May 26, 1989

MEMORANDUM FOR: Paul Swetland, Chief
Projects Section 2B
Division of Reactor Projects
Region I

FROM: Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

SUBJECT: SALP INPUT FOR HOPE CREEK, SAFETY ASSESSMENT/QUALITY VERIFICATION

Enclosed is the SALP input for Hope Creek in the functional area of Safety Assessment/Quality Verification. Input was received from the NRR technical staff, and the senior resident at Hope Creek. For additional information, contact Clyde Shiraki at 492-1445.

/s/
Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosure:
SALP Input

cc w/enclosure:
E. Wenzinger, RI
G. Meyer, SRI, Hope Creek

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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ENCLOSURE

HOPE CREEK GENERATING STATION SALP INPUT

SAFETY ASSESSMENT/QUALITY VERIFICATION

ANALYSIS:

This new functional area combines the previous functional areas of Licensing Activities and Assurance of Quality and assesses the effectiveness of the licensee's programs in assuring the safety and quality of plant operations and activities.

The previous SALP Report rated the Licensing Activities functional area as Category 2 and noted the inconsistent quality of licensing submittals regarding technical content and timeliness. The Assurance of Quality functional area was rated as a Category 2 with an improving trend. The report noted that PSE&G had established the programs, procedures, and working environment to promote high quality, and encouraged continued management attention to weak areas such as the engineering department.

Since it is a relatively new facility, HCGS does not have the same volume of licensing actions as most facilities. During the assessment period, sixteen actions (amendments, relief requests, exemptions, etc) were processed. The quality of the technical evaluations was generally good, indicating that the licensee has a general understanding of the technical issues, is aware of and participates in industry groups, and uses acceptable approaches to problem solutions. The licensee's responses to requests for additional information or necessary corrections were usually prompt and well handled. The one exception dealt with a license change request to the Filtration, Recirculation, and Ventilation System. The licensee made a commitment in August 1988 to send additional information, and it was April 1989 before it was received. There was one instance of supplemental information being requested due to an incomplete license change request dealing with an amendment to the Technical Specification surveillance test intervals and allowable outage times for the reactor protection system. The supplemental information was submitted promptly and correctly.

The licensee's response to regulatory initiatives (i.e. Generic Letters and Bulletins) has been timely and complete. Frequent communications indicate that they commence work on their responses sufficiently in advance that they are able to meet commitment dates without requesting extensions.

PSE&G went beyond technical specification requirements to ensure proper system operation; for example, all fourteen safety relief valves (SRVs) were lift tested at power following replacement, not just the required five SRVs, and the acceptance criteria for High Pressure Coolant Injection (HPCI) System response time testing were reduced for low pressure conditions. After an acceptable HPCI overspeed test, the test was repeated to confirm acceptability. When a test engineer raised concerns regarding the orientation of isolation valves in primary containment ventilation lines, the concern was expeditiously raised to the plant management level and corrective actions were initiated.

PSE&G's adherence to the concept of personal accountability is most noticeable when observing the Senior Nuclear Shift Supervisors (the "Seniors"), the licensed operators held accountable for plant operations on each operating shift. The Seniors ensure that they concur with decisions, such as technical specification interpretations, the acceptability of equipment being returned to service, and courses of action. Each morning, the department managers attend a meeting run by the Senior to discuss plant status and plans, which reinforces the Senior's responsibility and provides the opportunity for him to have department managers address his concerns. The meeting provides ready accessibility from the operating crews to upper and middle level management, as well as being a vehicle that quickly involves engineering talent in operational problems.

During the evaluation of the feedwater flow measurement errors that resulted in the facility being operated above its maximum power level, the engineering staff displayed a willingness and ability to analyze data and events independent of the vendor representatives. In this instance, an engineer did not accept General Electric (GE) Company assurances that their (GE's) calculations were correct. GE subsequently acknowledged they had made an error.

Problem identification occurred both from within and from outside each organizational element. Incident Reports continued to be used to identify and resolve plant problems and off-normal events and for tracking corrective actions to completion. Hope Creek had 170 Incident Reports in 1988, 36 of which were reportable to the NRC. PSE&G has also initiated the Human Performance Evaluation System (HPES), a detailed analysis method for determining root causes in incidents involving personnel errors. This analysis technique has the potential for providing a thorough, innovative analysis of personnel errors. The licensee should consider applications of this technique to the personnel errors that occurred during this assessment period such as those briefly described below.

> The facility violated Technical Specifications by operating at nominally 101.2%, worst case 102.2%. Root cause was personnel error in that the calibration span values for feedwater flow transmitters were incorrectly established using calculations that were not compensated for high line pressure compression. The omission by PSE&G of this correction factor was not in accordance with NSSS vendor specifications. (LER 88-24)
>
> An isolation of the High Pressure Coolant Injection (HPCI) System occurred during performance of a steam leak detection system surveillance procedure. The I&C technician who performed the surveillance failed to place a bypass switch in the BYPASS position per procedure and when terminal leads were lifted during the course of the surveillance, an isolation of the HPCI primary containment outboard steam supply valve occurred. Root cause was personnel error in not following an approved procedure. (LER 88-33)

The control room differential pressure was less than technical specification required values. An engineering review determined that Control Room Emergency Filtration (CREF) system operability had not been demonstrated. Two root causes were identified: 1) an inadequate surveillance procedure, and 2) inadequate interface testing following an HVAC design change in the area adjacent to the control room. (LER 88-25)

A Nuclear Steam Supply System Shutoff Channel "D" isolation occurred when a fuse was blown on a portion of the Channel "D" isolation logic. The fuse blew during performance of a Maintenance Department I&C surveillance procedure when a meter lead was inadvertently dislodged from test equipment and came in contact with a ground bus inside the Division 1 Reactor Protection System Logic Cabinet. The isolation caused the Reactor Water Cleanup (RWCU) System outboard suction isolation valve to auto close, the "A" and "B" RWCU pumps to trip, and isolated Main Steam Line drain valves and a Reactor Recirculation sample valve. Root Cause was the lack of accessibility to testing points inside the subject cabinet, which directly led to the meter lead becoming dislodged. A contributing factor might have been poor work practice or skill. (LER 88-35)

During the performance of an I&C surveillance test procedure, RHR shutdown cooling was isolated because the procedure did not call for lifting a lead to prevent a valve from closing. (LER 89-04)

Primary Containment Isolation Valves were declared inoperable due to a missed surveillance test that resulted from a personnel error. (LER 88-02)

A missed surveillance test of the refueling floor exhaust process radiation monitor Channel "B" caused by a personnel error resulted in a technical specification violation. (LER 88-04)

A Design Change Package (DCP) inadequacy resulted in the inputs to the primary containment isolation system being inoperable. (LER 88-05)

An isolation of the reactor water clean up system (ESF actuation) resulted from misuse of test equipment, which caused a blown fuse and ESF actuation. (LER 88-18)

A power reduction and ESF actuation (RWCU isolation) were caused by loose terminations on a cabinet internal power supply. (LER 88-34)

An inadequate Design Change Package (DCP) caused an oscillation in drywell average air temperature measurements. The DCP was inadequately reviewed to determine the impact of its implementation upon the drywell average air temperature measurement. Abnormally oscillating drywell average air temperatures existed for over a month without being detected by operations or engineering personnel, although this parameter is recorded daily to ensure compliance with Technical Specifications. (Inspection Report 88-24)

An uncontrolled electrical jumper was installed in the control circuit of the drywell equipment drain pumps and remained undetected for ten months. (Inspection Report 89-02)

These items were variously caused by technician error, inadequate procedure review, poor work practices, or a loss of control of equipment. They further affect the areas of post maintenance testing, workmanship, and management oversight. In some way, they deal primarily with the I & C area. Since the responsibility of QA is to work toward quality operation of the facility, and these items are clearly nonsupportive of this goal, QA should be heavily involved trying to determine the causes and recommending solutions. If QA is already involved, their participation is evidently ineffectual.

The Quality Assurance Department, the Onsite Safety Review Group, and the Offsite Safety Review Group are responsible for providing effective, independent review of plant activities. The station quality assurance (QA) organization should be providing day-to-day review in the quality control and in-process review areas and should be integrated into the station's resolution of problems. In light of the problems experienced in the I & C area, these groups need to reassess their level of involvement and determine if there is more they can do to be of assistance.

The Station Operations Review Committee (SORC) was composed of department managers and provided consistent, effective review of significant plant issues, including design changes, post-trip reviews, reportable events, and station-wide procedures. During the optical isolator failure, the SORC met during the night to review the course of action before its implementation, a good indication of the SORC's role.

Three areas of the QA program were assessed during this assessment period, procurement, receipt inspection, and audits. The three areas continue to be effectively documented and administratively controlled and implemented by trained and qualified personnel.

Station QA involvement in ISI and startup testing was apparent. In the ISI area QA performed surveillance of in-progress ISI contractor activities, in-house reviews of contractor ISI procedures and audits at the contractor facilities. QA performed many surveillance activities during the post refueling startup testing program. However, QA has not devoted the same level of attention to IST as evident by the NRC review and supported by the NRC identified weaknesses noted in the Maintenance/Surveillance Area. In addition, a missed surveillance test and an inoperable valve resulted in a technical specification violation. (LER 88-02)

In summary, the performance of the various quality assurance groups has been inconsistent. Their involvement in solving the problems of personnel errors, inadequate procedure review, and missed surveillance tests is either nonexistent or ineffectual. High level management attention is necessary to bring the quality assurance groups together to clearly define or reemphasize the responsibilities of each group. Commitments should be made, and responsible managers held accountable for the results.

Although the licensee's licensing action submittals were generally of good quality, in the functional area of Safety Assessment/Quality Verification, the lack of effective quality verification and its corresponding effects on plant safety resulted in a downward trend in this area. An improvement was indicated over the last few months of the assessment period but close attention is required to determine if this is a coincidental or significant change.