

January 10, 2005

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

SUBJECT: HOPE CREEK NUCLEAR GENERATING STATION - SUMMARY OF RESULTS
OF NRC REVIEW OF TECHNICAL ISSUES

Dear Mr. Bakken:

The purpose of this letter is to communicate the results of the Nuclear Regulatory Commission (NRC) staff review of two technical issues at the Hope Creek Nuclear Generating Station. Specifically, the first issue was associated with the 'B' reactor recirculation pump which has historically exhibited elevated vibration levels and has experienced other problems such as premature seal degradation. The second review looked at whether the exhaust piping for the high pressure coolant injection system had experienced a significant transient (i.e. a water hammer) that could have damaged the piping during the last refueling outage.

The enclosures to this letter provide summaries of the staff review of these technical issues. For the recirculation pump vibration issue, the staff concluded that your proposed continuous monitoring program for the Hope Creek reactor recirculation pumps provided reasonable assurance that a potential crack could be detected in time to allow the operators to take appropriate corrective actions prior to a shaft failure. We intend to confirm commitments related to the monitoring of the "B" recirculation pump and replacement of the pump shaft in a confirmatory action letter which will be issued prior to the re-start of Hope Creek.

For the high pressure coolant injection system exhaust line issue, the staff concluded that there was reasonable assurance that the high pressure coolant injection system exhaust line integrity had not been challenged by a "water hammer" event. Final documentation of these results will be included in NRC Inspection Report 05000354/2004005 which we expect to issue within 45 days.

Mr. A. Christopher Bakken, III

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Sincerely,

/RA/

Wayne D. Lanning, Director
Division of Reactor Safety

Enclosures:

1. Summary of Recirculation Pump Vibration Review
2. Summary of High Pressure Coolant Injection Exhaust Line Review

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

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Enclosure 1

Reactor Recirculation Pump Vibration Review

Background

The “B” Hope Creek reactor recirculation (RR) pump has had a historical problem involving high vibration levels—about double those on the “A” RR pump. Past licensee actions to reduce the vibration levels have not been effective. The high vibrations have been attributed, in part, to a slight bowing of the shaft in the area below the seal package area. The vibrations have led to frequent seal replacements (1.5-year intervals versus the expected 6-year intervals).

In addition to the bowing, the “A” and “B” RR pump shafts are expected to have some degree of thermally induced stress cracking based on industry operating experience described in GE Service Information Letter (SIL) 459. GE SIL 459 recommends three actions to address this problem: vibration monitoring, shaft inspections after about 80,000 hours of operation and action to mitigate the thermal stress initiators. Hope Creek’s RR pumps have over 130,000 hours of operation, and PSEG has not performed the recommended inspections.

In addition to the pump vibrations, there are vibrations on the associated RR and RHR system piping which have resulted in damage to system sub-components (MOV handwheel and limit switches). To date none of the vibration-induced component problems have rendered any safety-related system inoperable.

Sargent and Lundy (S&L) performed an independent assessment for PSEG which concluded that return of Hope Creek to service for the next operating cycle was acceptable given the current level of RR pump and piping vibrations. S&L’s conclusion was based upon data which indicated that the vibration level for Hope Creek’s “B” RR pump was consistent with RR pumps at other facilities and also based on an assumption that operators would be able to respond to an increasing vibration trend and take action to remove the pump from service prior to shaft failure.

The S&L assessment is summarized in the report, “Independent Assessment of Hope Creek Reactor Recirculation System and Pump Vibration Issues,” dated November 12, 2004. The staff reviewed the S&L report and developed a number of questions which were provided to the licensee on December 1, 2004. PSEG responded to the questions during a December 17, 2004, public meeting with the NRC. PSEG provided an additional response to the staff questions in a December 22, 2004, submittal. In addition, numerous teleconferences were held between PSEG and the NRC in December 2004 and January 2005 to discuss the “B” RR pump vibration issue.

The S&L Report concluded that there is no immediate need to replace the “B” pump rotor during the current refueling outage. S&L recommended that both pumps be monitored for vibrations and that a rapid rise in vibrations would be a sufficient reason to shut the pump down immediately for an internal inspection and shaft replacement, as the window between the rise in vibration and potential shaft failure is expected to be small.

PSEG also provided additional background information in Report H-1-BB-MEE-1878, "Hope Creek 'B' Recirculation Pump Vibration Analysis," Revision 1, dated December 16, 2004. The report concluded that, while the "B" RR pump has elevated vibrations when compared to the industry average, these vibration levels are not detrimental to the operation or reliability of the pump. The report also indicated that, although the risk of a RR pump shaft cracking event during any given cycle cannot be quantified, the operating experience of 29 RR pumps in operation longer than the Hope Creek "B" RR pump provides sufficient data to conclude that the risk of a shaft cracking event during the next cycle is minimal.

Staff Review

The staff review focused on the following key issues regarding the RR pump operation:

- (1) Does PSEG have a technical evaluation which shows that the RR pumps can be operated for another cycle without failure of the shafts considering the identification of shaft cracks that have been observed at other facilities with the same design RR pumps?
- (2) Can PSEG provide data which demonstrates that shaft cracks have been detected at other facilities with the same design RR pumps using vibration monitoring? Can the cracks be detected in time for the operators to take appropriate actions?
- (3) What are the consequences of a RR pump failure during plant operations?

GE SIL 459 indicates that all Byron Jackson RR pump shafts inspected have shown some degree of thermally induced cracking. The cracking occurs near the pump thermal barrier where mixing of cold seal purge system water and the hot reactor coolant water occur. The cracks initiate as axial cracks in the pump shaft. The licensee indicated that, if the cracks remain axial, the cracks will grow slowly and not affect the operation of the pump. However, the licensee also indicated that given sufficient mechanical loads, the cracks can become circumferential. The circumferential cracks can propagate to shaft failure under mechanical loading. The time it takes to transition from slow growing axial cracks to more rapidly growing circumferential cracks depends on the magnitude of the mechanical loads acting on the pump shaft. Since the licensee does not know the magnitude of the mechanical loads, it is difficult to predict the shaft life based on the magnitude of the operational loads.

The licensee has cited operating experience of other BWRs with similar Byron Jackson RR pumps. The licensee indicates that the age of the Hope Creek RR pumps is about average for the pumps of similar design at other BWRs. The staff notes that a number of the older pumps included in the licensee's comparison are much smaller than the Hope Creek pumps. While the operating experience provides some confidence that the pumps can be safely operated beyond the time interval recommended in GE SIL 459, the crack growth analyses provided by the licensee indicate that the time is highly dependent on the magnitude of the mechanical loads which are not well known.

The licensee also provided the level of vibration recorded at other BWRs with similar Byron Jackson RR pumps. The licensee concluded that measured vibration levels of the Hope Creek RR pumps are within the range of the vibration levels measured at other BWRs. However, the level of vibration of the "B" pump is toward the high end of the range of vibration levels measured at other BWRs. Therefore, the "B" pump is experiencing higher vibratory loadings than most of the pumps in the licensee's survey. In addition, the licensee cited a history of problems in its attempt to balance and align the pump shaft. These problems caused additional mechanical loadings on the pump shaft which could increase the potential for circumferential cracks to have developed in the shaft. On the basis of the above discussion, the staff concludes that the probability of a pump shaft failure of RR pump "B" during the next cycle of operation is indeterminate based on PSEG's evaluation of the potential thermal and mechanical loads on the pump shaft.

The licensee relies on vibration monitoring to detect circumferential cracking of the RR pump shaft with sufficient lead time for operators to secure the pump from complete shaft failure. The licensee developed a plan for monitoring the vibration levels of the RR pumps. The key elements of the plan involve continuous basic monitoring of the overall level of vibration and continuous monitoring of the vibration harmonics for enhanced detection capability of potential shaft cracking.

The licensee's continuous basic vibration level monitoring by the operations department consists of a pump vibration alarm and pump speed reduction if the "B" pump vibration level reaches 11 mils (0.011 inch), and removal from service if the pump vibration level reaches 16 mils (0.016 inch). The continuous monitoring of the vibration harmonics consists of pump vibration alarms and pump speed reduction if the synchronous speed (1X) vibration amplitude, two times synchronous speed (2X) vibration amplitude, 1X phase angle, or 2X phase angle exceed defined allowable limits. If the monitored values do not fall within their allowable limits at the reduced pump speed, the licensee will remove the RR pump from service. The allowable limits are established using ASME OM Standard, "Reactor Coolant and Recirculation Pump Condition Monitoring." The licensee will record baseline data to establish these allowable limits during plant startup. The licensee provided two technical papers in support of the proposed vibration monitoring criteria.

The first technical paper is entitled, "Case History Reactor Recirculation Pump Shaft Crack," Machinery Messages, December 1990. The paper discusses the RR pump shaft cracking experience at the Grand Gulf nuclear power plant. The paper indicates that the vibration level increased rapidly over a three hour period before the pump was secured at slow speed. Although the shaft did not experience a complete failure, subsequent inspection revealed the shaft was cracked approximately 320 degrees around the circumference. The paper indicates that it is necessary to monitor the 1X and 2X steady state vectors (1X and 2X amplitudes and phase angles) on a continuous basis and to compare these monitored values to an acceptance criteria. The paper also indicates that alarms are necessary to alert the user to amplitude and phase deviations that are outside the acceptance criteria.

The second paper is a Technical Bulletin from Bently, Nevada, "Early Shaft Crack Detection on Rotating Machinery Using Vibration Monitoring and Diagnostics." The technical bulletin indicates that shaft cracking can be detected by monitoring the 1X and 2X vectors. The technical bulletin also recommends continuous monitoring of machines that are susceptible to shaft cracking.

These papers recommend using continuous monitoring of the 1X and 2X vectors as a predictive method to detect significant shaft cracking. The staff requested that the licensee provide some evidence that vibration monitoring was effective for detecting shaft cracks in RR pumps similar to the Hope Creek RR pumps. The licensee cited the experience at Grand Gulf discussed above. The Grand Gulf RR pump shafts are hollow shafts as opposed to the solid shafts used in the Hope Creek RR pumps. Therefore, the Grand Gulf experience may not be directly applicable to Hope Creek. The licensee provided additional information which indicates that cracks in reactor coolant pump shafts were identified at Sequoyah (technical presentation to NDE Steering Committee by G. Wade, July 12, 2002) and Palo Verde Unit 1 (Palo Verde Nuclear Generating Station Cracked Reactor Coolant Pump Shaft Event, H. Maxwell, 1996) using vibration monitoring. Although these plants are Pressurized Water Reactors (PWRs), the reactor coolant pumps have solid shafts. The licensee indicated that these pumps had operated for a significant period of time after the first indication of shaft cracks by vibration monitoring. A staff review also identified that vibration monitoring successfully identified a reactor coolant pump shaft cracking at St. Lucie Unit 2 (LER Number: 1993-005). The PWR reactor coolant pump experience provides some indication that a solid pump shaft will provide better early crack detection capability than the hollow pump shafts, such as those used at Grand Gulf. PSEG has provided data which demonstrates that shaft cracks in pump shafts similar to those used at Hope Creek have been detected at other facilities, and that these cracks were detected in time for operators to take appropriate actions.

On the basis of the available operating experience, the staff concludes that continuous monitoring of the 1X and 2X amplitudes and phase angles provides reasonable assurance that circumferential shaft cracking can be detected with sufficient time for the plant operators to take appropriate actions. The licensee will either reduce the RR pump speed or remove the pump from service if the monitoring system detects vibration levels that exceed the limits specified in the vibration monitoring plan.

The staff also reviewed the licensee's assessment of the potential consequences of a RR pump shaft failure. The RR pump shaft axial cracking that has been reported occurred below the seal area and above the pump hydrostatic bearing. This is the region where a potential RR pump shaft failure would be expected to occur. The pump impeller would be expected to settle at the bottom of the pump casing, which could potentially result in some damage to the pump casing. The unsupported end of the upper part of a broken shaft may damage the shaft seal. A seal failure would result in leakage of reactor coolant through clearances around the upper half of the broken pump shaft. This leakage would be bounded by the design basis small LOCA event. If such an event were to occur, the licensee would be able to isolate the pump using the RR loop isolation valves, thereby terminating any reactor coolant system leakage.

Conclusion

The staff concludes that the licensee's continuous monitoring program for the Hope Creek RR pumps, as discussed above, provides reasonable assurance that a potential crack in the RR pump shaft can be detected in time for operators to take appropriate actions to reduce the pump speed or remove the RR pump from service prior to a complete shaft failure.

Enclosure 2

High Pressure Coolant Injection (HPCI) Exhaust Line Review

Background

On November 1, 2004, with the plant in Mode 5 for refueling outage 12, tandem snubbers from the HPCI turbine exhaust piping failed during dynamic testing. A followup inspection of the HPCI piping resulted in the observation of a damaged pipe support and a snubber anomaly that could have been the result of a water hammer event in the HPCI turbine exhaust line. A subsequent licensee evaluation of the reported observations found that there was no conclusive evidence that a water hammer had occurred in the HPCI turbine exhaust line.

Staff Review

The licensee provided an assessment of the tandem snubber failures performed by the snubber manufacturer, Lisega. The snubber failures occurred in the fluid reservoirs. Lisega indicated that the fluid reservoir failures were caused by stuck poppet valves that allowed fluid to leak into the reservoir during testing. Lisega concluded that repeated testing of the HPCI snubbers in compression resulted in over-pressurization of the reservoirs. Lisega also indicated that the snubbers would have functioned in response to a seismic event. The licensee's assessments of the other observations, identified during the initial inspection of the HPCI exhaust line, provided reasonable dispositions of the observed conditions.

A licensee inspection of the accessible portions of the HPCI exhaust line in the turbine room and the torus room found no evidence of large pipe distortion or excessive pipe movement at support locations which likely would have been present if a water hammer had occurred. This was confirmed by the NRC inspectors. The licensee also performed non-destructive examination (NDE) of all field welds on the 20 inch HPCI exhaust line. All welds were found satisfactory. The inspections and weld examinations performed by the licensee are the type of actions the NRC staff would require after a water hammer event.

Conclusion

The licensee provided plausible explanations for the snubber failures that occurred during snubber testing and for the identified support damage and snubber anomaly identified during the followup HPCI inspection. In addition, the licensee performed the type of inspections and NDE examinations that the NRC would require after a water hammer event and found no adverse results. Therefore, the staff concluded that there was reasonable assurance that the integrity of the HPCI exhaust line had not been challenged by a water hammer event.