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

MAR 2 3 1988

MEMORANDUM FOR: Sharon R. Connelly, Director
Office of Inspector and Auditor

FROM: Victor Stello, Jr.
Executive Director for Operations

SUBJECT: REFERRAL OF TECHNICAL ISSUES PROVIDED TO OIA DURING THE HOPE CREEK/SHOREHAM INVESTIGATION

By your memorandum dated January 29, 1988, you referred to me six technical issues at the Hope Creek and Shoreham nuclear stations that had been provided to the Office of Inspector and Auditor (OIA) during an ongoing investigation. You indicated that a alleged to OIA that he had previously reported these problems to Region I personnel and believed that the problems had not been appropriately resolved. You suggested that while OIA is reviewing the staff's handling of these issues, I might want the staff to review apparent technical disagreements between the alleger and Region I.

The Office of Nuclear Reactor Regulation (NRR) has reviewed the six technical issues attached to your memorandum, as well as the enclosures to that attachment, which include a transcript of the interview with the alleger dated April 16, 1987.

The conclusions of that review are documented in the enclosed technical evaluation.

Victor Stello, Jr.
Executive Director
for Operations

Enclosure:
As stated

cc: See attached list

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ENCLOSURE

NRR TECHNICAL EVALUATION OF OIA-REFERRED TECHNICAL ISSUES FROM THE HOPE CREEK/SHOREHAM INVESTIGATION

1. The first technical issue is an allegation that a big metal hanger was installed within a foot of a 480-volt motor control center (MCC) at Hope Creek in violation of the National Electric Code (NEC). This type of clearance problem was subsequently found to exist on a total of four MCCs at Hope Creek. The Hope Creek SAFETEAM (the licensee's organization responsible for reviewing employee safety concerns) had concluded that while the supports installed in front of three of the MCCs are not a safety hazard they are an inconvenience for those performing maintenance work on these MCCs. As a result, the licensee issued a Design Change Package to increase the distances between the front of the MCCs and the support members. The SAFETEAM concluded that the supports for the fourth MCC were acceptable as is because they are installed above the MCC in an area where there are no internal adjustable or renewable parts, and there is adequate space for the performance of routine maintenance. Although this problem has been resolved, the alleger remains concerned because (a) the SAFETEAM denied that the original configuration was a safety violation and (b) the licensee's electrical inspector did not recognize these installations as a violation of the NEC. The alleger also had additional non-technical concerns that are not addressed here.

   Requirements for clearances that allow work space in front of electric equipment are contained in the NEC, in National Fire Protection Association (NFPA) Standard 70E ("Electrical Safety Requirements for Employee Workplaces"), and in the Institute of Electrical and Electronics Engineers (IEEE) Standard 141 ("Recommended Practice for Electric Power Distribution for Industrial Plants"). In addition, the Occupational Safety and Health Administration (OSHA) requires such clearances in front of MCCs. However, none of these standards or codes is normally used by the NRR staff in its reviews of plant electrical systems. The codes, standards, and criteria used by the NRR staff

in reviews are given in NUREG-0800 ("Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants"). In the Final Safety Analysis Reports (FSARs) submitted as a part of their application for an operating license, utilities usually reference additional codes or standards in their Final Safety Analysis Report (FSAR) relating to specific design criteria. These then also become part of the basis for the staff's acceptance of the plant design.

The staff reviewed the electrical sections of the Hope Creek FSAR to determine if any of the codes or standards relative to working clearances in front of MCCs were referenced. The staff found that the NEC was referenced only for cable sizing, raceway fill, and electrical equipment and system grounding criteria. The fact that there was no reference for the working clearances is not unusual because such clearances are primarily a personnel safety concern rather than a reactor safety concern, which is the subject of the FSAR and the staff's licensing review.

The statement that this is a personnel safety concern is supported by the fact that the requirement for working clearances in front of electric equipment was excerpted from the NEC by NFPA Standard 70E. The foreword to NFPA 70E states that the standard was prepared to assist OSHA in preparing electrical safety standards that would serve OSHA's needs which relate to personnel safety. Given that no direct relationship to reactor safety has been identified (such as inadequate clearances for seismic purposes or electrical separation purposes), the staff is satisfied the issue has been acceptably resolved.

2. The second issue involves an allegation that the licensee (Public Service Electric & Gas Company, PSE&G) had repeated problems with the loss of offsite power (LOP) tests at Hope Creek. The alleger stated that the first test had 24 significant events and a repeat test 8 days later had 17 significant events. The alleger would "like to have the NRC comment on whether

or not this failure level is typical, since Hope Creek has been described by the NRC, who had lauded its Quality Control Program, as a model in the industry." On the basis of the alleger's comments on pages 32 and 33 of the transcript provided with the OIA memorandum, it appears that the alleger is concerned that the quality assurance (QA) personnel at Hope Creek were not auditing the work in the field and this is the cause of the LOP test problems observed.

The alleger is correct in stating that 24 test observations were made in the first LOP test and 17 in the second. The number of test observations made during the LOP tests at Hope Creek is not typical. After the second test, which was conducted on September 19, 1986, Region I formed an Augmented Inspection Team (AIT) to investigate and evaluate the test observations. The Region sent a Confirmatory Action letter to the licensee on September 24, 1986 that required, among other things, (a) that any additional LOP integrated testing be deferred until the AIT team leader determined that such testing could continue and (b) that, prior to restart, the licensee provide the Regional Administrator a written report that includes the licensee's analysis of the LOP testing conducted on September 11 and 19, 1986.

The AIT, which was composed of representatives from Region I and the Office of Nuclear Reactor Regulation and the Office of Inspection and Enforcement, examined in detail all 41 LOP test observations. The AIT determined that the results of LOP tests indicated certain weaknesses in the design, construction, and testing programs for Hope Creek. With the exception of the problems found related to the Bailey solid state logic module system, the team found that the weaknesses were minor in nature. For the Bailey system, however, the team identified concerns with the adequacy of bench and surveillance testing and with the significant failure rate, which was higher than expected.

It does not appear that lack of field inspections or audits was a major contributor to the LOP test observations made at Hope Creek. QA appears to be a factor in less than half of the test observations made during the first two LOP tests performed, and where QA was involved it was mostly related to design control shortcomings and the difficulty of performing integrated preoperational tests. Examples of test observations made where QA was not a factor include those related to system responses that were in accordance with system design, where inadequate operating procedures were involved, and other miscellaneous categories such as operator error, training, and security. Examples of test observations made where QA was or may have been a factor include those that were the result of system designs that did not correctly implement FSAR commitments or regulatory requirements (design control) and observations made that were the result of system interactions that might have been identified if a more fully integrated preoperational test had been performed. Specifically with regard to the Bailey logic modules, there were failures that might have been found if a more thorough bench test capability existed and there had been a more thorough system for checking the field-programmable logic arrays and staple jumper placements.

The licensee and the AIT have addressed each of the 41 test observations made during the LOP tests conducted on September 11 and 19, 1986 at Hope Creek. A review of the AIT inspection report gives no indication that lack of field inspections or audits was a major contributor to the problems observed.

3. The third issue involves an allegation that "radiated water" could leak into the Long Island Sound at Shoreham. The alleger stated in part that: "And, I said, they are talking about the condenser pipe and something that -- radiated water leaking out, somebody had made the allegation that radiated water would get out into the sound. And, the NRC's re-

sponse was, no, water would not get into the sound, it was a vacuum area, and that the only thing that would happen would be sound water would come in."

The precise location of the alleged leakage was not specified. Therefore, because the only clue to the alleged leakage location is the statement "condenser pipe," the staff assumed that the alleger was referring to the main condenser, in which steam from the turbine exhaust is condensed by cooling water supplied from the outside of the plant. The main condenser maintains negative pressure during operation by steam condensation and steam jet air ejectors, which take suction on the main condenser so that a vacuum of approximately 25 in. Hg is maintained. In the unlikely case of a loss of the condenser vacuum as a result of the malfunction of an air ejector, the turbine would trip when the vacuum drops to approximately 22.5 in. Hg. When the vacuum drops to 7 in. Hg, operation of the turbine bypass system is prevented by the tripping of the turbine bypass valves. Additionally, when the condenser vacuum reaches 7 in. Hg, the main steam isolation valves are closed and the main steam safety-relief valves are opened. Two full-capacity, motor driven, air removal pumps are provided for startup to establish a vacuum of approximately 25 in. Hg prior to the transfer to the air ejectors. The condenser cooling water is pumped through the condenser under pressure. If leakage were to occur, the outside cooling water would leak into the condenser. Thus, the liquid condensate (or "radiated water" as the alleger refers to it) cannot leak against higher pressure to the outside of the plant.

The above quoted NRC response as the alleger stated it, is simplified but it is correct.

4. The fourth issue involves an allegation that a pressure switch installed on the high pressure coolant injection (HPCI) system was improperly installed. NRC Inspection Report No. 50-322/85-22, dated April 10 - May 10, 1985, describes the actions Region I was taking in regard to the installation of HPCI pump suction pressure switch PS-1212. The inspection report discusses the design and the setpoint methodology for this pressure switch and leaves one question and one assumption unanswered in the conclusion section (4.7.3). These were:

   (1) "justification for the nominal setpoint, as well as for the as-left reset pressure which is slightly outside of the required set-range, is required."

   (2) "the assumption that the impulse line is filled solid must be verified periodically."

   It is NRR's understanding that Region I personnel will be taking or have already taken follow-up actions to resolve these items. The NRR staff concurs in this course of action.

5. The fifth technical issue is part of two allegations, one related to an overheated recombiner and the other to operation of Hope Creek with two breaches in containment. According to the allegation, a licensee engineer at Hope Creek overheated a recombiner and failed to report it. The alleger stated that, during preoperational testing of the 'A' hydrogen recombiner on January 17, 1986, the inlet and outlet thermocouples were found to be reversed. The alleger stated that this condition resulted in overheating of the discharge pipe during the test in which the cooling water was turned off. The alleger is concerned that this overheating may have caused damage to the recombiner.

According to the NRC Inspection Report No. 50-354/87-22, dated November 24, 1987, an inspection was performed to independently verify the operability of the containment hydrogen recombiner system and to identify equipment conditions that might degrade performance. An inspector also reviewed the original copy of preoperational test procedures dated January 18, 1986. The inspector did not find evidence of overheating in his visual inspection. The preoperational test package did not indicate that reversed thermocouples had been found. A review of the technical manual provided by the recombiner manufacturer indicates that the cooling water referenced by the alleger is not necessary for the operation of the recombiner; it is an operational feature to limit the amount of heat discharged to the torus air chamber. Damage to the recombiner from overheating is prevented by a recombiner heater wall temperature high-high and a reactor chamber shell temperature high-high trip setpoints. The staff safety evaluation did not identify any safety-related function performed by the cooling water spray relating to the hydrogen recombiner other than limiting the exhaust temperature discharged to the torus. Therefore, on the basis of the inspection report and the staff's original safety evaluation, the staff concludes that no damage to the recombiner could have occurred.

In the second allegation, the alleger stated that during startup in October or November 1986, licensee personnel operated the Hope Creek reactor with two breaches in containment. According to the alleger, one of these was a 1-inch pipe going through the wall into the radwaste area that was open. This pipe was later identified as a 3/4-inch line at the post-accident sampling station that penetrated secondary containment according to Inspection Report 50-354/87-22, dated November 24, 1987. The second alleged breach was a 1/4-inch tube opening that penetrated the secondary containment. According to Inspection Report No. 50-354/87-22, the inspector conducted repeated inspections of the secondary containment boundary from May to September 1987, placing special emphasis on the areas mentioned

by the alleger. No breaches of containment were identified during these inspections; however it was not possible to determine precisely what conditions existed at the time of the alleged incident. The inspector did verify that containment integrity tests have been successfully completed, as required by the plant Technical Specifications (TS), and that administrative and technical procedures control the operation of the post-accident sampling station. The inspector verified that during plant operation, the secondary containment is maintained at a slight vacuum, in accordance with TS limiting condition for operation. This condition is verified twice a day. The staff concludes that the containment integrity tests satisfactorily demonstrate that no significant containment integrity breaches presently exist. The staff has no further information to add to the inspection report or to substantiate whether the alleged breaches occurred during the 1986 time period.

6. The sixth technical issue relates to over-ranging of instruments. In its memorandum, OIA indicates that the information for this issue is identified on pages 46-47 of the enclosed transcript of the interview with the alleger dated April 16, 1987. The information contained in these pages however is vague with respect to what instruments were involved and how they were over-ranged. Subsequent to the alleger's interview however, he sent a letter to the OIA investigators dated April 23, 1987 that is also enclosed with the OIA memorandum. In that letter he specifically addresses the question put to him by the investigators, "What would you like the NRC to do?" He states that he would like the staff to review the switches discussed on pages 12, 13, 14, 15, and 16 of Region I Inspection Report #50-322/85-22 to see if they agree with the inspector's findings regarding over-ranging of the instruments. Although it is not clear that this issue is the same as that identified by him during his interview, the following staff evaluation addresses the letter because it provides specific, detailed information.

The Region I inspection report referenced by the alleger addresses pressure switches PS-124 and 125 and their subsequent over ranging. These switches are used as interlocks in a reheat steam system and are located approximately 61 inches above turbine building floor elevation 37'6". The switches sense low pressure turbine inlet steam conditions and are not safety related. Actuation of these switches satisfy logic in (a) the open and close control circuits of valves 1N11*MOV 031A and B which isolate reheat steam to the moisture separator reheaters (MSRs) and (b) the circuits for MSR condensate drain valves SOV 07A and B, 08A and B, and 09A and B. These switches are used for normal operations and are not relied on to perform safety functions. As such, they have no impact on the safety analysis of the plant. Furthermore, the staff effort in the resolution of Unresolved Safety Issue (USI) A-47 has not identified this particular control system as one that has adverse action upon any safety system.