboratory verified the licensee's capability for analyzing radioactive effluents, and concluded that, except for wet radiochemistry results, which will be reviewed in a subsequent inspection, Hope Creek had effective programs in place for measuring radioactivity in-process and effluent samples.

Assessment/analysis: Performance in the plant support areas continued to be excellent. The radiological controls program, including staffing, quality assurance oversight, and ALARA programs were considered licensee strengths. Plant support staff, with the strong support of engineering and the onsite safety review group, completed a detailed self assessment of the radwaste system. The radiological environmental monitoring program was effective. Radiological samples were accurate and correlated well with NRC results.

Continued outstanding housekeeping practices ensured that the radiologically controlled area remained free of debris.

The licensee's emergency planning program, including training, staffing and support of offsite responders, was considered a strength.

Good performance by security department personnel was evident in the conduct of their routine duties.

## III. FUTURE ACTIVITIES

- At the November 1, 1995, Hope Creek Plant Performance Review (PPR), recent licensee performance trends were reviewed in detail and the listed inspections were planned. Additional consideration was given to including Hope Creek with Salem Assessment Panel (SAP) activities.

Regional Initiative:

| Procedure | Title |
|---|---|
| 71715 | Sustained Control Room Observation |
| 60710 | Refueling (Shutdown) Activities |
| 71001 | Operator Requal Program Review |
| 83728 | ALARA |
| 73753 | ISI- Core Shroud Inspection |
| 62700 | Review of Maintenance Practices |
| 93806 | Readiness Assessment Team Inspection |
| 93808 | Integrated Performance Assessment Process (IPAP) |

DATA SUMMARY

I. PRA

A. PRA Insights

Hope Creek is a BWR 4 with Mark I containment design. Because BWRs are equipped with diverse and redundant systems to mitigate LOCAs, station blackout (SBO) sequences tend to be the dominant contributor to core damage frequency (CDF) in BWR PRAs. In the Hope Creek IPE, over 70% of the contribution to CDF stems from SBO sequences. Hope Creek has four EDGs (A, B, C, & D) at the site, two of which are necessary to safely shutdown the plant. However, not all combinations of two EDGs power the necessary equipment. Either A and C, or, B and D are necessary to power one division to shutdown the plant, making Hope Creek have the equivalent of two EDGs, any one of which is necessary to shut down the plant. When compared to GE BWR 3 and 4 plants, the relative contribution of SBO to the IPE CDF is high, though the absolute contribution is comparable to other PRAs.

The Hope Creek IPE identified one vulnerability from its front-end analysis: the inability to supply long-term cooling to switchgear room or Class 1E Panel room upon HVAC system failure. Prior to recovery, loss of HVAC contributed 3.29E-03/yr to the estimated CDF. To address this vulnerability, Hope Creek developed a risk-based recovery procedure that supplies alternate ventilation to prioritized rooms as determined from the IPE's room heat calculations. This procedure is credited in the IPE.

The IPE indicated that ATWS is a small CDF contributor due to the automatic operation of the SLC system.

The following equipment failures and operator actions were found important based on both risk increase and risk reduction importance measures: the EDG failures, HPCI/RCIC failures, SSW and SACS failures, and failure to provide alternate ventilation to the Class 1E Panel Room within 12 hours after a loss of HVAC.

B. PRA Profile

The licensee submitted their IPE on May 31, 1994. The Hope Creek IPE is based on Level 2 PRA. RES is currently reviewing the Hope Creek IPE; the IPEEE is due July 31, 1997. The IPE was performed by PSE&G with support from SAIC, NUS, Gabor, Kenton and Associates, ERIN, ABB Impell, and Reliability and Performance Associates. The IPE reported the CDF to be 4.58E-5/yr from internal events excluding internal flooding. The CDF from internal flooding was 5.5E-07/yr.

The contributions to CDF reported in the IPE by internal initiators are listed below:

| Initiator | CDF/yr | % Total CDF |
|---|---|---|
| Loss of offsite power | 3.39E-05 | 73.8 |
| Transients | 6.79E-06 | 14.8 |
| LOCAs | 3.07E-06 | 6.7 |
| ATWS | 7.45E-07 | 1.6 |
| Special Initiators | 1.42E-06 | 3.1 |

The Hope Creek IPE shows that five dominant sequences have a CDF greater than 1E-06 and they contribute approximately 84.2% of the total CDF. The most dominant sequence is a LOOP/SBO (TeEDG).

The back-end analysis provided the following key results:

- The frequency of high early and medium early release are 9.42E-06/yr (21% given core damage) and 6.14E-06/yr (14%).
- Late containment failure occurs in 17.7% of the sequences. The containment does not fail in approximately 20% of the cases, with venting from wetwell taking place in approximately half of these cases.
- The results were found to be very sensitive to AC power recovery assumptions and two uncertainties in ex-vessel phenomena: drywell shell melt-through and debris coolability.

C. Core Damage Precursor Events

On the basis of the precursors identified by ORNL for 1993 (NUREG/CR-4674, Vols. 19 and 20) and the preliminary precursors for 1994 and 1995, SPSB did not identify any precursor events for Hope Creek that have a conditional core damage probability (CCDP) of 1E-5 or greater.

Also, the staff identified the following "Significant Event" for the Performance Indicator Program:

> July 8, 1995: A shutdown cooling bypass event was initiated when the operating crew left the Reactor Recirculation Pump discharge valve in a partially open position to mitigate potential thermal binding. During the shutdown 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. Later, bypass flow increased to 4000 GPM when the valve was further opened in an attempt to re-close the valve. The valve was manually closed on July 9, terminating the event almost 19 hours after the beginning of the event. It was determined later that an operational condition change occurred from Cold Shutdown to Hot Shutdown, resulting in several violations of TS LCOs.

D. Hope Creek Recent Safety System Failures

Hope Creek has experienced a recent increase in the number of LERs reporting safety system problems. A review was performed to determine if the events could have a major impact on the assumptions made in the plant's IPE for safety system reliability. Most events were found to have minimal impact on the plants IPE for power operation. It should be noted that major increases in equipment downtime due to maintenance could have an impact if they continued to occur over a long time. A review of the events follows:

LERs impacting HPCI

LER 95-008 - On May 29, an on-the-spot change (OTSC) was written to prevent the affects of the barometric condenser vacuum tank pump on the HPCI jockey pump in-service test. The change isolated the condensate pump return from the HPCI pump suction. Subsequent review to incorporate the OTSC into the permanent procedure revealed the revision would render HPCI inoperable during the jockey pump test. The valve that was closed also isolated the HPCI turbine lube oil cooling water outlet. The root cause of this event was personnel error. The impact of this event on the plant's IPE would be minimal and would involve an increase in the out of service time for HPCI during the one test prior to discovery of the mistake.

LER 95-011 - On June 22, 1995, the HPCI system was declared inoperable. This determination came a day after discovery of a small leak due to a crack in a weld of the minimum flow check valve. The delay in recognizing that the pressure boundary was affected and the necessity to isolate the valve were due to the inappropriate engineering judgement that the valve was operable. The licensee has attributed the root cause of the weld failure to personnel error in that it is assumed that someone must have stepped on the line to cause the observed failure. The impact of this event on the plant's IPE is minimal, and would involve an increase in the out of service time for HPCI from the time of discover until the time of repair of the leak.

LER 95-014 - On July 3, an automatic ESF actuation occurred when the HPCI pump suction path swapped-over from the CST to the Suppression Chamber on high Torus level. The swap-over occurred following surveillance testing of the HPCI Minimum Flow Valve logic channel calibration and subsequent performance of routine Drywell Nitrogen make-up activities. Both activities slightly raised Torus water level. The licensee states the root cause was shift personnel did not fully understand the potential effects of the 'total loop' instrument accuracy when determining the consequences of performing activities that would either increase the mass of water within the Torus or effect the level transmitters readings (see LER 95-020). This event was self revealing and the lineup promptly corrected. The impact is minimal.

LER 95-020 - On September 8, an automatic ESF actuation occurred when the HPCI pump suction path swapped over from the CST to the Suppression Chamber on high Torus level. Prior to the event Torus water level and

temperature were higher than normal values due, in part, to in-leakage from 3 weeping SRVs. To lower water temperature, the RHR system was placed in the Torus Cooling mode. Eleven hours later, operators began routine Drywell Nitrogen make-up activities which was expected to raise Torus level slightly. One hour after nitrogen addition, a high Torus water level alarm was received while control room indicators were reading below the trip setpoint of 78.5 inches. The licensee attributes the apparent cause of the event to inadequate and ineffective corrective actions in preventing a recurring event (see LER 95-014). This event was self revealing and the lineup promptly corrected. The impact is minimal.

LER 95-021 - On September 20, the HPCI system was declared inoperable because the system oil reservoir sample indicated a moisture content of 0.23% which is above the specified limit of 0.20% established by GE. The apparent cause of this event was steam leakage through the turbine steam admission valve. The HPCI was determined to be functional, although it was technically inoperable. There would be an increase in equipment unavailability due to the maintenance time to change the oil. IPE impact is minimal.

LER during shutdown

LER 95-016 - (SIGNIFICANT EVENT) On July 8, a Shutdown Cooling Bypass Event occurred which rendered the shutdown cooling mode of the RHR inoperable. The event was initiated when operators positioned the Reactor Recirculation Pump discharge valve partially open 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. Weeks after the event, the licensee determined that the plant's operational condition had changed from Cold Shutdown to Hot Shutdown. The licensee identified the root causes as procedural non-compliance, lack of questioning attitude, not believing indications, and lack of follow-up regarding verification and validation of plant indications. This event, although significant to the plant, does not impact the IPE for power operations.

LERs due to inadequate testing

LER 95-009 - On June 13, in preparation for refuel outage 6, outage planners reviewed the previous work order history for the TIP system explosive squib valve firing and replacement. The records indicated that the last squib surveillances were performed during refuel outage 3 in 1991. TS 4.6.3.5.b requires firing of one squib every 18 months. In addition, all five squib valve explosive charges were found to be past their manufacturers expiration date. The licensee identifies the root cause as personnel error. Equipment unavailability increases due to maintenance for replacement. This event had minimal impact on the IPE.

LER 95-013 - During June, the licensee determined that piping for the Turbine First Stage Pressure Inputs to the RPS instruments and Low Vacuum Input to the Nuclear Steam Supply Shutoff System were not

visually inspected during pressure tests in the second inspection period of the 10 year In-service Inspection interval. This surveillance is considered to have been overdue since April 27, 1995. The root cause was identified as personnel error. This event had no impact on the IPE.

LER 95-017 - On July 13, in response to the discovery of a drawing discrepancy, a licensee review revealed surveillance testing of the vital bus load shedding circuits was incomplete. As a result, the required TS surveillance testing was considered to have been missed and all four diesel generators were declared inoperable. The licensee attributed the cause to surveillance procedures that did not provide sufficient overlap. This event had no impact on the IPE.

LER 95-018 - On July 20, the licensee discovered that a single channel calibration of the ADS actuation instrumentation performed on June 28, 1995, had been improperly credited as a functional test involving three channels. A licensee review of test procedures and work histories, dating back to 1991, revealed that on three different occasions, multi-channel testing of ECCS instrumentation was improperly credited based on a single channel calibration. The licensee attributes the missed surveillances to functional test procedures that improperly allowed crediting of multi-channel test based on performance of a single channel calibration. This event had no impact on the IPE.

Other equipment operability LERs

LER 95-015 - On June 30, the licensee entered a 7 day LCO due to an inoperable 'A' Control Room Ventilation train following a trip. On July 7, 1995 the LCO expired and the licensee reduced power in accordance with the TS action statement. Shutdown was completed on July 8, 1995. The licensee determined the root cause to be a momentary power interruption to the control circuit due to a problem with a freon temperature switch and lengthy cable runs which resulted in large voltage drops sufficient to prevent the control circuit from recovering. The risk would involve the transient risks associated with shutdown, which are not modeled in the IPE for power operations.

LER 95-019 - On July 31, the RCIC Jockey pump was declared inoperable during the performance of the quarterly In-Service Test (IST). The operability declaration was due to pump cavitation from insufficient NPSH when the pump suction was aligned to the torus. The previous suction path from the CST made it difficult to monitor equipment performance in accordance with ASME codes since variations in CST level between periodic ISTs made it difficult to obtain reliable pump performance data for trending purposes. Therefore, the torus suction path was added to the RCIC IST procedure to minimize the effects of level variations between tests. The apparent cause of this event was poor initial design when sizing torus suction piping to the jockey pump. This event has minimal significance.

II. ENFORCEMENT HISTORY

6/95 - SEVERITY LEVEL IV VIOLATION: The action was based on an Office of Investigations (OI) investigation initiated in March of 1994 resulting in an OI conclusion that a licensee senior reactor operator (SRO) deliberately falsified his 1985 application for an SRO license. The falsification consisted of the individual reporting on his license application that he had received a B.S. Degree in Mechanical Engineering when, in fact, he did not possess a degree. As a result of the individual's wrongdoing, the licensee was in violation of the technical specifications. The licensee terminated the SRO after determining that he had falsified the application and the NRC cited a Severity Level III violation against the SRO's Part 50 license.

7/95 - SEVERITY LEVEL III PROBLEM: The action was based on the results of a special inspection to review the details of an unplanned release of radioactive materials from the station on April 5, 1995. The inspection determined that the release occurred because of an inadequate design of the liquid radwaste system and an inadequate radiation monitoring system. The action consisted of four violations that were grouped into a single Severity Level III problem. Because the facility had not been the subject of escalated enforcement action within the last 2 years, the factor of identification was not considered. Based on the licensee's prompt and comprehensive corrective action, credit was warranted and the action was issued without proposing a civil penalty.

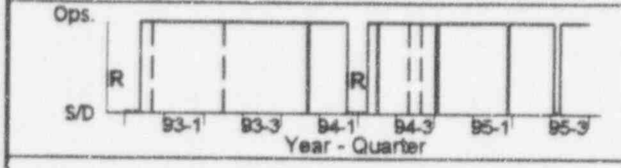
9/95 - SEVERITY LEVEL IV VIOLATIONS: The action was based on an OI investigation completed on October 11, 1994, resulting in an OI conclusion that an SRO failed to disclose a known violation of the technical specifications by failing to prepare a corrective action document after determining that there was not an SRO in the control room for approximately 3 minutes while the reactor was operational on June 3, 1992. Violations were cited for the technical specification violation and for not making the required report to the NRC pursuant to 10 CFR 50.73. The licensee removed the SRO from duties associated with plant operations and the NRC cited a Severity Level III violation against the SRO's Part 50 license.

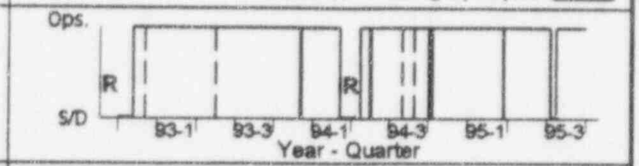
12/95 - CIVIL PENALTY: The action was based on the licensee's failure to follow procedures during a shutdown period that resulted in shutdown cooling flow being bypassed around the reactor vessel and caused the reactor to inadvertently change modes on July 8, 1995. A predecisional enforcement conference on this matter was held on November 6, 1995. Based on the fact that the licensee had had escalated action within the past 2 years, the agency evaluated whether credit was warranted for actions related to identification and corrective action. In both cases, credit was not warranted, therefore, a civil penalty equal to twice the base civil penalty was proposed. ($100,000)

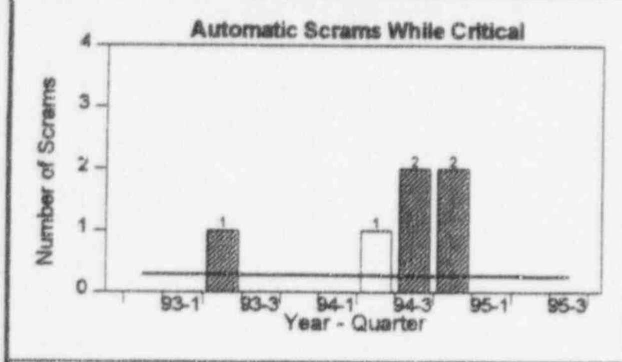
# HOPE CREEK

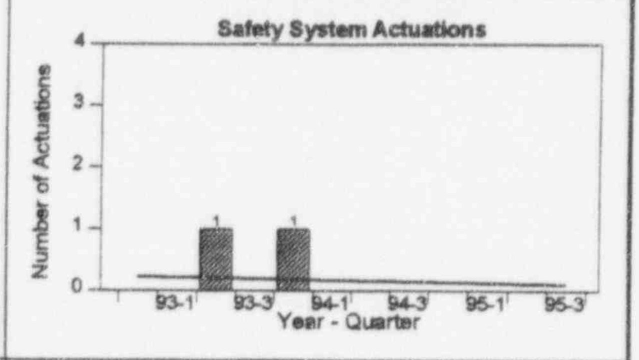
92-4 to 95-3 Quarterly Data

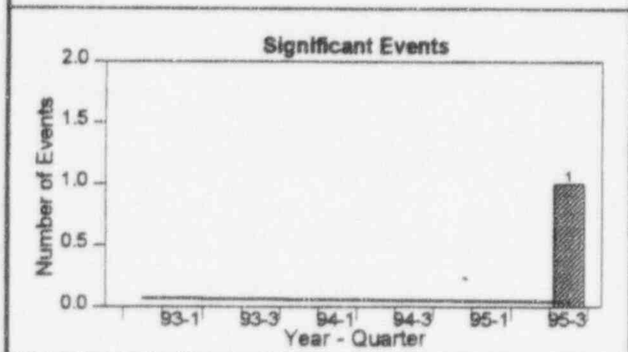
Legend:
Shutdown < approx. 72 hrs ¦ StartUp
Refueling R Operation
Industry Avg. Trend ——— ShutDown
Not Shown Using Op. Cycle

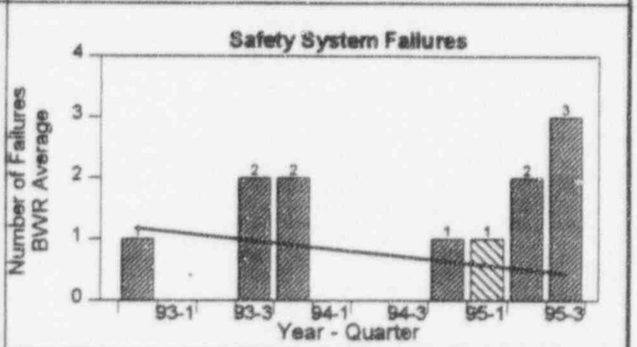


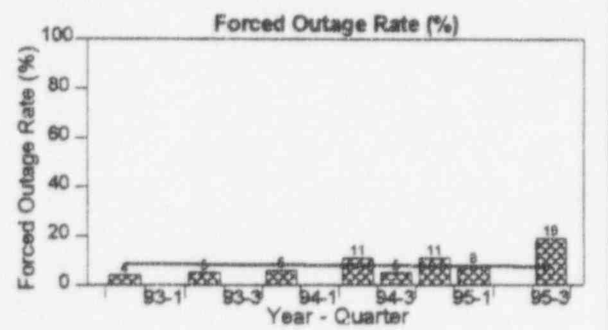



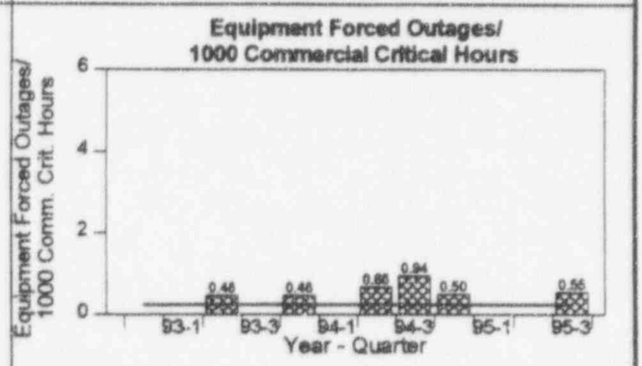



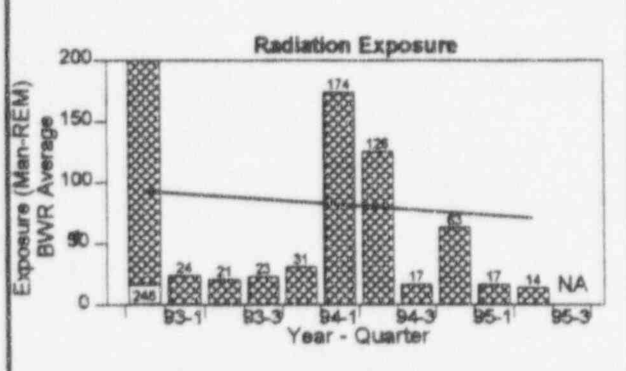



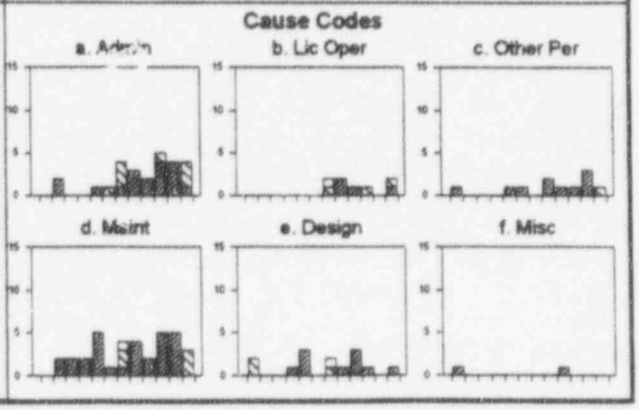



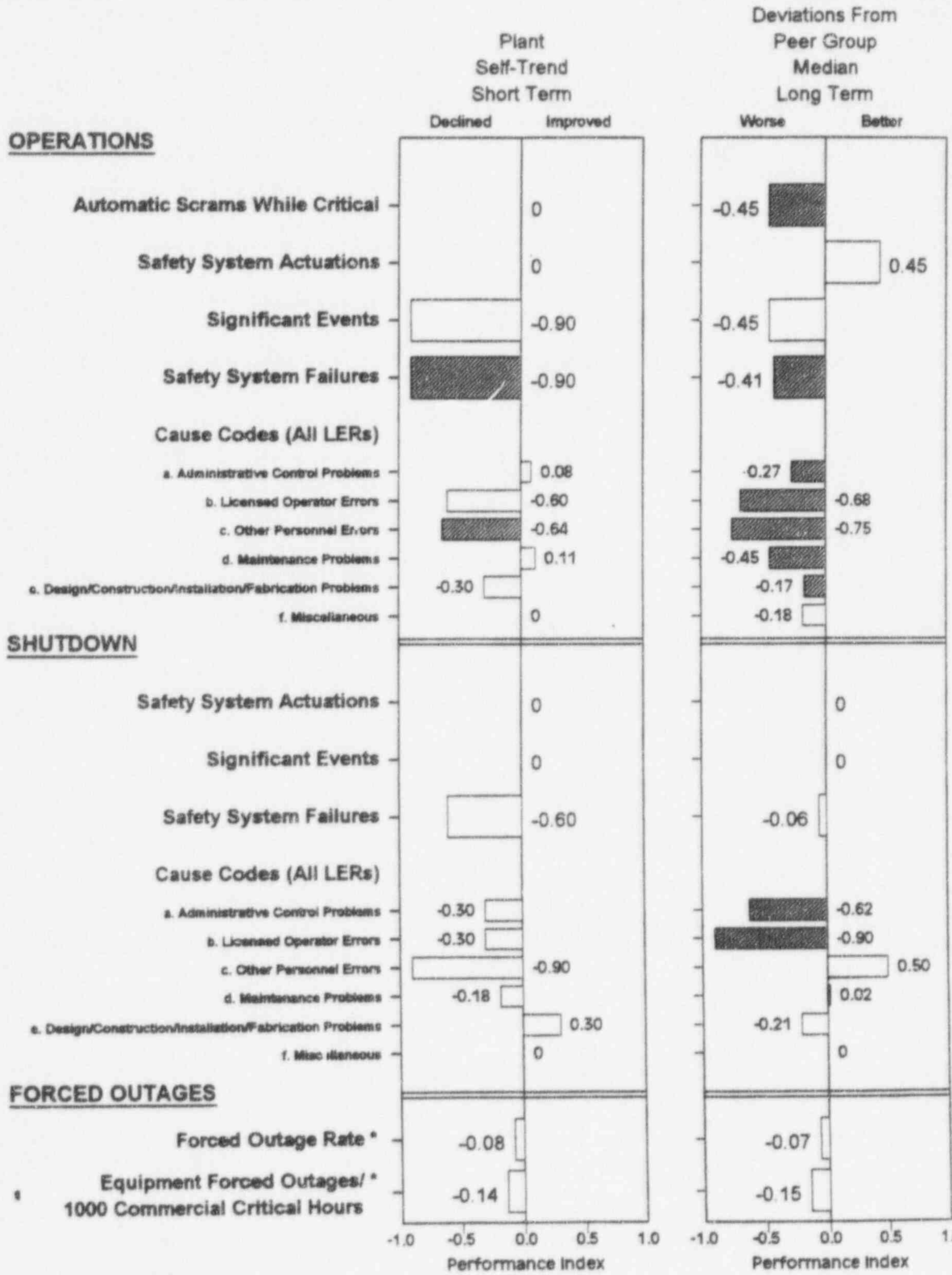



# HOPE CREEK

Peer Group: General Electric Post-TMI

92-4 to 95-3 Trends and Deviations

Legend: Statistical Significance — High, Medium, Low




* Not Calculated for Operational Cycle

## UPDATE FOR 1/96 SMM
## HOPE CREEK

### CURRENT PLANT STATUS:

* Core shuffle/reload almost complete. Outage restart now at 2/24, delayed primarily from EDG breaker problems.

### RECENT ACTIVITIES/PROBLEMS:

* Excessive RHR pipe deflection found on common suction to decay heat removal. Operators failed to recognize and enter LCO in timely manner. DHR considered inop pending engineering eval. DRS review ongoing, could be escalated enforcement.

* PSE&G discovered safety auxiliary cooling system (intermediate, closed loop system providing cooling to majority of safety loads) was and had routinely operated at temp below 65 degrees minimum specified in FSAR. Engineering evaluated lower temps as acceptable. DRS review agreed, inspection ongoing.

* Operators were continuously challenged by equipment issues just prior to entering refuel outage, including UE for failing to maintain adequate d/p between drywell and torus.

* NRC challenged PSE&G over acceptability of their planned full core offload. PSE&G ultimately decided 1/3 core fuel shuffle was safer. Unresolved item to review past performance.

* Several other performance issues have surfaced during the outage; 1) they discovered that they had never adequately tested the vital bus undervoltage relays, subsequently tested satisfactorily, 2) a fuel bundle was found misoriented from the last refueling, 3) abnormal rad conditions from incorrect bending of LPRM & 4) EDG auto-start when the bus was removed from service without first defeating the EDG auto-start.

* PPR letter sent 12/1/95 expressing concern over performance decline, particularly in operations. Management meeting conducted on 11/6 to discuss outage plans. Ltr sent on 12/15/95 asking for response articulating basis for adequate corrective action this outage to support restart (copies of letters attached).

* Operations manager transferred to training. Interim OPS manager on loan from Peach Bottom while PSE&G searches outside for replacement.

* Hope Creek response to IEB 95-02 an outlier; they do not intend to conduct special pump run for strainer clogging. NRC review ongoing.

* $100,000 CP issued 12/12/95 on bypass of shutdown cooling.

B/11

UPCOMING ACTIVITIES:

* Management mtg scheduled for 1/18/96 for PSE&G to discuss their letter (due 1/10/96) on the adequacy of corrective actions to support restart.

* RATI scheduled for 2 weeks, starting 2/5/96.