April 20, 2004

EA-04-086

Mr. Roy A. Anderson
Chief Nuclear Officer and President
PSEG LLC - N09
P. O. Box 236
Hancocks Bridge, NJ 08038

SUBJECT: HOPE CREEK NUCLEAR GENERATING STATION - PRELIMINARY WHITE FINDING (NRC Inspection Report 05000354/2003006)

Dear Mr. Anderson:

On February 11, 2004, the NRC issued Inspection Report 50-354/2003-006. This report documented a finding identified during an inspection completed on December 31, 2003. The finding, which was discussed at an exit meeting on January 21, 2004, involved service water traveling screen maintenance problems for which the safety significance had yet to be determined.

We have completed our evaluation of this finding using the At-Power Reactor Safety Significance Determination Process (SDP) as a potentially safety significant finding and have preliminarily determined it to be White, i.e., a finding with some increased importance to safety, which may require additional NRC inspection. The finding has low to moderate safety significance because the unavailability of the traveling screen due to inadequate maintenance increased the likelihood of the loss of service water initiating event and affected the ability of a service water pump train to mitigate the effects of initiating events. A copy of our risk significance determination is enclosed.

This preliminary White finding also appears to be an apparent violation of NRC requirements set forth in 10 CFR 50 Appendix B, Criterion V, "Instructions, Procedures, and Drawings," because during maintenance performed in June 2003 a traveling screen part was modified without instruction or procedure guidance and the traveling screen chains were not tensioned using applicable procedure guidance. Therefore, the violation is being considered for escalated enforcement action in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600. The current Enforcement Policy is included on the NRC's Website at http://www.nrc.gov/what-we-do/regulatory/enforcement.html.

We believe that we have sufficient information to make a final risk determination for the performance issue for this preliminary White finding. However, before the NRC makes a final decision on this matter, we are providing you an opportunity to either submit a written response or to request a Regulatory Conference. If you choose to request a Regulatory Conference, we encourage you to submit your evaluation and any differences with the NRC evaluation at least one week prior to the conference in an effort to make the conference more efficient and effective. At the Regulatory Conference or in your written response, your discussion of differences with the NRC risk evaluation should address all areas of uncertainty within the evaluation, including increases and reductions of the risk significance in both the internal and the external portions of the evaluation. If a Regulatory Conference is held, it will be open for public observation. The NRC will also issue a press release to announce the Regulatory Conference. If you choose a written response, such a submittal should be sent to the NRC within 30 days of the date of this letter.

Please contact Mr. Glenn Meyer at (610) 337-5211 as soon as you have decided and within a maximum of 10 business days of the date of this letter to notify the NRC of your intentions. If we have not heard from you within this time, we will issue our final significance determination and enforcement decision. You will be advised by separate correspondence of the results of our deliberations on this matter.

Since the NRC has not made a final determination in this matter, no Notice of Violation is being issued for this inspection finding at this time. In addition, please be advised that the characterization of the apparent violation may change as a result of further NRC review.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if you choose to provide one) will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. To the extent possible, your response should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the Public without redaction.

Sincerely,

/RA/

A. Randolph Blough, Director
Division of Reactor Projects

Docket No. 50-354
License No. NPF-57

Enclosure: Risk Significance Determination

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Risk Significance Determination

Statement of Performance Deficiency

The A station service water traveling screen was out of service from June 20 through 26, 2003 to perform corrective maintenance. This work was completed under work order 60037345 and included replacement of the screen head-shaft and sprockets. The screen was returned to service on June 26. On June 28 the shear pin on the A service water screen motor failed and was replaced.

On July 1 a self-revealing finding occurred when the A service water traveling screen stopped rotating due to the head-shaft shifting laterally and binding the screen. The screen was out of service for replacement of the damaged head-shaft and returned to service on July 9, 2003.

PSEG performed an apparent cause evaluation of the A service water traveling screen failure under order 70032466 and concluded the head-shaft shifted laterally while in operation due to maintenance personnel improperly cutting a vendor supplied shaft key in June 2003 without instructions or procedure guidance. PSEG identified a contributing cause regarding inadequate tensioning of the traveling screen carrier chains because of repeatability problems with the installed load cells used for this task.

The inspectors identified an additional casual factor regarding screen tensioning, in that although PSEG procedure HC.MD-PM.EP-0001(Q) provided qualitative and quantitative acceptance criteria for leveling the screen head-shaft and tensioning the screen carrier chains, the applicable section of this procedure was not included and completed in June 2003 under work order 60037345 to ensure the A service water traveling screen chains were tensioned correctly. Also, the description of work completed under work order 60037345 did not indicate this tensioning criteria was considered. PSEG initiated notification 20160886 to address this problem in their corrective action process.

NRC Inspection Report 50-354/2003-006 dated February 11, 2004 described an apparent violation of NRC requirements regarding inadequate maintenance work practices. 10 CFR 50 Appendix B, Criterion V, “Instructions, Procedures and Drawings” requires that activities affecting quality be accomplished in accordance with documented instructions, procedures or drawings appropriate to the circumstances, and that these documents shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to this, maintenance was completed under work order 60037345 from June 20 though 26 that trimmed a vendor supplied key without procedure guidance and tensioned the A service water traveling screen chains without using applicable acceptance criteria.

Significance Determination Process (SDP) Evaluation

SDP Phase 1 Screening

In accordance with Inspection Manual Chapter (IMC) 0612, Appendix B, “Issue Disposition Screening,” the inspectors determined that the issue was more than minor because it was associated with the equipment performance attribute of the initiating events and mitigating systems cornerstones. Specifically, the inadequate maintenance work practices resulted in the failure of the A service water traveling screen, which directly affected the ability of the A service water pump train to mitigate initiating events. It also resulted in an increase in the likelihood of the loss of service water initiating event.

In accordance with IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” the inspectors conducted a significance determination process (SDP) Phase 1 screening and determined that the finding degraded both the initiating event and mitigating systems cornerstones. Therefore, a SDP Phase 2 evaluation was required.

SDP Phase 2 Evaluation

The inspectors conducted an SDP Phase 2 evaluation of the risk significance of the performance deficiency and determined that the finding was of low to moderate safety significance (White). The inspectors used the following assumptions in the Phase 2 evaluation.

- The failure of the A service water traveling screen was attributed to inadequate maintenance work practices. As a result, the A service water pump train was unavailable for approximately 219 hours between June 28 and July 9, 2003, while the traveling water screen was repaired. Therefore, the inspectors used an exposure time of between 3 and 30 days.

- The A service water pump train being unavailable increased the likelihood of the loss of service water initiating event by one order of magnitude (from 5 to 4).

- The A service water pump train was not recoverable during the maintenance on the A traveling water screen.

SDP Worksheet Results:

LOSW - Loss of Service Water
LOSW (4) + CV (2) = 6
LOSW (4) + LI (3) = 7
LOSW (4) + HPI (2) + LPI (3) = 9
LOSW (4) + HPI (2) + DEP (3) = 9

Based on the SDP counting rule, this issue is considered to be of low to moderate safety significance (White) due to internal initiating events.

During the SDP benchmarking effort at the Hope Creek Nuclear Generating Station, the staff identified that the SDP Phase 2 process potentially overestimates the significance of inspection findings associated with one service water pump train by up to one order of magnitude.

Therefore, the analyst determined that the finding should be evaluated using the SDP Phase 3 process.

SDP Phase 3 Evaluation

Internal Initiating Events:

The NRC's Standardized Plant Analysis Risk (SPAR) model, Revision 3.03A, was used to evaluate the significance of this finding. The regional Senior Reactor Analyst conducted the SDP Phase 3 analysis using the following assumptions.

1. The analyst revised the model to include the NRC's current best estimate of both the likelihood of each of the LOOP classes (i.e., plant-centered, grid-related, and severe weather) and their recovery probabilities as described in NUREG/CR-5496, "Evaluation of Loss of Offsite Power Events at Nuclear Power Plants: 1980 - 1996."

2. The analyst updated the common cause failure alpha factors for the service water components.

3. The analyst revised the post processing rules to remove maintenance configurations that were prohibited by the facility's administrative procedures and Technical Specifications.

4. The performance deficiency resulted in the A service water pump train being unavailable for 219 hours between June 28 and July 9, 2003. Therefore, the analyst used an exposure time of 219 hours.

5. The analyst assumed that the increase in the loss of service water initiating event frequency was proportional to the ratio of the failure probability of the service water system with the A service water pump train unavailable and the base case failure probability of the service water system.

   Probability of insufficient service water (A SW pump train unavail) = 2.171E-4
   Probability of insufficient service water (base case) = 1.602E-5

   2.171E-4 ÷ 1.602E-5 = 13.55

   Loss of service water initiating event frequency = 1.1E-7 per hour

   (1.1E-7 per hour) x 13.55 = 1.49E-6 per hour.

   Therefore, the analyst assumed that with the A service water pump train unavailable, the loss of service water initiating event frequency was 1.49E-6 per hour.

6. The analyst assumed that the performance deficiency did not result in an increased likelihood of failure of the remaining service water traveling screens because the licensee's maintenance staff had not performed these maintenance activities on the other traveling screens and subsequent inspection of the traveling screens did not identify any degradation of these components.

7. The analyst assumed that the A service water pump train was not recoverable during the maintenance on the A traveling water screen.

The analyst revised the model to reflect assumptions 1, 2, and 3 which resulted in a baseline core damage frequency of 1.99E-5 per year. The analyst then revised the model to reflect the remaining assumptions, determined a revised core damage frequency for the degraded condition (6.55E-5 per year) and calculated the change in core damage frequency (ΔCDF) for this finding due to internal initiating events.

ΔCDF = [(6.55E-5 per year) - (1.99E-5 per year)]x[(219 hrs ÷ 8760 hours per year)]
= 1.14E-6 per year (White)

This result was dominated by the following accident sequences.

Contribution to ΔCDF	Core Damage Sequence Description
5.92E-7	• IE - Loss of offsite power • Emergency AC power fails (SBO) • Reactor core isolation cooling successfully provides inventory control until battery depletion • Operators fail to recover AC power within 5 hours
1.74E-7	• IE - Loss of service water • Operators fail to recover service water • All high pressure injection fails
1.21E-7	• IE - Loss of service water • Operators fail to recover service water • Operators fail to successfully vent containment
4.56E-8	• IE - Loss of offsite power • Emergency AC power fails (SBO) • Reactor core isolation cooling fails • High pressure coolant injection successfully provides inventory control until battery depletion • Operators fail to recover AC power within 5 hours

External Initiating Events:

High Winds, Floods, and Other External Events (HFO):

As documented in the licensee's Individual Plant Examination of External Events (IPEEE), the licensee examined HFO events using the progressive screening approach delineated in NUREG 1407. The licensee concluded that HFO events satisfied the screening criteria and did not significantly contribute to core damage frequency. Therefore, the analyst concluded that HFO events did not contribute significantly to ΔCDF.

Seismic:

The licensee's PRA staff estimated the risk significance of the A traveling water screen being unavailable due to seismic events using the information contained in the Hope Creek IPEEE.

The licensee’s approach involved evaluating each of the seismic damage states, determining which were impacted by the A service water pump train being unavailable, applying a conditional core damage probability for the appropriate seismically induced initiating event with the A service water pump train being unavailable, and calculating a change in core damage frequency. The licensee concluded that the change in core damage frequency due to seismic events was approximately 3.3E-8 per year. This result was dominated by seismically induced losses of offsite power that progressed to station blackout due to random failures of the emergency diesel generators.

The analyst reviewed the licensee’s methodology and concluded that it provided a reasonable approximation of the risk contribution due to seismic events. The analyst identified that the licensee screened several sequences as being negligible contributors that should have been quantified. When these sequences were included in the calculation, the change in core damage frequency due to seismic events was approximately 3.7E-8 per year. However, because the licensee’s evaluation used the EPRI seismic hazard curves, the analyst re-performed the evaluation using the Lawrence Livermore National Laboratory (LLNL) seismic hazard curves in order to estimate the uncertainty attributable to the choice of seismic hazard curves. The analyst determined that the change in core damage frequency due to seismic events using the LLNL seismic hazard curves was approximately 8.3E-8 per year. Therefore, the analyst concluded that seismic events did not contribute significantly to the change in core damage frequency.

Fire:

The licensee’s PRA staff estimated the risk significance of the A traveling water screen being unavailable due to fire events using the information contained in the Hope Creek IPEEE. The licensee’s approach involved evaluating each of the fire scenarios, determining which were impacted by the A service water pump train being unavailable, applying a conditional core damage probability for the appropriate fire induced initiating event with the A service water pump train being unavailable and calculating a change in core damage frequency. The licensee concluded that the change in core damage frequency due to fire events was approximately 5.0E-8 per year. This result was dominated by fires in the reactor and auxiliary buildings that resulted in a transient with closure of the main steam isolation valves. The analyst reviewed the licensee’s methodology and concluded that it provided a reasonable approximation of the risk contribution due to fire events. Therefore, the analyst concluded that fire events did not contribute significantly to the change in core damage frequency.

Potential Risk Contribution due to Large Early Release Frequency:

In BWR Mark I containments, only a subset of core damage accidents can lead to large, unmitigated releases from the containment that have the potential to cause prompt fatalities prior to population evacuation. Core damage sequences of concern for BWR Mark I containments are inter-system loss of coolant accident, anticipated transient without scram, small break loss of coolant accident, station blackout and transient sequences. Because the dominant accident sequences for this finding involved station blackout and LOSW transient initiators, the finding was screened for its potential risk contribution due to large early release frequency (LERF) using Inspection Manual Chapter 0609, Appendix H, “Containment Integrity SDP.”

The analyst determined that the dominant station blackout sequences did not result in a contribution to LERF because these sequences did not proceed to core damage until after the high pressure injection sources failed due to battery depletion several hours into the event. The analyst also determined that the accident sequence involving the unrecovered loss of service water with a failure of containment venting did not result in a contribution to LERF because these sequences resulted in core damage following containment failure. Thus, evacuation of the population would have been carried out in sufficient time so that these accident sequences would not have resulted in a contribution to LERF.

The analyst determined that the only dominant accident sequence that progressed to early core damage was the accident sequence involving the unrecovered loss of service water with a failure of high pressure injection sources (ΔCDF = 1.74E-7 per year). In this sequence, the licensee's emergency operating procedures directed the operators to manually depressurize the reactor and flood the containment using the fire water system. Therefore, the analyst concluded that this sequence was a low pressure core damage sequence that would occur with the containment flooded. The conditional containment failure probability from Appendix H of IMC 0609 for this category of core damage sequence is 0.1. Consequently, the analyst determined that the ΔLERF is approximately 1.74E-8 per year (Green)

Licensee's Risk Assessment:

The licensee performed a risk evaluation of the A service water traveling screen failures and concluded that the ΔCDF for internal events was approximately 2.4E-7 per year, the ΔCDF for external events was 8.3E-8 per year, and the ΔLERF was negligible. The analyst reviewed the licensee's results and concluded that one of the primary differences was in the loss of service water initiating event frequency assumed between the licensee's PRA model and the NRC's SPAR model.

The licensee's PRA model used a loss of service water initiating event frequency of approximately 1.98E-4 per year which was developed using a fault tree logic structure. The NRC's SPAR model uses the loss of service water initiating event frequency of 9.7E-4 per year published in NUREG/CR-5750, "Rates of Initiating Events at U.S. Nuclear Power Plants: 1987 - 1995." The licensee asserted that the initiating event frequency published in NUREG/CR-5750 was conservative, because it was developed using a rare event approximation of all of the industry loss of safety-related cooling water system events, regardless of plant design. The licensee's assertion focused primarily on the contention that the one loss of safety-related cooling water event, identified as the basis of the frequency in NUREG/CR-5750, was not applicable to Hope Creek because of the differences in plant design between the facilities. The licensee asserted that the initiating event frequency should be reduced to a value consistent with a frequency derived from data excluding this one event. However, the licensee's assertion did not include an evaluation of precursor events or unique site characteristics (e.g., seasonal detritus loading) which would tend to increase the likelihood of a loss of service water event at Hope Creek. The NRC recognizes that a total loss of service water is an unlikely occurrence; and as such, the estimate of its frequency contains greater uncertainty than events that occur more frequently. As a result, the NRC uses the initiating event frequencies published in NUREG/CR-5750 to ensure consistency in the significance determination process and because they represent the NRC's best estimate of the likelihood of these events.

The licensee's staff also indicated that alternate service water success criteria should be credited in core damage accident sequences that involved a failure of all residual heat removal provided the operators successfully vent containment. In these instances, one service water pump train would be adequate to remove heat from cooled components. The NRC typically does not credit selective failures that invoke alternative success criteria. In addition, modeling the service water system with alternate success criteria creates a level of complexity which is beyond the scope of the NRC's SPAR models.

In addition, the licensee's staff indicated that the NRC's analysis should give some credit for the operators ingenuity in finding a way to provide core cooling during conditions when service water was unavailable. For example, the licensee asserted that the possibility existed that operators could alternate the operation of equipment that required service water for cooling in such a fashion that the safety function could be satisfied without damaging the equipment due to insufficient cooling. The NRC recognizes that operator actions of this type may be possible; however, the NRC does not credit these types of heroic operator actions that are not proceduralized or trained upon.

Sensitivity Studies:

Case 1:

In order to assess the impact of performing the analysis using the loss of service water initiating event frequency published in NUREG/CR-5750 (9.7E-4 per year), the analyst re-performed the analysis using an initiating event frequency which was derived from industry data excluding the event challenged by the licensee (3.2E-4 per year). The remaining assumptions were unchanged. The analyst determined that the change in core damage frequency for this case due to internal and external events was approximately 1.03E-6 per year.

Case 2:

In order to assess the impact of not crediting alternate service water success criteria in core damage accident sequences that involved a failure of all residual heat removal, the analyst re-performed the analysis and applied a post processing rule to the dominant core damage sequences which required failure of containment venting in these instances. The remaining assumptions were unchanged. The analyst determined that the change in core damage frequency for this case due to internal and external events was approximately 1.06E-6 per year.

Conclusion:

The analyst determined that the safety significance of the inspection finding based on the increase in core damage frequency is White. The safety significance of the inspection finding based on the increase in large early release frequency is Green. Therefore, the safety significance of the inspection finding is White. A White finding represents a finding of low to moderate safety significance.