PSEG NUCLEAR LLC

HOPE CREEK GENERATING STATION

INDEPENDENT ASSESSMENT OF HOPE CREEK REACTOR RECIRCULATION SYSTEM AND PUMP VIBRATION ISSUES

November 12, 2004

Sargent & Lundy LLC

Project No. 11050-356

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Date: November 12, 2004

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EXECUTIVE SUMMARY

This independent assessment was authorized by Mr. A. Christopher Bakken to provide a complete, accurate, and credible evaluation of the issues associated with the Hope Creek Reactor Recirculation (RR) System and pump vibrations. The assessment was chaired by Mr. John Carlin. The assessment was performed by an S&L team of experts. The assessment summary follows.

RR Pumps and Seal Assessment

An investigation was made to determine Hope Creek experiences with the (RR) pumps and seals, and to assess their present conditions for future operability. Pertinent documentation was reviewed, and discussions were held with Station personnel and representatives of Flowserve, the pump and seal manufacturer.

The “B” pump vibrations have historically been about twice as high as the “A” pump but both are in the range of vibration levels that are within industry experience. The “B” pump seals have approximately ¼ the life expectancy of the “A” pump seals. The industry expects a six year life for seals, which the “A” pump generally meets.

There is no trend of increasing vibrations for either RR pumps. Therefore, the vibration data does not indicate signs of degrading condition for either pump. The predominant ‘B’ RR pump shaft displacements occur at the pump running speed. This is consistent with unbalance or a bent shaft.

The RR pumps speed is limited to 1510 rpm, and at this speed the maximum core flow achieved in the past was 103 million pound mass. During the last operating cycle the flow dropped to approximately 100 million. The reduction may be due to instrument changes, RR pump degradation, or to jet pump fouling. If the degraded flow is deemed a significant operational concern, it should be investigated.

Alignment of the “B” pump coupling and checking alignment when the pump is recoupled during RF12 is recommended. In addition, the coupling should be checked for concentricity and squareness, and balanced.

The present RR pump seals are an old design that has a carbon stationary seal ring. An updated Flowserve seal design includes a silicon carbide ring, which is a more durable material. Both the “A” and the “B” RR pumps should have the updated seals installed when the present seals are replaced.

Subsequent to RF12, the “A” pump should be monitored with the “B” pump, for capacity and vibrations. A rapid rise in vibration amplitude would be sufficient reason to shut the pump down immediately for an internal inspection and rotor replacement, as the window between the rise and potential shaft failure is expected to be small.

Based on the above, there is no immediate need to replace the “B” pump rotor during RF12. However, in consideration of the RR pumps service hours, the pumps should be upgraded by installing Generation IV rotor components that were developed by Flowserve post GE SIL 459S2. The upgrade should be made on the “B” pump during RF13. The “A” pump upgrade may be deferred to RF14 unless performance and vibration monitoring indicate degradation prior to RF13. If degradation is evident, it too should be upgraded during RF13.

Integrated System Assessment

Reviews of completed and planned RR and RHR system vibration evaluations were performed. The vibration monitoring scopes, acceptance criteria and evaluation of monitoring results were reviewed.

A vibration monitoring effort was implemented following the March 2004 outage. The piping vibrations measured following the March 2004 outage were acceptable and indicated, based on comparisons with previous vibration monitoring results, which vibrations have not changed significantly over time. RHR valve components were not evaluated. The 2004 monitoring results indicated that the likely excitation source for the RHR valve component vibrations is pressure pulsations at the RR pump vane passing frequency.

The planned RR and RHR vibration monitoring scope for Extended Power Uprate (EPU) is not sufficient for verifying that pipe stresses are acceptable, evaluating valve components, or acoustic model benchmarking. Specific recommendations are provided in the body of the report.

Acceptance criteria have not yet been developed for RR and RHR EPU piping vibration monitoring. Assessment comments relative to the acceptance criteria developed for the 2004 monitoring effort should be considered when developing the EPU acceptance criteria. Acceptance criteria for the monitored RHR valve components also need to be developed.

EPU vibration data should be collected at predetermined pump speeds or power levels during power ascension up to the maximum speeds at which the RR pumps will be operated. Data should also be collected during the RHR shutdown cooling mode of operation and during planned downpower evolutions.

Failed Components Assessment

Prior to the March 2004, failed and degraded hardware in the RR/RHR system inside drywell were either addressed by Design Engineering on a case by case basis or addressed by the station as a repair and replace activity.

As a result of the findings identified during the March 2004 outage, a collective review and index of hardware failures were initiated by NUCR 70037702 through the developed Common Cause Report, H-1-BB-CEE-1862. As a result of that set of investigations and the recommendations, additional Operations (Tasks) under NUCR 70037702 were assigned to each activity and are being tracked. In addition, as new information becomes available, additional tasks are added to the overall scope of NUCR 70037702.

DCP 800722673 is currently being prepared to address selected recommended corrective actions tracked by NUCR 70037702. Not all design and analysis elements have progressed to a point to permit a meaningful review at this time. Selected portions of the available supporting analytical documentation were reviewed to evaluate the consistency of method with the intended results. This assessment includes a review of past and current component failures, system modification documentation and considered several attributes including those related to the current status of Design Engineering activities, Design Inputs, and Design Outputs. The assessment recommendations are:

- Perform a separate failure mode assessment using the Finite Element Model (FEM) results for the F060A/B and F077 valve operator assemblies.

- The criteria for the modification of the valve operator assemblies should proactively consider EPU pump speeds and associated 5X frequencies with margin and be based on 150 Hz or greater.

- The proposed post modification testing of the manual gate valve top works should include collection of test data and be implemented to ensure goals are met. This includes the collection of data during cycle 13 at pump speeds above 1500 RPM.

- A disassembled valve inspection should be done to conclusively determine the current condition of the F050A/F060A valve internals. If indications are noted for the F060A valve, as a minimum, a similar inspection of valve F060B should be conducted.

- The noise monitoring system should be implemented for the full operating cycle rather than be only implemented during power ascension.

- Implement F050A valve operator modifications.

- The removal of the hand wheels for valves F060A/B and F077 has been evaluated and judged to be acceptable. This modification should be implemented.

Assuming that the above recommendations are implemented, we have high confidence that the RR and RHR system components will perform satisfactorily post RF12.

<u>NDE Assessment</u>

The ISI program as applied to the RR system and its interface with the RHR system was found to be in accordance with ASME Section XI requirements, Code Case 578-1 and EPRI TR-112657 for risk informed programs have been implemented as required.

Early small bore connection vibration fatigue failures have been effectively addressed and examinations of previously failed RR connections and RHR connections near areas of equipment damage, conducted since 2001 have not identified additional occurrences of vibration fatigue failures. One small bore weld was noted during radiographic examination to have an existing fabrication anomaly. These small bore examinations are planned to be discontinued after RF12 based on the acceptable results. It is recommended that the weld that contains the fabrication anomaly continue to be subject to augmented ISI examination, while system vibration issues continue. In addition, it is recommended that small bore connections be examined in RR and RHR system where equipment/hardware damage is or will be noted in the future, while system vibration issues continue.

The RR pump shafts are included in the ISI program, but require pump disassembly to conduct meaningful SIL-578 recommended examinations. These examinations have not been performed, which is consistent with US industry practice.

Two large bore RR lines found previously with surface indications during ISI were caused by shop welding material issues and no vibration fatigue issues have been identified. Vibration monitoring performed subsequent to the March 2004 forced outage show that system vibration is well below design allowable levels. Thus, additional NDE is not necessary.

Based on the NDE information reviewed for this assessment, we have high confidence that the RR pump vibrations have not degraded the system's piping capability to perform its design functions.

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1. INTRODUCTION

This independent assessment was authorized by Mr. A. Christopher Bakken to provide a complete, accurate, and credible evaluation of the issues associated with the Hope Creek Reactor Recirculation (RR) System and pump vibrations.

The assessment was chaired by Mr. John Carlin. The assessment was performed by an S&L team of experts as follows:

Donald Cavi - Pump
Ismail Kisisel - Component Engineering
Keith Miller - NDE
Paul Olson - Component & Piping
Brian Voll - Vibrations
A. K. Singh (Lead)

2. ISSUES EXAMINED

Issues examined in this assessment are consistent with the Charter for the assessment (Appendix A) developed by PSEG. These issues are:

A. Assess the condition of the "B" Reactor Recirculation (RR) pump.

B. Evaluate the "B" RR pump performance and integrated recirculation system performance including:

   - Whether the continuing vibrations being detected in the Hope Creek drywell may have been caused by the RR pump operation.

   - Whether vibrations may increase during transients or when the unit is down-powering and how other components in the drywell have been or could be affected by this high vibration including valves and limit switches.

C. Review the PSEG Nuclear proposed plan including the historical and proposed corrective actions including:

   - Adequacy of the corrective actions to address pump and structural vibration concerns including the input basis for the corrective actions (inspections, data, etc.)
   - The current NDE program and past inspections results. Make recommendations for any changes that may be warranted.
   - Review available accelerometer data, evaluate the basis and adequacy of the planned probe placement and provide any recommendations for improvement.
   - Review the validity of PSEG proposed acceptance criteria for post-refueling pump and structural vibration levels.

- Evaluate the root cause report and assess the recommendation to replace the pump rotor, the timing of that replacement and any follow-up actions that were recommended.

3. ASSESSMENT APPROACH

The assessment was performed to address the issues identified in the scope of work. The assessment was performed by respective S&L subject matter experts based on information acquired through the following:

A. Interviews, meetings, and telecon with PSEG and RR pump manufacturer personnel
B. Root Cause and Common Cause reports on the issues
C. Measured vibrations data
D. Planned corrective actions to resolve the issues
E. Design basis documents
F. Applicable information from other BWR MARK I plants

Planned corrective actions and supporting evaluations are in progress. The assessment was based on the best available information as of November 11, 2004.

4. PERSONNEL INTERVIEWED

As part of the independent assessment, the S&L team interviewed or had meetings with the following PSEG and vendor personnel familiar with the issues related to the RR pump vibrations and its effect on the RR and Reactor Heat Removal (RHR) systems and components. These individuals included:

Peter Koppel	–	Component Engineering
Joseph Flanagan	–	Vibration Engineer
Richard Herron	–	Vibration Engineer (Maplewood)
Peter Kordizel	–	RR or Reactor Recirculation System Engineer
Daniel Boyle	–	Operations Assistant Manager
Mike Reed	–	Operations
Randall Decker	–	Flowserve Pump Engineer
Art Olson	–	Flowserve Seal Engineer
John O'Neal	–	ISI Program Administrator
Tom Roberts	–	NDE Level III
Wayne Denlinger	–	NDE Level III
Alan Johnson	–	Civil/Structural Supervisor
John Barkhammer	–	Civil/Structural Engineer
James Stavely	–	Reactor Engineering Supervisor
Paul Durant	–	NDE
Howard Berrick	–	Licensing
Michael Moser	–	Licensing

5. ASSESSMENT OF THE “B” RR PUMP

A. Background

The Hope Creek “B” Reactor Recirculation Pump (RRP) has a long history of premature shaft seal failures, accompanied by vibration levels that are about twice as high as the “A” RRP. Several abnormal events have contributed to the seal failures, which are correctable; however, history indicates that even during times when operation is normal (for the “B” RRP), the seals fail short of the six-year expected life. Typical life of the seal in the “A” RRP is six years.

The Station has three Flowserve N-7500 cartridge type seals. Two are installed in the “A” and “B” pumps, and one is a spare. One of the seals is suspected of being less reliable than the other two; however, it is rotated in the pumps (usually pump B) on an as needed basis, following a failure.

Failed seals are rebuilt in the Station shop using new parts purchased from Flowserve the pump and seal manufacturer.

The pumps are driven by variable speed motors, having a rated speed of 1680 rpm. However, when the pump speed is above 1529 rpm, there are reports that the noise level in the reactor recirculation system sounds like a “freight train.” Consequently, there is an administrative limit of 1510 rpm on pump speed.

It was reported by operation personnel that at the speed level of 1510 rpm, flow in the reactor core recirculation system has gradually decreased from 103 million lbm./hr. to about 100 million lbm./hr. This reduction in reactor core flow is a concern.

B. Measured Pump Vibration

The RR pump vibration levels and trends were reviewed to identify signs of excessive vibrations or pump degradation. The available vibration data were in terms of pump shaft displacement measured by two proximity probes on each pump. Trends of overall vibration amplitude dating back to 1997 are provided in Exhibits 5-1 and 5-2. The ‘A’ RR pump vibrations are generally 4-6 mils, other than some unexplained aberrations between RF08 and RF10, with no apparent trend up or down. Prior to RF09 and after RF11 the ‘B’ RR pump vibrations are generally 9-11 mils with no apparent trend up or down.

Between RF09 and RF10 the ‘B’ RR pump vibrations were significantly higher. According to the Hope Creek rotating equipment vibration engineer, this increase was due to the unsuccessful application of balance weights to the pump. According to the Hope Creek component engineering supervisor, this increase was due to out-of-roundness of the coupling where the proximity probes were installed. After the balance weights were removed and the proximity probes were moved, the vibration amplitudes dropped back down to lower levels. In fact, between RF10 and RF11 the ‘B’ RR pump vibrations were at their lowest levels, approximately 6-9 mils. The reason for the step change increase after RF11 is not known. However, the vibration levels after RF11 were no higher than the levels prior to RF09. Since there is no trend of increasing vibrations for either pump, the vibration data does not indicate signs of degrading condition for either pump.

The “B” pump vibrations are approximately two times higher than the “A” pump vibrations. This is consistent with the “B” pump mechanical seals having reduced life and requiring more frequent replacement than the “A” pump mechanical seals. However, the RR pump vendor has indicated that the “B” pump vibration levels are acceptable and comparable to other plants. For comparison purposes, vibration data obtained for the RR pumps at Browns Ferry Unit 2 during power ascension is provided in Exhibit 5-3. The maximum shaft vibrations are 7 to 7.5 mils for the “B” pump and 9 to 9.5 mils for the “A” pump. Both of the Browns Ferry RR pumps have higher vibrations than the Hope Creek “A” RR pump. The most recent Hope Creek “B” RR pump vibrations are only slightly higher than the Browns Ferry “A” RR pump vibrations.

The predominant “B” RR pump shaft displacements occur at the pump running speed. This is consistent with unbalance or a bent shaft. The two times the pump running speed (2X) and five times the pump running speed (5X) harmonics also contribute to the overall shaft vibration.

C. Vibration Monitoring Instrumentation

Currently the RR pumps are instrumented with two proximity probes to measure shaft vibration and one sensor to measure radial velocity at the top of the motor. No vibration data for the velocity sensor could be located for this assessment. This is likely consistent with the amount of vibration instrumentation initially installed on RR pumps at other plants but less than what is currently installed at other plants.

According to the Hope Creek component engineering supervisor, additional instrumentation is being installed in conjunction with the EPU vibration monitoring instrumentation. The added instrumentation will consist of two radial accelerometers at the lower motor bearing and two radial accelerometers and one axial accelerometer at the upper motor bearing. This is likely more in line with how the RR pumps at other plants are instrumented and should be sufficient for monitoring the equipment condition. It is recommended that a review of the RR pump instrumentation at other plants be performed, including how the data from the instrumentation is being used, to help verify that the type and amount of instrumentation being added is appropriate.

At this time, the instrumentation being added is being installed per a temporary modification. This implies that the instrumentation will be removed after the EPU vibration monitoring effort is complete. An effort should be initiated to make the instrumentation a permanent installation. This would include the data acquisition and recording devices and control room interfaces that would be required for the permanent installation.

D. Pump Support Configuration

The Hope Creek support configuration was compared to Dresden, Quad Cities, Browns Ferry, and Clinton RR pump support configurations. This comparison is presented in Appendix C. The key conclusions are:

- The Hope Creek RR pump and motor supports are similar to other plants except that Hope Creek motor supports consist of two snubbers compared to three snubbers for the motor supports for the other plants. The 3rd motor support snubber at Hope Creek was removed during the snubber reduction project. Thus, the RR pump motor at Hope Creek is some what less restrained than other Mark I plants and Clinton (Mark III).

- The Hope Creek RR pump casing is restrained at the bottom by a rigid restraint that has the potential to constrain free thermal movement. Other MARK I plants do not have this rigid restraint.

The difference in Hope Creek "A" and "B" pump rigid restraint configuration should be investigated to ensure that it is not the cause for the high "B" RR pump vibrations.

E. Pump Vibration Assessment

Typical overall shaft vibrations during normal operations are 4-6 mils peak-to-peak on the "A" pump, and 9-11 mils peak-to-peak on the "B" pump. (Refs. 5.9 and 5.10)

It is generally accepted in the industry that high vibrations contributed to premature wear and possibly failure of pumps components. Typical and correctable causes of high pump vibrations are:

- Rotor out of balance
- Misalignment with the driver shaft
- Bowed shaft (runout)
- Damaged coupling (not squared and concentric to pump and driver shafts)
- Coupling out of balance

<u>Rotor Balance</u>

Prior to RF10 (2001), an attempt was made to investigate rotor balance on the "B" pump. The results were inconclusive. Flowserve feels rotor balance is not a problem. Balancing can only be accurately checked when the rotor is removed from the pump which has not been done. Therefore, rotor unbalance remains a possibility. (Ref. 5.23)

<u>Pump to Driver Alignment</u>

Alignment between the driver and pump shafts was checked during RF09 (2000). A significant misalignment was found and corrected. The effect on "B" pump vibrations was a minor reduction in amplitude. Vibrations are still twice those in "A" pump. (Ref. 5.23)

Bowed Pump Shaft

An industry survey shows that RR pumps at other stations with similar Flowserve pumps have experienced bowed pump shafts. Quad Cities reported having a shaft with 8 mils runout due to bowing of the shaft. Flowserve acknowledges this fact and advised that bowing typically occurs at the thermal barrier section of the shaft which is located just above the hydro-dynamic bearing, and below the seal. Operability has not been affected because of shaft bowing. (Refs. 5.2 and 5.24)

Flowserve inspected a seal that failed in February 2003 in the "B" pump. Examination of the wear patterns led Flowserve to conclude the pump shaft is bowed. (Ref. 5.2)

During RF12 (2004), the station conducted two tests with the rotor still in the pump to determine if the shaft is actually bowed. (Ref. 5.7)

One test used a temporary bearing to support the shaft which was uncoupled from the motor. The temporary bearing sat loosely on a machined surface at the bottom of the seal chamber. Any runout in the shaft would put an unbalanced load on the temporary bearing allowing one side of the bearing to lift off the machined surface. Movement of the shaft was measured in the radial direction at several points in the seal chamber area. Results were not conclusive as the data was not repeatable. However, it did indicate a possible bow in the shaft.

The supplemental test involved taking radial shaft readings at several locations in the seal chamber area while coupled to the motor. The pump was then de-coupled and the impeller allowed to rest on a machined surface at the bottom of the casing. Shaft readings were taken as done with the pump coupled. The coupled and uncoupled measurements were compared to determine if the upper shaft had tilted, which would indicate shaft runout, assuming the machined resting surfaces on the impeller and casing were true and free from dirt accumulation. The measurements indicated shaft runout (bow) of about 4.8 mils.

While the tests are not conclusive, they indicate that shaft distortion, such as bowing, may exist.

It appears unlikely that a shaft as heavy and robust as those in the RR pumps would be bowed from a mechanical malfunction without serious consequences. None were reported; therefore, it can be assumed the shaft did not bow mechanically. If bowing did occur due to thermal effects during operation, it seems logical the shaft would straighten itself when shut down and temperatures are equalized. The shaft tests indicate the shaft may be bowed even with equalized temperatures. (Ref. 5.7)

The pump cross-section drawing (Ref. 5.5) shows that the hydrodynamic bearing journal is welded to the shaft just below the thermal barrier. The shaft could have had residual stresses due to welding that were relieved while the pump was in operation, causing the shaft to take a permanent bow. If in fact a bow does exist, this would contribute to vibrations at operating speed frequency, which are experienced.

Damaged Pump Coupling

The drive coupling was designed and provided by the pump manufacturer, and is a rigid, adjustable (in the axial direction), spacer type. During RF10 (2001), it was discovered that the coupling was out of round. The flange at the lower end of the spacer, where the proximity vibration probe was focused, was found to be egg shaped and caused erroneous vibration readings. The coupling rabbet fits were machined to restore alignment with the driver, and the probe was moved to focus on the pump coupling flange O.D. which is considered to be round. (Refs. 5.23 and 5.24)

It was stated by Flowserve, however, that the pump coupling flange O.D. is a machined surface but there are scratches on the surface that could distort vibration readings by 1 or 2 mils. (Ref. 5.24)

Coupling Balance

A sizable force would be required to distort the coupling as it was. The force could cause the coupling to become unbalanced. The coupling was not balanced after machining, and could therefore be a contributor to high vibrations. (Ref. 5.24)

F. Impact of GE Nuclear SIL 459 S2

GE issued SIL No. 459 S2 (Ref. 5.21) which states that RR pumps with more than 80,000 hours of operation should be inspected, because shaft cracks were detected at all GE BWRs at which shafts were removed and inspected, although some were still operating after more than 100,000 hours of service. Hope Creek RR pumps have exceeded this and are at approximately 130,000 to 140,000 hours. (Ref. 5.24)

Flowserve also reported surface cracking on the shaft at the thermal barrier, which is likely to be caused by temperature differences between the hot pumped water and the relatively cool water which is recirculated through the seal and a cooling coil in the pump cover. A small flow, approximately equal to the seal purge flow, migrates into the pump through the thermal barrier. (Ref. 5.24)

The SIL (Ref. 5.21) recommends that shaft vibrations be monitored to detect possible severe shaft cracking in time to take corrective action before a shaft failure. Some pumps manufactured by Flowserve (Byron Jackson was the OEM) that were inspected had shaft cracks ranging from 0.1 in. to 0.6 in. in depth, with service times ranging from 26,000 to 116,000 hours. The crack depth did not correlate to service time or vibration. If vibration increases rapidly, it could indicate severe cracking occurred. In that case, the pump should be shut down quickly to avoid imminent failure as the window between the vibration increase and shaft failure is expected to be small. (Ref. 5.24)

See Section C for planned vibration monitoring which describes current and planned vibration instrumentation, and is sufficient to satisfy the SIL recommendation.

G. Pump Condition

Assessment

Operations of the “B” RR pump does not appear to be impaired by the vibration levels that were experienced. Vibration levels were stable following RF10 and RF11. (Ref. 5.9) Amplitudes after RF11 were somewhat higher than after RF10. Seals on the “B” pump were replaced during RF11 and the coupling was removed to facilitate the work. The reassembled alignment with the motor may not have been as accurate during RF11 as during RF10, which could cause the increase. However, amplitudes on the “B” pump were stable from May 2003 to the end of September 2004.

In addition, overall shaft orbit plots (Ref. 5.9) taken on May 13, 2004, July 8, 2004, and September 27, 2004, are very similar which indicates the rotor dynamic characteristics have not degraded in the recorded time span. Flowserve indicated that similar pumps in other plants are operating with higher vibration amplitudes than the “B” pump, and operability has not been a problem. (Ref. 5.24)

It is therefore likely that the “B” pump can operate in its present configuration until RF13, which is expected to occur after 18 months of operation.

Pump Upgrade

Parts are on hand at Hope Creek to upgrade the RR pumps. The rebuild parts are a Flowserve fourth generation design which was made following the GE SIL459S2 issue. The existing Hope Creek RR pumps are the Flowserve second generation design, however, the new fourth generation rebuild components are designed to fit the Hope Creek pumps.

H. Seal Concerns

Leakage

In a telephone conference call on November 9, 2004, the Flowserve’s mechanical seal expert was asked if axial vibrations could be tolerated by a mechanical seal, and what would be the limit of vibration. He responded he has not seen a mechanical seal that failed due to axial vibration, and it appears not to be a threat.

He was then asked how much radial vibration could safely be tolerated. His response was that vibrations are not as much a factor as shafts that rotate in an eccentric orbit. The radial sliding motion of the rotating seal face in an eccentric orbit can capture suspended particles that may be on the stationary face, thereby causing the particle to become lodged between the sealing surfaces and subsequently cause a leak path.

It is noted that the seal removed from the "B" pump in March 2003 had wear bands on the rotating seal faces that were 1/64" (15 mils) wider than the noses on the stationary faces. (Ref. 5.2) This indicates the shaft was rotating in an eccentric orbit that was offset from the true centerline. The effect is likely caused by shaft runout (bowed shaft), or misalignment between the pump and motor shafts. If the condition exists after RF12, premature seal failure can again be expected on the "B" pump.

When pressure in the pump is reduced as during a reactor shutdown, the shaft will move downward by as much as 14 mils due to relaxation of the upthrust load on the motor thrust bearing. If in the process the stationary seal ring binds with the O.D. of the balance sleeve due to friction, the seal faces will open, allowing particles to enter between the seal faces. As a result, the seal faces will be damaged and leakage will occur. (Ref. 5.24)

<u>Seal Upgrading</u>

In a telephone conference call on November 9, 2004 Flowserve's mechanical seal expert advised that an upgrade seal design has silicon carbide (SiC) stationary and SiC rotating seal faces. S&L experience with large high speed pumps indicates this combination of seal face materials is considerably more durable than the combination of carbon stationary and SiC rotating seal face materials that are presently installed in the RR pumps. The upgraded materials should extend seal life on both the "A" and "B" pumps when installed.

The seal upgrade also includes a "wavy face" on the rotating seal ring, that is designed to act as a pumping ring to push water and suspended solids away from the mating seal faces. The intent is to reduce the potential for solids to migrate to the sealing surfaces where they could mar the surfaces and cause a leak path.

A third upgrade to the seal is on the balance sleeve which extends under the stationary seal ring. To reduce the potential for binding, as noted above, the O.D. of the balance sleeve under the stationary ring will be knurled to reduce friction.

I. Recommended Actions

<u>RF12 Actions</u>

Alignment of the coupling and checking alignment when the pump is recoupled during RF12 is recommended.

In addition the coupling should be checked for concentricity and squareness, and balanced. Alternately, a new duplicate coupling may be available on short notice, from another plant, for replacement during RF12. If a new coupling is purchased, it should also be checked for squareness and balance as noted above, before installation. Flowserve agreed to advise Hope Creek which other stations have couplings that can be used on the "B" RRP.

Post RF12 Actions

The Hope Creek RR pumps most likely have some degree of shaft cracks and therefore should be monitored closely. A rapid rise in vibration amplitude would be sufficient reason to shut the pump down immediately for an internal inspection and rotor replacement, as the window between the rise and potential shaft failure is expected to be small. (Ref. 5.24)

The “A” pump should be monitored with the “B” pump, for capacity and vibrations.

Considering the age and time in service of the RR pumps, and the potential for shaft failure, (Ref. 5.21) the Station should be ready to rebuild the RR pumps. In order to minimize down time, a plan should be in place for access and rigging.

It is recommended that the new spare updated parts on hand be checked for rotor balance and shaft straightness before installing in the pump casings. The parts have been in storage for some time and the impellers are welded to the shafts. (Ref. 5.24) Potential residual stresses may have caused the shafts to become distorted over time. New couplings are included in the upgrade packages, and should be checked for concentricity, squareness, and balance.

The access and rigging plan and inspection of the replacement parts should be done during or as soon after RF12 as possible.

The upgrade parts on hand do not include seal cartridges. The intent is to rebuild existing seals at the Stations, using upgraded parts furnished by Flowserve. Instead, new generation seals with SiC stationary and rotating seal rings should be purchased. This should be done soon after RF12, in anticipation of an unscheduled outage.

RF13 and RF14 Actions

The vibrations for both “A” and “B” pumps have been stable over the last operating cycle. There is no indication that the current vibration levels are impacting the pump performance other than short seal life in “B” RRP. However, both “A” and “B” RRPs have operated over 130,000 hours and are approaching a perceived end of useful life. Reliability can be expected to decline with time in service if not upgraded. Thus, it is recommended that the “B” RRP be upgraded during RF13 and “A” be upgraded during RF14, unless monitoring shows capacity or vibration degradation earlier. “B” RRP upgrade is recommended earlier than “A” because of the higher vibration levels.

J. Reference Documents

5.1 RRP Technical Manual

5.2 Root Cause Analysis – Hope Creek “B”
Recirculation Pump Excessive Seal Leakage – Condition Report No. 70029861
Event Date: March 2003

5.3 N-7500 Seal Drawing No. L001491, Rev. B

5.4 Seal Parts List MC-1096, Rev. C

5.5 Pump Sectional Drawing 1E-3429-4, Rev. C

5.6 Pump Parts List Manual No. 8020

5.7 “B” RRP Stuffing Box Measurements Summary

5.8 PSEG Research Corp. Report No. 71125 of June 3, 1987

5.9 1B-P-201 Recirculation Pump vibration Data

5.10 1A-P-201 Recirculation Pump Data

5.11 Recirculation Seal History

5.12 RR Pump B Vibration Trends

5.13 “B” Recirculation Pump Seal Root Cause Info 70079861 dated 12/14/99

5.14 “B RRP Vibration Troubleshooting Summary

5.15. Summary of Hope Creek Vibration Issues and Planned Actions

5.16. Interview Notes between Brian Voll (S&L) and Joe Harragan (PSEG) on 11/3/04

5.17 Memo from R. Kaminski to J. Stavely sent 10/20/99

5.18 P&ID of Reactor Recirculation System M-43-1 Rev. 18

5.19 Flowserve Tech Note No. 9309-08-D22

5.20 GE SIL No. 459S1

5.21 GE SIL No. 459S2

5.22 Shaft and Cover Crack Issues in Nuclear Main Coolant Pumps by S. Gopalakrishnan (Flowserve)) dated September 8, 2004.

5.23 Reactor Recirculation History dated 10/21/04

5.24 Other Sources – Discussions and Meetings

Exhibit 5-1.
RR Pump A Vibration Trends

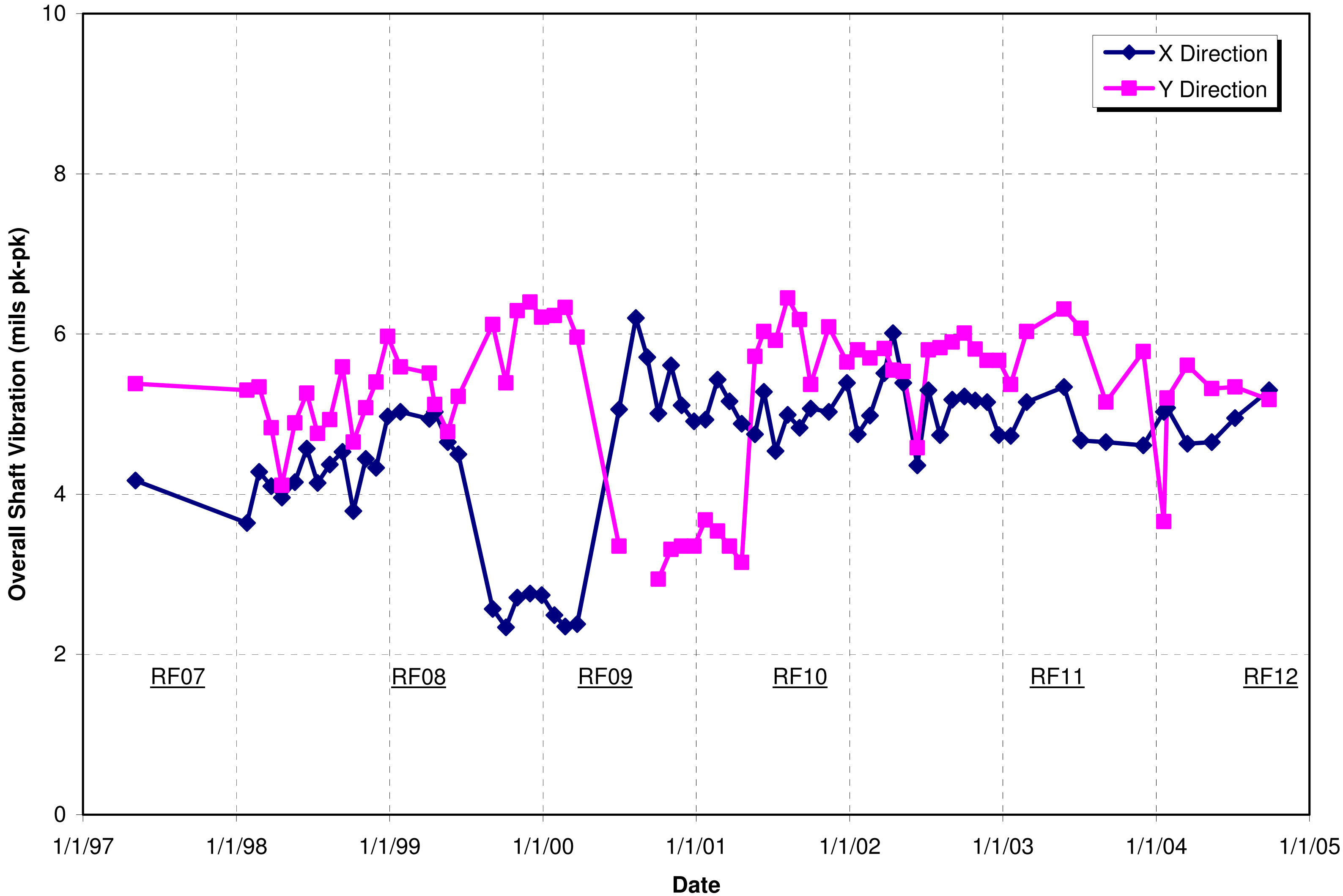


Exhibit 5-2.
RR Pump B Vibration Trends

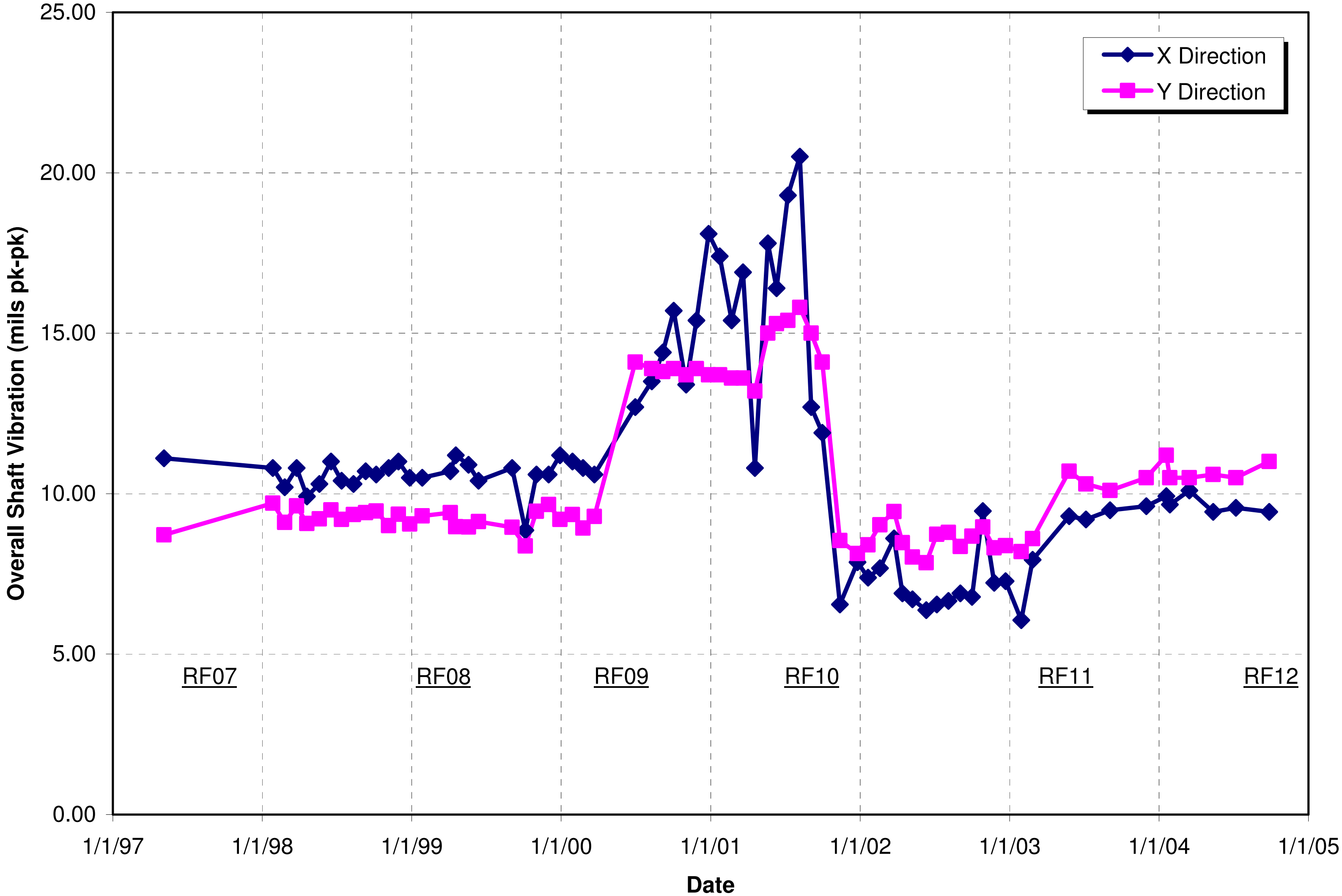
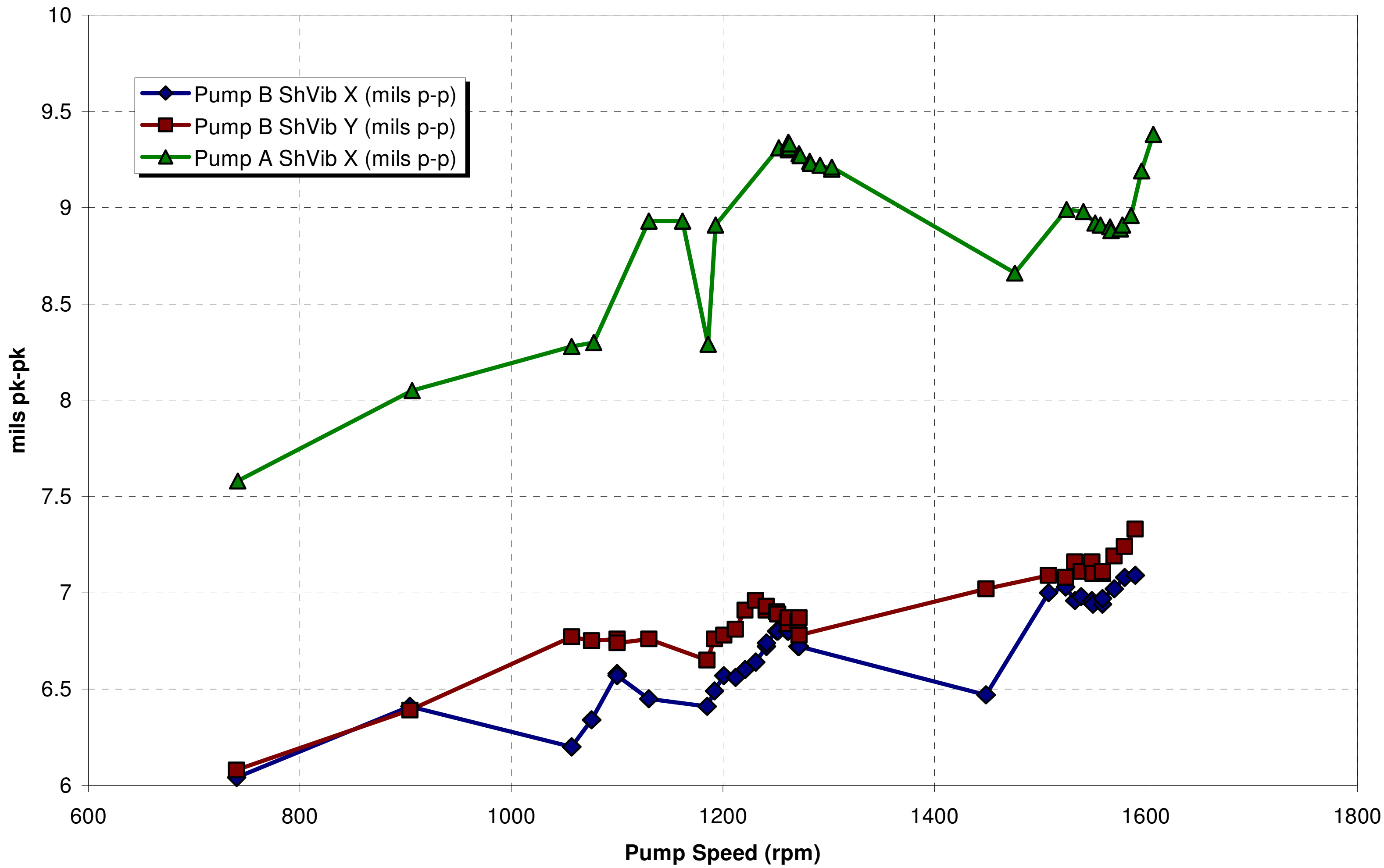


Exhibit 5-3
Browns Ferry Unit 2 RR Pump Shaft Vibration vs. Pump Speed



6. INTEGRATED SYSTEM ASSESSMENT

This section summarizes reviews performed of completed and planned RR and RHR system vibration evaluations. The vibration monitoring scopes, acceptance criteria and evaluation of monitoring results are reviewed.

A. Assessments prior to March 2004

Vibration monitoring of the RR piping was performed as part of initial startup in 1986. This monitoring addressed pipe stresses only and showed that vibrations were low and within acceptable limits. This program was not reviewed as part of this effort and the results are taken at face value.

Vibration monitoring of the RR piping was performed again in 1991 as part of an investigation of small bore piping failures. This monitoring showed that stresses in the modified small bore lines were within acceptable limits. This effort addressed only the small bore branch lines. Again, this program was not reviewed in detail as part of this effort and the results are taken at face value.

Additional vibration monitoring of the RR piping was recommended in 1995 as part of a large bore pipe cracking evaluation (Ref. 6.6). The purpose of this monitoring was to measure piping responses at critical pump speeds (102% core flow). This monitoring was never performed.

It is evident that there were opportunities prior to March 2004 to address the overall RR system vibration issues, which could have allowed them to be resolved earlier in the plant life.

B. March 2004 Monitoring Assessment

Vibration monitoring of the RR and RHR piping was performed following the March 2004 forced outage (Refs. 6.2, 6.3). Based on the discussions presented in Refs. 6.1 and 6.2, the objectives of the monitoring were to verify that pipe stresses were below acceptable limits and to obtain information to better understand the vibrations causing the valve component damage.

The scope, acceptance criteria and results for the RR and RHR vibration monitoring performed in 2004 were reviewed. The review results are summarized as follows:

<u>Monitoring Scope and Measurement Locations</u>

A total of 13 accelerometers were installed on the piping. This total was based on the available number of accelerometer cables, according to Ref. 6.2.

Some of the locations were near locations previously monitored in 1986 and 1991. This was done for comparison purposes to determine whether or not vibrations have changed over time. Some of the locations were near the RHR valves that have experienced component failures. This was done to aid in the analysis of the damage. One location was selected because it had a high modal response near the vane pass frequency (from a previous GE analysis).

Generally, a more extensive effort would be required to satisfy the March 2004 vibration monitoring objectives. The reviewed documents do not provide the complete thought process of PSEG at the time the vibration monitoring scope was established.

Even considering the limited number of accelerometer signals available, it is arguable that the effectiveness of the monitoring program could have been increased by placing some of the accelerometers directly on the valve operators where the component failures occurred. This would have provided a much better understanding of the behavior of the components and better data for benchmarking analytical models of the components than the pipe vibration data that was obtained. In addition, this would likely have initiated an effort to develop acceptance criteria for the valve components, which would also result in a better understanding of the characteristics and behavior of the components. The trade-off would be that fewer accelerometers might be available for measuring and evaluating pipe vibrations. Again, any thought process regarding this issue was not provided in the reviewed documents.

Acceptance Criteria

The acceptance criteria for the RR and RHR piping vibration monitoring performed in 2004 was developed by GE (Ref. 6.4). The following observations are provided based on a review of Ref. 6.4:

The vibration limits were established based on the stress limits described in ASME OM-S/G-1990, Part 3, “Requirements for Pre-operational and Initial Startup Vibration Testing of Nuclear Power Plant Piping Systems” (OM-3). This is the appropriate criteria for piping steady-state vibrations. However, the allowable stress for stainless steel is 13,600 psi, not 10,880 psi as stated in Ref. 6.4. A factor of 0.8 was apparently applied to the alternating stress allowable that is only applicable for carbon steel. The resulting allowable stress is conservative.

The acceptance criteria were established by performing response spectrum analyses for frequencies up to 200 Hz. The frequency range is acceptable, however, response spectrum analyses are applicable when the piping is being shaken by the building structure. For flow-induced vibration, on the other hand, the piping is being shaken by unbalanced forces acting on the piping legs. The axial forcing functions from flow-induced vibration result in a different relationship between maximum pipe stresses and displacements than forcing functions applied externally from the building structure. Therefore, analyses that simulate the axial forcing functions are more applicable for developing acceptance criteria for steady-state flow-induced vibration.

The acceptance criteria are in terms of displacement. This is generally the preferable parameter to measure for piping vibrations since displacement is directly proportional to bending stress. Also, displacement measurements emphasize the lower frequency responses (compared to velocity or acceleration) in the range where significant piping vibrations typically occur (approx. 30 Hz and below). Likewise, the largest displacements and stresses from the response spectrum analysis results will likely be from lower structural modes.

When the acceptance criteria for the RR and RHR piping vibrations were developed consideration should have been given to the probability that the vibrations could have a significant component at the vane passing frequency, which is approximately 125 Hz depending on the pump speed. If the 125 Hz component happened to be the dominant component in terms of vibration response, then the potential would exist for the acceptance limits to be unconservative. This is because displacements at high frequencies are more severe than equivalent displacements at low frequencies. For example, the displacement limit provided in Ref. 4 for one of the measurement locations is 280 mils pk-pk. If the vibration were occurring at a frequency of 5 Hz, the corresponding acceleration would be 0.36 g's pk. If the vibration were occurring at 25 Hz (1X), the corresponding acceleration would be 8.95 g's pk. Finally, if the vibration were occurring at 125 Hz (5X), the corresponding acceleration would be 224 g's pk.

It is evident from the above comparison that the displacement allowable is weighted toward lower frequencies, otherwise the allowable would be lower. One method to address the potential for higher frequency predominant vibrations is to also establish an acceleration limit, which would be more sensitive to higher frequency vibrations.

It is noted in Ref. 6.4 that the response spectrum is adjusted higher in the range of the 1X pump speed component. If the 5X component is also expected to be significant, the response spectrum should also be adjusted higher in that range.

Ref. 6.4 provides velocity criteria in addition to the displacement criteria. These limits are calculated from the analytical displacements and accelerations. This seems to presume a predominant vibration frequency. The basis and purpose (e.g., why does it make sense to use the velocity limit when the displacement limit is exceeded) for the velocity criteria is not provided.

Vibration Monitoring Results

The results of the RR and RHR piping vibration monitoring performed in 2004 are provided in Ref. 6.3 and summarized in Ref. 6.2. The following observations are provided based on reviews of those documents:

The measured pipe vibrations (with electrical noise effects accounted for) are less than 10 mils pk-pk and are significantly less than the allowable displacement limits. In addition, the measured vibrations are similar to measurements obtained in 1986 and 1991. Also, the maximum acceleration was approximately 1.2 g's pk, which is not excessive considering the predominant frequency is near 125 Hz. These results demonstrate that pipe vibrations have not changed significantly over time and that pipe stresses are acceptable.

The predominant vibration frequencies were at multiples of the RR pump running speed. Generally, the highest acceleration amplitudes corresponded to the 5X (impeller vane passing frequency) component. This was especially true for the RHR lines near the valves where the component failures occurred. These results suggest that the primary excitation source for the RHR valve component vibrations occurs at the vane passing frequency and that the valve components experiencing failures have natural frequencies near the vane passing frequency.

Vibration data was not collected with RHR operating in the shutdown cooling mode. The reason for this is not specified in the reviewed documents. Although there is no evidence of excessive vibrations during this mode, the most significant low frequency vibrations in the vicinity of the RHR valves will most likely occur during shutdown cooling since there is flow through the RHR piping.

One potentially significant component of the vibration could occur at the 10X harmonic, which is the second harmonic of the vane passing frequency. This component was not evaluated. Note that the 160 Hz upper limit of the band pass filter used in the data reduction would significantly attenuate this component.

The use of accelerometers for measuring the vibrations is acceptable since the expected predominant frequency components are relatively high for piping. For typical low frequency vibrations, measuring displacements directly would be preferred. This is because double-integration of acceleration signals to obtain displacement results in significant “integration noise” at low frequencies and makes interpretation of the data difficult. This problem with double integration was identified during the RR and RHR vibration monitoring effort and required a significant trouble-shooting effort. This is an additional reason why acceleration acceptance criteria, at least as a supplement to the displacement acceptance criteria, is recommended when the predominant frequencies are expected to be relatively high.

C. Planned Vibration Monitoring for EPU (Post RF12)

The RR and RHR vibration monitoring scope planned for after RF12 and for EPU was reviewed. The purpose of this review is to assess the effectiveness of the EPU vibration monitoring scope in addressing the current vibration issues. The review results, including recommendations for additional monitoring, acceptance criteria development and monitored operating modes, are provided as follows:

<u>EPU RR and RHR Vibration Monitoring Scope</u>

The RR and RHR EPU vibration monitoring locations are provided in Ref. 6.5. Three monitoring locations, with three accelerometers per location, are specified for each loop, for a total of 18 accelerometers. The locations were selected based on modal analyses previously performed by GE. No other bases for selecting the monitoring locations, other than installation concerns, were given. Specifically, neither the valve component damage that has occurred nor concerns regarding operation of the RR pumps at speeds above 1510 rpm were discussed. Considering the significance of these issues, the proposed monitoring scope is insufficient. The vibration monitoring scope should be increased as discussed below.

Piping Vibration Monitoring

One of the recommendations provided in Reference 6.2 is to obtain vibration measurements at RR pump speeds above 1500 rpm to determine whether or not the trend of increasing vibrations from 1440 rpm to 1500 rpm continues. In addition, operation of the RR pumps above the current administrative speed limit of 1510 rpm may be necessary as a result of EPU. Vibrations at these higher pump speeds could increase significantly, as evidenced by the reported "freight train effect" that occurs at pump speeds above 1510 rpm. Therefore, more comprehensive monitoring of the RR and RHR piping is warranted for EPU. Monitoring locations should be selected to verify that pipe vibration stresses are acceptable on all of the following piping for both RR loops:

- RR pump suction piping
- RR pump discharge piping
- Jet pump riser ring headers
- Jet pump risers
- RHR shutdown cooling return lines
- RHR shutdown cooling supply line
- RWCU branch lines
- Small bore branch lines

The intention is not to monitor each identified segment of piping. The intention is to ensure that pipe stresses are acceptable in each identified segment. The number of required vibration measurement locations can be kept to a reasonable level if the analyses performed to determine the monitoring locations and acceptance limits are based on the appropriate type of loading that provides representative overall system response characteristics.

Component Vibration Monitoring

The vibration characteristics of the valve components that failed and will likely have been modified prior to monitoring cannot be fully understood without obtaining vibration data. Also, the effectiveness of the modifications performed cannot be evaluated without obtaining vibration data. Therefore, the susceptible valve component should be included in the monitoring program. Again, the most effective number of sensors required can best be determined if analytical models are developed that provide accurate vibration response characteristics.

Acoustic Model Benchmarking

Another recommendation in Ref. 6.2 is to determine the acoustic characteristics of the RR system, both to understand how acoustics are contributing to the failures that have occurred and how the system will respond acoustically during EPU conditions. This will require development of an acoustic model of the system. In order to verify the accuracy of the acoustic model, dynamic pressure data needs to be obtained during power ascension up to the maximum speeds at which the RR pumps will be operated. Dynamic pressure should be measured near the source of the pressure pulsations (e.g., the RR pumps) and at locations where maximum acoustic responses may occur (e.g., near closed valves).

The acoustic modes predicted by the acoustic model will be strongly dependent on the speed of sound used in the analysis. The speed of sound can be calculated for the fluid conditions and effects of pipe wall flexibility, however, there are sources of inaccuracy in the calculated value, such as entrained air in the fluid. Therefore, means for measuring the acoustic velocity should be investigated.

RR Pump Parameters

The signals from the RR pump vibration and speed sensors should be tied into the data acquisition system used for EPU vibration monitoring so they can be directly correlated to the system vibration and acoustic responses. This will provide an understanding of the interaction between the pump and system responses.

Acceptance Criteria

The acceptance criteria for RR and RHR EPU piping vibration monitoring have not been established as of this time. The observations regarding acceptance criteria in Section 6B should be considered in development of the EPU acceptance criteria. As an added note, the acceptance limits should be in terms of peak values (displacement or acceleration) to correlate with peak stresses. Measurements taken in terms of rms vibration cannot be reliably correlated to peak values due to the quasi-random nature of pipe vibrations. Acceptance criteria would also need to be developed for the monitored valve components.

Monitored Operating Modes

Vibration data should be collected at predetermined pump speeds or power levels during power ascension up to the maximum speeds at which the RR pumps will be operated. Data should also be collected during the RHR shutdown cooling mode of operation. Data should also be collected at planned downpower evolutions to determine the effects of potential transient loading on RR and RHR system components.

D. References

6.1 Engineering Evaluation H-1-BB-CEE-1862, “Hope Creek Recirc/RHR Pipe Vibration Common Cause Analysis,” Revision 0.

6.2 Engineering Evaluation H-1-BB-CEE-1830, “Evaluation of Hope Creek In-Drywell Pipe Vibration,” Revision 02.

6.3 Structural Integrity Calculation HC-06-301, “Hope Creek Recirculation System Vibration Data Reduction,” Revision 0 (VTD 326747).

6.4 GE Report GENE-0000-0027-4832-01, “PSEG Nuclear LLC Hope Creek Generating Station Recirculation & RHR Piping Start-up Test Criteria,” Revision 1 (VTD 326534).

6.5 Structural Integrity Calculation HC-04Q-301, “Recirculation Piping Vibration Monitoring Locations,” Revision 0 (VTD 326528).

6.6 Engineering Evaluation H-1-BB-MEE-1050, “Hope Creek Recirculation System Large Bore Pipe Cracking Resolution,” Revision 0.

7. FAILED COMPONENT ASSESSMENT

This assessment includes a review of past and current component failures, system modification documentation and considered several attributes including those related to the current status of Design Engineering activities, Design Inputs, and Design Outputs.

A. Assessment of Actions Related to Component Failures

Prior to the March 2004 identification of failed and degraded hardware in the RR/RHR system inside drywell, the identified individual cases were either addressed by Design Engineering on a case by case basis or addressed by the station as a repair and replace activity.

As a result of the findings identified during the March 2004 outage, a collective review and index of hardware failures were initiated by NUCR 70037702 through the developed Common Cause Report, H-1-BB-CEE-1862. As a result of that set of investigations and the recommendations provided therein, additional Operations (Tasks) under NURC 70037702 were assigned to each activity. In addition, as new information becomes available, additional tasks are required to be added to the overall scope of NURC 70037702.

The effective implementation of the process, implemented in the spring of 2004, will provide for a more comprehensive assessment of repetitive component failure than the case by case engineering assessments and repair/replacement actions taken prior to that time.

DCP 800722673 is currently being prepared to address selected recommended corrective actions tracked by NUCR 70037702. Not all design and analysis elements have progressed to a point to justify a review at this time. Therefore, a final conclusion as to the real effectiveness of the final actions and associated modifications can not yet be made. Selected portions of the available supporting analytical documentation have been reviewed to evaluate the consistency of method with the intended results.

A comprehensive review and a sound decision making process by Design Engineering is a critical element as to the effectiveness of the final form of the corrective actions.

Table 7-1 reflects a summary and status (as of 11/09/04) of the specific actions planned and tracked by NUCR 70037702 and in progress to address the Recommended Actions contained in Table 8-1 of the Common Cause Analysis for the RR/RHR system component failures.

Table 7-1		
Brief Description of Common Cause Report Recommended Actions	**NUCR 70037702 Operation No.**	**Status Summary and Commentary**
Disassemble the F050A actuator.	0120	Status: Open Due Date: 11/15/04 Comment: Based on Notification 20208119 (10/2004) identifies initial phases of potential degradation and additional observations on the current general condition also need to be resolved. See further discussion Item 7.
Modify the F050A valve actuator.	0120	Status: Open Due Date: 11/15/04 Comment: The final decision to modify the actuator via DCP will be based on the as found condition. See further discussion Item 7.
Inspect the F050A actuator in subsequent outages.	0130	Status: Open Due Date: 01/11/05 Comment: Action is to create recurring inspection task and is judged to be an appropriate monitoring plan given the prior history of degradation on this valve.
Inspect additional valves similar in design to F050A.	0132	Status: Closed Actual Date: 01/09/04 Comment: Inspection summary states no visual damage was detected. Concern: Based on the above statement it is not clear if the inspector verified that the air piston cylinder had not loosened as was intended. See further discussion Item 9.
Remove hand wheel from the F060A and F060B.	0140	Status: Closed Actual Date: 10/30/04 Comment: Operations has agreed that it is acceptable to remove hand wheels during RF12. Operation 0320 created to verify action. See further discussion Item 8.
Inspect hand wheels to determine if lock wire was effective.	NA	Comment: Based on recent observations documented in Notifications 20208116 and 20208117 (10/2004) hand wheels on F060A and F060B were again found loose. Therefore, prior method of using a lock wire can be concluded to be ineffective.
Evaluate F060A/B top works frequencies.	0150	Status: Open Due Date: 01/12/05 Comment: Valve manufacturer is providing calculations based on finite element models. Evaluation has been expanded to include the F077 valve. Design Engineering is currently reviewing analytical documentation. Concern: Results of calculation do not provide a correlation to actual damage. See further discussion Item 1.
Modify F060 A/B top works to avoid pulsation frequencies.	0150	Status: Open Due Date: 01/12/05 Comment: Valve manufacturer is providing calculations based on finite element models. Modification effort has been expanded to include the F077 valve. Design Engineering is currently reviewing analytical documentation. Modification to be provided

Table 7-1		
Brief Description of Common Cause Report Recommended Actions	**NUCR 70037702 Operation No.**	**Status Summary and Commentary**
		in DCP 80072763. Concern: Analytical results for currently proposed modification do not significantly alter the frequencies. See further discussion Items 2 and 3.
Modify F060A/B stem protector to increase slot clearance.	0150	Status: Open Due Date: 01/12/05 Comment: Current analysis based on proposed modification recommends increasing slot clearance. Modification to be provided in DCP 80072763.
Determine if acceptable to remove limit switches.	NA	The removal of the limit switches is not being given further consideration.
Inspect F060A/B valve externals for distress.	NA	Comment: A plant walk down conducted 10/21/04, supported by photographs, documents additional failures of limit switch hardware.
Inspect F050A actuator.	120	Status: Open Due Date: 11/15/04 Comment: Based on Notification 20208119 (10/2004) the additional recent observations on the current general condition of F050A hardware also needs to be resolved. (A plant walk down conducted 10/21/04, supported by photographs, documents additional hardware degradation.) See further discussion Item 7.
Inspect F060A hand wheel.	NA	Comment: Based on recent observations documented in Notifications 20208116 and 20208117 (10/2004) hand wheels on F060A and F060B were again found loose.
Inspect F060A/B operator internals.	0170	Status: Open Due Date: 11/15/04 Comment: Task also includes F050A operator. Preliminary inspections of F050A are documented in Notification 20208119 and note three conditions of degradation. Preliminary inspections of F060A are documented in Notification 20208920 and notes conditions of degradation. See further discussion Item 4. Concern: Indicative of further hardware degradation since corrective actions taken in spring 2004 outage.
Inspect F050A/F060A valve internals for signs of degradation, loose fitting parts.	0180	Status: Open Due Date: 11/15/04 Comment: Planned initial inspection to be based on radiography to determine evidence of loose parts. Concern: Based on valve size, wall thickness radiograph may be of limited value especially to determine if loose fitting parts exist. See further discussion Item 5.
Trend noise in north pipe chase routinely.	0190	Status: Open Due Date: 01/29/05 Comment: Current plan states that monitoring will conducted during power ascension. Includes recording of plant parameters at notification of noise. Concern: The plan should include capability to gather data as

Table 7-1		
Brief Description of Common Cause Report Recommended Actions	**NUCR 70037702 Operation No.**	**Status Summary and Commentary**
		required during power operation. See further discussion Item 6.
Provide remote visual observation.	NA	This action is not being pursued further.
Determine modal characteristics of F077 valve top works.	NA	This evaluation is being conducted in a similar manner as activities supporting Operation 0150. Status: Open Due Date: 01/12/05 Comment: Valve manufacturer is providing calculations based on finite element models. Design Engineering is currently reviewing documentation. Concern: Results of calculation do not provide a correlation to actual damage. See further discussion Item 1.
Inspect specific additional valves for external hardware degradation.	0200	Status: Closed Actual Date: 11/09/04 Comment: No visual damage was found.
Verify instrumentation lines adequately supported.	0210	Status: Open Due Date: 11/12/04 Comment: Design Engineering to conduct. Activity not yet started.
Evaluate cause for past failures of valve F077	NA	This evaluation is being conducted in a similar manner as activities supporting Operation 0150. Status: Open Due Date: 01/12/05 Comment: Valve manufacturer is providing calculations based on finite element models. Design Engineering is currently reviewing documentation. Concern: Results of calculation do not provide a correlation to actual damage. See further discussion Item 1.

B. Discussion Items Noted in Table 7-1

1. The analytical finite element model results generated for the current configurations of the F060A/B and F077 valve operator assemblies have not been processed and assessed in a way which provides a correlation to the damage observed. This correlation should address a comparison of the observed damage, (such as gear box cover plate deformations, cover plate cap screw failure, damage to the stem extender/stem interface and internal yoke nut failure) to the analytically predicted results. If a reasonable correlation can not be demonstrated, the proposed modification is not likely to achieve the intended goal.

   Recommendation: Provide a separate failure mode assessment using the FEM results.

2. The calculations establish that the first and second mode frequencies of the existing assembly are in the range of 94-98 HZ (F077) and 60-63 (F060A/B). Prior test data has shown that RHR branch piping has notable

accelerations primarily at the 5X condition of 125 Hz. Hence the damage associated with high amplitude vibrations is likely to be associated with the modal frequencies of specific components such as the gear box cover plate. Because the primary operating pump speeds expected to be used for current operation and future EPU operation range from 1300 to 1600 RPM, the criteria for the modification should based on 150 HZ or greater.

Recommendation: The criteria for the modification of the valve operator assemblies should proactively consider EPU pump speeds and associated 5X frequencies with margin and be based on 150 HZ or greater.

3. The stated action to be taken for Operation 0150 is to include sufficient post mod testing to ensure goals are met and Operation 0155 implements a re-inspection of limit switches during R13. At this time a determination of what is necessary to provide a “sufficient test” has not been made. In addition, during R13 it is planned to gather EPU vibration data at pump speeds above 1500 RPM.

   Recommendation: The proposed post mod testing of the manual gate valve top works should include collection of test data and be implemented to ensure goals are met. This includes the collection of data during cycle 13 at pump speeds above 1500 RPM.

4. Repetitive failures of yoke nuts have been established. Degradation of valve F060A is further discussed in e-mail, W. J. Nealon to T. A. Babin et al., dated 10/29/04, H1BC-1-BC-V183, Yoke Sleeve Nut. This write up identifies that a similar yoke nut failure was encounter on the F077 valve in the April 2003 outage.

   Recommendation: The failure of these components should be addressed in the recommended failure mode assessment.

5. The proposed radiograph would only provide an indication of gross damage and general condition and would not be expected to yield indications of loose connections. The existence of loose connections caused by high cycle wear would be best accomplished by an opened valve inspection.

   Recommendation: An open valve inspection valves should be done to conclusively determine the current condition of the F050A/F060A valve internals. If indications are noted for the F060A valve, as a minimum, a similar inspection of valve F060B should be conducted.

6. The current plan states that noise monitoring will be conducted during power ascension. Component degradation does not generally start for weeks to months into the operating cycle.

   Recommendation: The monitoring system should be available and/ or the program implemented, as needed, during the full operating cycle.

7. Based on discussions with PSEG Civil/Structural personnel, prior to Notification 20182397 (03/2004), there were no previously identified issues related to the integrity of the actuator tube assembly up until its failure being reported in 03/2004. We believe, it can not be easily determined over what relative time period the component degraded. It would seem, however, that any significant prior degradation of the cylinder would have limited its ability to provide full force during previous valve tests. Therefore, a measure of the time frame in which significant damage occurred would seem to be sometime since the last time the actuator was pressurized to test the check valve. If this were the case, it would suggest a more recent change in the loading that lead to final failure of the threads and set screws. Based on discussions with PSEG Civil/Structural personnel, the spring 2004 outage modification to repair the failed cylinder included replacement of like for like parts provided by the valve manufacturer. Therefore the current installation is no more robust that the hardware that was originally installed. Initial walk down inspections conducted during RF12 are documented in Notification 20208119 (10/21/2004) and note indications that the actuator cylinder exhibits play. This condition is noted after having been installed for less than eight months. Therefore, there is some reason to believe that these components will continue to fail if simply replaced.

   Recommendation: Implement the proposed option identified in Operation 120 to modify the valve operator.

8. Removal of the F060A/B and F077 hand wheels, as was suggested, has been determined to be acceptable as documented in the closure of NUCR 70037702. Separately, the valve vendor has recommended that hand wheels be removed in calculations RAL-7482 and RAL-7483 and those analytical evaluations of the valve operator assemblies exclude the mass of the component.

9. The formal inspection activity task descriptions should be clearly stated as to the specific attributes to be investigated and reported. General instructions such as “visual inspection” may not be sufficient to address the intent of the inspection. Both Design Engineering and the responsible discipline engineer should contribute to the planned inspection instructions.

C. Cross reference to the primary past and current documents reviewed for specific RR/RHR system component failure histories.

   Detachment of the F050A (V111) Actuator

   Reference:
   H-1-BB-CEE-1862, Hope Creek Recrirc/RHR Pipe Vibration Common Cause Analysis, Rev. 0, 07/27/2004
   Notification 20182397 (03/2004)
   Notification 20208119 (10/2004)

Detachment of the F060B (V074) Valve Hand Wheel

Reference:
H-1-BB-CEE-1862, Hope Creek Recrirc/RHR Pipe Vibration Common Cause Analysis, Rev. 0, 07/27/2004
Notification 20182400 (03/2004)

Limit Switch Failures of the F060A (V183) Valve

Reference:
H-1-BB-CEE-1862, Hope Creek Recrirc/RHR Pipe Vibration Common Cause Analysis, Rev. 0, 07/27/2004
Notification 20182396 (03/2004)
Notification 20208116 (10/2004)

Limit Switch Failures of the F060B (V074) Valve

Reference:
H-1-BB-CEE-1862, Hope Creek Recrirc/RHR Pipe Vibration Common Cause Analysis, Rev. 0, 07/27/2004
Notification 20182395 (03/2004)
Notification 20208117 (10/2004)

Gear Box Cover Plate Deformation for the F077 (V078) Valve

Reference:
Notification 20208118 (10/2004)

Broken Yoke Sleeve Nut on the F060A (V183) Valve

Reference:
Notification 20208920 (10/2004)
E-Mail, W. J. Nealon to T. A. Babin et al., dated 10/29/04, H1BC-1-BC-V183 Yoke Sleeve Nut

Broken Yoke Sleeve Nut on the F077 (V078) Valve (Previously identified in April 2003)

Reference:
Notification 20141040
Notification 20141176
WO 60036361
NUCR 700031101

D. Documents Reviewed:

The portions of the following documents which relate directly to the failure of components were reviewed as part of this assessment.

Documentation Prepared to Address Specific Concerns Related to the Issue

NUCR 70037702
H-1-BB-CEE-1862, Hope Creek Recrirc/RHR Pipe Vibration Common Cause Analysis, Rev. 0, 07/27/2004
H-1-BB-CEE-1830, Evaluation of Hope Creek In-Drywell Pipe Vibration, Rev. 2, 04/05/2004
HC-06-301, Hope Creek Recirculation System Data Reduction, Rev. 1 (selected portions)
VTD Number 326534, Recirculation & RHR Piping Start-Up Test Criteria, GENE-0000-0027-4832-01, DRF-0000-027-4832, Revision 1, April 2004, Class III (selected portions)
VTD Number 326534, Evaluation and Vibration Testing of Recirc. & RHR Piping Instrumentation, Report No. 04055BP, Revision 0, May 12-14, 2004 (selected portions)
DCP 800722673, (modification package activities as of 11/10/04)
Notification 20208116, Limit Switch on BCF060A, 10/21/2004
Notification 20208117, Limit Switch on BCF060B, 10/21/2004
Notification 20208118, Gear Box Cover BCF077, 10/21/2004
Notification 20208119, Problems With BCF050A, 10/21/2004
Notification 20209339, Ineffective Corrective Action, 11/01/2004
Notification 20208920, Handwheel OP Spins Freely, 11/01/2004
E-Mail, W. J. Nealon to T. A. Babin et al., dated 10/29/04, H1BC-1-BC-V183 Yoke Sleeve Nut
FLOWSERVE Calculation RAL-7482, Rev. 0, Design Modification Report, (Client Comment Copy)
FLOWSERVE Calculation RAL-7483, Rev. 0, Design Modification Report, (Client Comment Copy)

Documentation Related to the Design in General

M-51-1, Sht.1, Rev. 32, Residual Heat Removal
M-51-1, Sht.2, Rev. 31, Residual Heat Removal
M-43-1 (Q) -27, Sht.1, Rev. 32, Reactor Recirculation System

DCP 80062466, EPU Piping Vibration Monitoring Installation Package

Dwg. No. 93-15122, Rev. E, 12”-900 Weld Ends, Carbon Steel Flex Wedge Gate Valve
Dwg. No. 93-14347, Rev. D, 20”-900 Flex Wedge Gate Valve
Dwg. No. 14053-01-H, 12”-900 Testable Check Valve

Dwg. No. 761E593, Recirc. Loop Piping Restraints
Dwg. No. 131C7810, Recirc. Pump Restraint
Dwg. No. 762E197, Recirculation Loop Suspension, Sht. 1, Rev. 11
Dwg. No. 762E197, Recirculation Loop Suspension, Sht. 2, Rev. 13
Dwg. No. 762E197, Recirculation Loop Suspension, Sht. 3, Rev. 10
Dwg. No. 762E197, Recirculation Loop Suspension, Sht. 4, Rev. 11
FSK-P-169, Sheet 1 of 1, Rev. 12, Recirculation Piping Loop-A, (Unit-1)

FSK-P-170, Sheet 1 of 1, Rev. 14, Recirculation Piping Loop “B”, (Unit-1)

General Requirements

HC.OP-SO.BB-0002(Q)-Rev. 50, Reactor Recirculation System Operation

E. Documents given a cursory review to determine their relationship to the RR/RHR system component failures.

Notification 20209339, Ineffective Corrective Action, 11/01/2004

This Notification 2029339 reiterates re-currant issues related to potential RR loop vibration, component failures and recommends a number of evaluations be conducted.

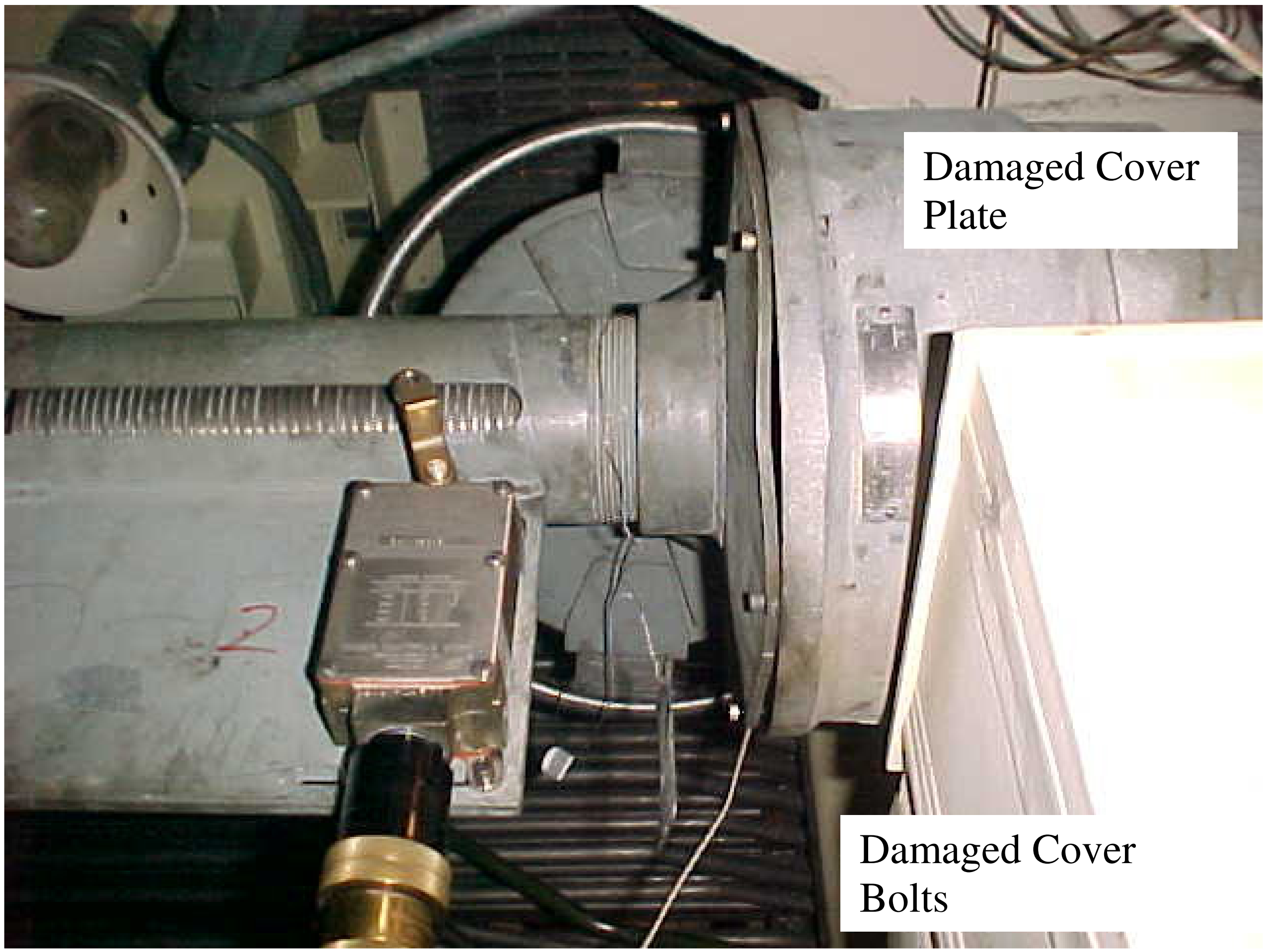
Due to the recent issue of this document, the disposition of this notification is currently in progress. Therefore, an assessment of those resolutions cannot be made at this time.

The following additional component related issues and the associated documents identified in the Common Cause Analysis (H-1-BB-CEE-1862) were not reviewed since they were identified as not being directly related to the RR/RHR system operation.

Notification 20182454 (03/2004)
Notification 20182505 (03/2004)
Notification 20182394 (03/2004)

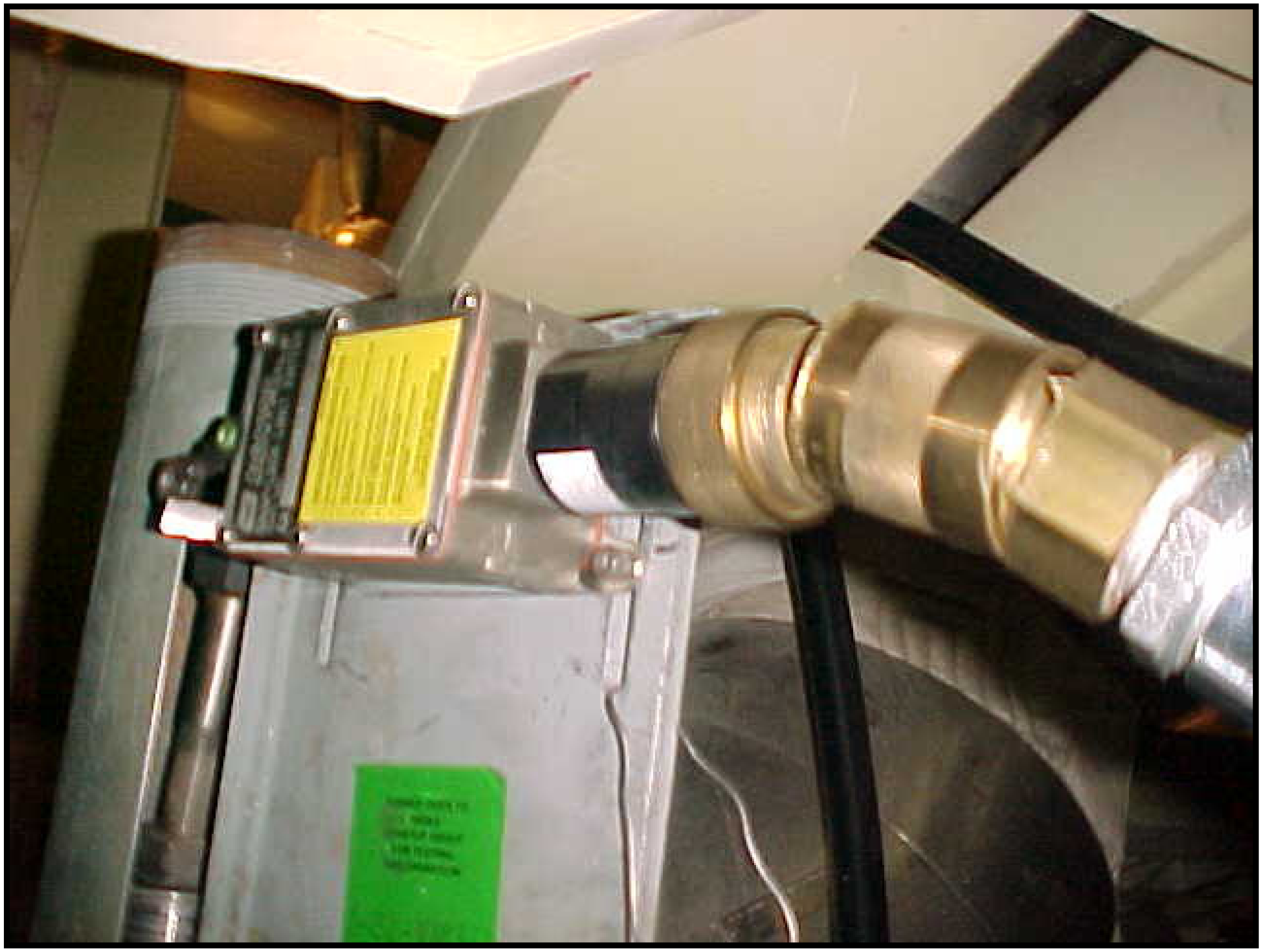
Section 7

Exhibit 7-2
F077 Valve



Section 7

Exhibit 7-1
F060A Valve



8. NDE ASESSMENT

A. Background

i. Early Failures of Small Bore Lines

Small bore line failures in the RR system during early plant operations (1987 - 89) were ultimately addressed by removal of all double isolation valves from the RR Loop A and B small bore valve drains and instrumentation taps. Other BWR plants had similar experiences and this is not unique to Hope Creek. This issued was addressed in stages with the initial action being the modification of the configuration of the line that failed, subsequent elimination of some of valves after the next failure, and removal of all the valves after the next line failure. A total of 24 valves were ultimately removed from valve drains and these lines were capped. 16 valves were removed from the elbow instrument taps and piping reconfigured. See Table 8-1. These actions were appropriate and the removal of the valves from these locations has eliminated the majority of failures in these areas. Prior to the most recent, single, failure, noted in December 2001, the previous identified failure was in December 1989. This most recent failure was attributed to the mass of vibration monitoring equipment attached to the line, which has since been removed.

ii. NDE Techniques

Due to the nature of the origination of these small bore fatigue failures at the root of the socket welds, surface NDE methods are not useful to locate these indications until a through-wall failure has occurred. Hope Creek has chosen to supplement surface examination with volumetric methods.

Ultrasonic (UT) examination is possible, and it was stated that a procedure was developed in the mid 1990’s to examine these small bore welds for subsurface indications such as vibration fatique failures. It was determined to be limited in effective usefulness to Schedule 80 wall thickness which precluded use on Hope Creek lines which are Schedule 160 in these locations. It was also stated that this decision was supported by demonstrations at another nuclear plant that uses the procedure for examination of Schedule 80 small bore welds. These demonstrations were not able locate Hope Creek calibration standard defects reliably.

In the absence of a reliable UT examination technique, radiography is being used at Hope Creek in limited cases where volumetric examinations is deemed necessary. This was recently performed on the "A" loop RR line suction elbow connection as a follow-up to the previous failure that occurred in October, 2001. This method is appropriate for such examinations though it does exhibit logistical challenges that UT would not present.

B. In Service Inspection (ISI) Program

The Hope Creek In Service Inspection program began with a traditional Section XI based ISI program to monitor piping system condition for service induced conditions. The current station commitment and program basis is the 1998 edition of Section XI with the 2000 addenda. This has been implemented during the entire 1st 10 year inspection interval and the 1st and 2nd periods of the 2nd 10 year interval.

Review of the program information for the RR system and associated portions of the RHR system against the Section XI requirements determined that the welds and components are completely and accurately listed in the ISI program database and the selected welds and components properly implement the Section XI requirements. The database was verified to reflect the completion of the required examinations.

Prior to the current outage a total of 35 % of the Loop B RR welds have been examined. At this time in the current (2nd) ten year interval approximately 22% of the pipe welds have been examined. Upon completion of this outage this total will be 28%, in compliance with the traditional Section XI requirements. After this outage, no “B” loop or “A” loop RR pipe welds are scheduled to be completed prior to the end of this inspection interval.

Pipe welds 1" and under are exempt from examination by Section XI and are not included in the basic examination population in the Hope Creek Program. However, Hope Creek has included some RR small bore lines for augmented examination based on the previous failures.

The ISI program also includes the Loop A and B RR pumps for inspection of the internal surfaces of the pumps. Section XI does not require disassembly specifically for such inspections and the information provided indicates that maintenance has not been performed which would provide internal access. Therefore these examinations have not been performed for either pump as of this date, which is within the provisions of Section XI.

C. Risk Based ISI Program

Hope Creek has updated the ISI program to a "Risk Informed" based program as provided for in Code Case 578-1 and EPRI TR-112657. These provisions address an alternate program for Class 1 and 2 pipe welds, based on evaluation of risk for systems and portions of systems. This evaluation is then used to select for examination, those welds with the highest risk. This evaluation was conducted for Hope Creek and the ISI program has been updated to reflect this revised population. These provisions are used in conjunction with the other ISI program requirements of ASME Section XI.

Review of the new risk based ISI program classification for the RR system piping, nozzles, and supports found that they correspond with the Code Case provisions and EPRI base document requirements.

It was noted that some welds included based on risk were also examined under the traditional program, which the risk of determination identified welds that were not included in the original population.

The impact of this program change is the application of the inspection resources to pipe welds with the highest risk, in lieu of the more arbitrary traditional selection process. Numerically the inspections are reduced, however the inspections performed are on the more risk significant pipe welds. The other aspects of the program including examination method remain unchanged

The new program during one 10 year interval for RR Loop B pipe welds will cover a total of 7 welds (2 safe end to pipe welds and 5 pipe welds) compared to the current population of 30 pipe welds required by Section XI (4 safe end to pipe welds, 12 pipe welds, 9 long seams and 1 lug weld, and 5 augmented pipe welds) and which were included in the traditional program.

For the RR Loop A pipe welds, the new program during one 10 year interval will cover a total of 7 welds (3 safe end to pipe welds and 4 pipe welds) compared to the current population of 34 pipe welds (3 safe end to pipe welds, 12 pipe welds, 16 long seams, and 3 augmented pipe welds) which were included in the traditional program for this same interval.

This outage is the 1st which begins implementation of the risk informed program provisions and the RR welds identified for examination in both loops are currently in process. As previously noted, upon completion of these examinations this outage, no additional RR welds in either the A or B loop are scheduled for examination during the remainder of this 10 year inspection interval, which ends in 2007. During the next (3rd) 10 year interval beginning in 2007, the population of RR pipe welds examined will be only those selected based on the risk identified criteria supplement any augmented examinations as identified by Hope Creek.

The ISI program change has been submitted to the NRC on a relief request and it is anticipated that final approval will be received this month.

D. ISI Inspections Scheduled During RFO12

During the current outage the ISI schedule includes examination of 4 RR Loop B and 4 Loop A pipe welds using ultrasonic techniques. In addition, the RPV1-N1BSE and RPV1-N1ASE outlet nozzles to safe end welds are also being examined ultrasonically.

As a volumetric method, ultrasonic examination is capable of identifying internal discontinuities, including fatigue induced failures, in the weld and adjacent base materials examined.

The previous examinations of the RR and RHR large bore welds with ultrasonic examinations have not identified any indications that could have been related to vibration or fatigue failures.

E. ISI Program Augmented Inspections

i. Small Bore RR Lines

As noted, Section XI excludes 1" and under pipe, however, due to previous failures in RR small bore lines, some have been added as augmented examinations and date back to the 1st refuel outage

The original plant configuration for small bore RR lines included 20 lines. Of these, 12 have been removed and capped and 8 have had the double isolation valves removed due to the various line failures.

Currently 4 RR small bore lines remain with an attachment to the RR elbow outer radius that has been shown to be subject to vibration induced fatigue failure in the past. The 4 RR lines with attachments to the inner elbow radius remain but have not failed in the past.

The four lines with the outer elbow attachment have been added to the ISI program for augmented inspection of the weld connecting to the elbow based on failure history. These are the connection to the RR suction elbows for Loop A and B for lines 1-BB-1CCA-225-1, 1-BB-1CCA-223-1, 1-BB-1CCA-218-1, and 1-BB-1CCA-220-1. The four inner elbow connections for lines 1-BB-1CCA-222-1, 1-BB-1CCA-224-1, 1-BB-1CCA-219-1, and 1-BB-1CCA-221-1, have not been added for augmented inspections under the ISI program.

The augmented inspections for these four welds require both surface examination and radiographic examination. The examination performed during R011 verified their acceptable condition. The examinations during R012 that was recently performed also determined each to be in acceptable condition. One weld radiograph for 1-P-BB-321-FW46 exhibited an area of question which was documented on Order 70041984. Enhancement of the radiographs at the EPRI NDE center was requested and performed. Evaluation of the enhanced radiographs for this weld determined that it appeared to be a volumetric weld anomaly from fabrication and was not an indication that may be the start of a fatigue induced weld failure. Based on these results, no further augmented examinations are currently scheduled within the ISI program for these four welds.

Prior to the determination of the weld condition after radiograph enhancement, extent of condition liquid penetrant examinations on the other lines with similar configuration were performed, as well as visual examination of the associated supports. The lines examined included the 8 connections to the suction elbows and the 24 capped connections to the suction and discharge valves. No indications were found. The results of the augmented ISI examinations and the additional examinations performed under Order 70041984 for the RR small bore lines, form the basis for the determination that future augmented inspections are not necessary. This is an appropriate action. However, it is recommended that as a minimum, the one weld that has shown an internal weld anomaly be subject to continued augmented examinations until system vibration issues are resolved.

Based on review of the operating experience information contained in H-1-BB-CEE-1830 actions taken by Hope Creek appears to be consistent with actions taken by other stations with small bore weld failures.

ii. NDE Outside of the ISI Program

Some NDE examinations have been performed due to the most recent hardware/equipment failures noted on the RHR system. No failures have been identified in these lines in the past and they are not currently included as augmented examinations under the ISI program. Performance of these additional examinations is a prudent action.

In response to the failed equipment noted in the drywell liquid penetrant examinations were conducted in March 2004 on the line located adjacent to the F050 valve and the line adjacent to F060 valve on the "A" and "B" RHR lines. The connection to the line and the connection to the first isolation valve (V586 and V589) were examined. No indications were noted.

Review of hardware/equipment failure history noted that the F077 valve is also located on an RHR line and this location has exhibited the largest number of equipment/hardware failures observed by plant personnel. Near this valve is a small bore line (1-DBA-154) with a double isolation valve configuration (V-302 and V-303) which is the configuration removed from the RR system after the earlier failures. It does not appear that the connections for this line have been examined, as were the similar lines recently examined after noted equipment damage.

It is recommended that these connections be examined and any others adjacent to hardware or equipment exhibiting vibration related damage or failure in both the RHR and RR systems noted in the future at each outage be examined at each outage until system vibration issues are resolved.

F. Large Bore Weld Indications

Review of previous ISI examination information and engineering document H-A-BB-MEE-1050 noted that two RR line welds were found with surface indications during the 3rd refuel outage during ISI examinations.

This report noted that two large bore (28") RR line welds had been identified with surface indications during ISI. This is the pipe to reducing tee connection on each loop (1-BB-18VCA-014-6 and 1-BB-28VCA-013-6). Per the evaluation it was determined that these indications were a result of a shop weld material issue and therefore were not related to vibration fatigue. Investigation at that time identified 50 welds that potentially containing this weld material. Of this sample, all 16 of the 28" shop welds, all 4 of the 22" welds, and 4 of the 30 12" welds were inspected in outages RF03, 4, and 5. No additional indications were noted.

Some of these welds have been subject to scheduled ISI examinations since that time and no indications have been found.

It was confirmed that this data was reflected in the ISI database for these welds examined, Section XI expanded sample requirements met and the affected welds re-examined during the next period as required.

Vibration analysis performed subsequent to the March 2004 forced outage verified that pipe stresses are below design allowables and additional NDE is not necessary.

G. GE SIL 459

Review of GE SIL 459, 459S1, 459S2 and 459S3 related to pump shaft cracking noted that SIL 459 recommends that the user review their ISI program and establish plans to schedule inspections to detect cracks in pumps with over 80,000 hours of operations. At this time, it is estimated that the pumps have between 130,000 and 140,000 hours of operations and would therefore be subject to the recommendation of the SIL.

Discussion with ISI personnel and review of program database information determined that augmented inspections were included in the Hope Creek ISI program for both the A and B pump shafts. It was stated that these examinations was reviewed and discussed and an ultrasonic demonstration was observed at Salem on a similar shaft. Due to the lack of access to the Hope Creek RR pump shafts without pump disassembly and the accuracy of available drawings for the shafts, such examination was judged to require shaft removal to be meaningful and accurate. This is consistent with other US BWR actions. As listed in the ISI program at this time, when the pumps are disassembled, these shaft inspections would be required if they are not replaced.

Since meaningful shaft inspections can not be performed without pump disassembly, performing the examinations is not practical since the next pump disassembly is planned to include shaft replacement.

H. Recommendations

It is recommended that the small bore connection noted by radiography with a weld anomaly be included in the ISI program for continued augmented radiographic examination at each outage until system vibration issues are resolved.

It is recommended that each small bore connection to the RHR system in the vicinity of the areas of the past pipe and equipment failures, be examined with surface and visual examination at each outage until system vibration issues are resolved.

I. References

i. H-1-BB-CEE-1862, Revision 0 7/27/2004: Hope Creek RR/RHR Pipe Vibration Common Cause Analysis

ii. H-1-BB-CEE-1830, Revision 2, 4/5/2004: Evaluation of Hope Creek In-Drywell Pipe Vibration
iii. H-1-BB-MEE-1050, Rev 0, dated 12/26/95, Hope Creek Recirculation System Large Base Pipe Cracking Resolution
iv. Condition Report No. 70029861, 4/1503: Root Cause Analysis of Hope Creek "B" Recirculation Pump Excessive Seal Leakage
v. PSEG Order Number 70041984, A Loop Recir 1" Instru Line Linear Ind.
vi. NRC Information Notice 95-16: Vibration Caused by Increased Recirculation Flow in a Boiling Water Reactor
vii. ASME Section XI-1998 with 2000 Addenda
viii. ASME Code Case N578-1: Risk Informed Requirements for Class 1,2, and 3 Piping Method B Section XI, Division 1
ix. EPRI TR-112657 Revision BA: Revised Risk Informed Inservice Inspection Evaluation Procedure
x. EPRI Fatigue Management Handbook
xi. GE Nuclear Services Information Letter: SIL No. 459 dated December 15, 1987
xii. GE Nuclear Services Information Letter: SIL No. 459S1 dated March 23, 1990
xiii. GE Nuclear Services Information Letter: SIL No. 459S2 dated October 21, 1991
xiv. GE Nuclear Services Information Letter: SIL No. 459S3 dated August 31, 1993
xv. ISI Weld Identification Figures A-26, A-29, A-32

Table 8-1

Historical Small Bore Line Failures: RR Loops A and B			
Location	ValveNumber	Failure/Date	Actions
RR Loop A Suction Valve Vents and Drains	V001	N/A	(9/87) Removed stem and gland vent lines (11/88) Removed seat drain lines and capped.
RR Loop A Discharge Valve Vents and Drains	V002	(2/87) Cracked seat drain line to valves V-017 and V-018 (9/87) Cracked gland vent line to valves V-034 and V-035	(2/87) Replaced with revised configuration (9/87) Removed stem and gland vent lines and capped. (11/88) Removed seat drain lines and capped.
RR Loop B Suction Valve Vents and Drains	V004	N/A	(9/87) Removed stem and gland vent lines and capped. (11/88) Removed seat drain lines and capped.
RR Loop B Discharge Valve Vents and Drains	V005	(11/88)	(9/87) Removed stem and gland vent lines and capped. (11/88)

Historical Small Bore Line Failures: RR Loops A and B			
Location	ValveNumber	Failure/Date	Actions
		Cracked seat drain line to valves V-028 and V-029	Removed seat drain lines and capped.
RR Loop A Suction Elbow Instrument Taps	N/A	(10/01) Cracked outer elbow tap line to Valves V-634 and V-633	(9/87) Removed double isolation valves from tap line (10/01) Repaired (10/01) Removed vibration monitoring equipment
RR Loop B Suction Elbow Instrument Taps	N/A	(9/87) Cracked outer elbow tap connection line to valves V-653 and V-654. Cracked outer elbow tap connection line to valves V-656 and V-655 (12/89) Cracked outer elbow tap connection line to valves V-656 and V-655	(9/87) Repaired (9/87) Removed double isolation valves from tap line (12/89) Added tie backs to tap connections

APPENDIX A

CHARTER FOR INDEPENDENT ASSESSMENT

November 12, 2004

INDEPENDENT ASSESSMENT OF THE HOPE CREEK "B" REACTOR RECIRCULATION PUMP

CHARTER

I. STATEMENT OF PURPOSE

A. Christopher Bakken, III, President and Chief Nuclear Officer of PSEG Nuclear LLC, has approved an independent assessment regarding issues raised regarding the Hope Creek "B" Reactor Recirculation Pump ("RRP"). The assessment will focus on the issues, as described below. The assessment may be expanded to include related issues. The assessment will be completed as expeditiously as possible, without compromising its ability to provide a complete, accurate, credible, an independent report to Mr. Bakken. This document establishes the Assessment Charter and identifies the issues more fully infra.

Mr. John Carlin, PSEG Nuclear Vice President Nuclear Assessments will chair the Assessment. Mr. Keenan, PSEG Services Assistant General Solicitor, will provide assistance to the Assessment effort.

II. ISSUES TO BE EXAMINED

The issues define the scope of this Assessment. Specifically, the assessment will address the following:

J. Assess the condition of the "B" Reactor Recirculation Pump.

K. Evaluate the "B RRP performance and integrated recirculation system performance including:

   1. Whether the continuing vibrations being detected in the Hope Creek dyrwell may have been caused by the RRP operation.

   2. Whether vibrations may increase during transients or when the unit is down-powering and how other components in the drywell have been or could be affected by this high-vibration including valves and limit switches.

L. Review the PSEG Nuclear proposed plan including the historical and proposed corrective actions. Specifically review the following:

   1. Adequacy of the corrective actions to address pump and structural vibration concerns including the input basis for the corrective actions (inspections, data, etc.)

   2. The current NDE program and past inspections results. Make recommendations for any changes that may be warranted.

   3. Review available accelerometer data, evaluate the basis and adequacy of the planned probe placement and provide any recommendations for improvement.

4. Review the validity of PSEG proposed acceptance criteria for post-refueling pump and structural vibration levels.

5. Evaluate the root cause report and assess the recommendation to replace the pump rotor, the time of that replacement and any follow-up actions that were recommended.

M. Document all findings, recommendations and conclusions in a written report, including maintaining all supporting reference material used to form the basis of the report.

III. KEY ELEMENTS OF THE ASSESSMENT

The assessment conclusions shall be documented in a report for the President and Chief Nuclear Officer, PSEG Nuclear LLC, stating factual findings. Consistent with the full development of the facts, the assessment shall be conducted in a manner that minimizes the disruption of station operations, and an adverse impact on PSEG Nuclear employee morale.

IV. IMPLEMENTATION

A. Organization

- Mr. John Carlin, will chair the Assessment.
- An engineering team from Sargent & Lundy LLC will conduct the independent review. Members include, but are not limited to: Dr. A.K. Singh, Mr. Brian Voll, and Mr. Ira Owens.
- Mr. Jeff Keenan will provide assessment support.
- Mr. A. Christopher Bakken, III, President and Chief Nuclear Officer – PSEG Nuclear LLC, has reviewed and approved this Charter. The assessment shall proceed under his authority to conduct interviews and review documentation as deemed necessary by the team. Mr. Bakken will approve any changes to this Charter as necessary.

B. Support Personnel

- Clerical support and personnel assistance from outside wil be provided on an as needed basis.
- Sharing of assessment-related information with PSEG Nuclear employees will be conducted on a need-to-know basis.

C. Conduct of Assessment

The assessors will identify issues as follows:

- Interviews. The assessors may conduct interviews of employees associated with the charter issues. The assessors will take notes regarding answers and commentary during each interview. The assessors should afford all interviewees the opportunity to review, correct, and concur with interview notes.

- Review of records and procedures: The assessors will review relevant documentary evidence and use it as appropriate

- Review of regulatory requirements. The assessors will review relevant Nuclear Regulatory Commission ("NRC") and other regulatory agency requirements as part of this review.

The assessors will brief the President and Chief Nuclear Officer, PSEG Nuclear LLC, on the status of the investigation, the schedule for remaining work, and the resolution of the issues.

D. Results/Report

- A comprehensive report, including recommendations as appropriate, will be issued to the President and Chief Nuclear Officer, PSEG Nuclear LLC.

V. SCHEDULE

A. Assessment and Report

- The assessment and written report is required to be completed by November 16, 2004.

Approved: ________________________________
A. Christopher Bakken, III
President and Chief Nuclear Officer
PSEG Nuclear LLC

APPENDIX B

GE NUCLEAR SIL 459S2

APPENDIX C

COMPARATIVE REVIEW OF THE HOPE CREEK RR PUMP SUPPORT CONFIGURATION TO DRESDEN, QUAD CITIES, BROWNS FERRY, AND CLINTON RR PUMP SUPPORT CONFIGURATION

Following are the results of the comparative review of the Reactor Recirculation Pump/Motor assembly support configurations for various plants:

Dresden and Quad Cities

The Dresden Pump/Motor support configurations for both units are similar to that of Quad Cities units 1 and 2.

From Dresden Reference Drawing No. ISI – 114, "Inservice Inspection Class I Nuclear Boiler and Reactor Recirculation Piping", Detail "6", the following are the support quantities and types:

- Attached to the Pump Motor: 3 Snubbers M-1193D-1124, 1125 and 1126
- Attached to the Pump Bowl: 3 Constant Supports M-1193D-1141, 1142 and 1143
- Attached to the Pump Bowl: 3 Snubbers M-1193D-1121, 1122 and 1123

In addition the above supports there is a "Cable Restraint" on the suction side piping that is not designed for normal operating loads.

Browns Ferry

The Pump/Motor support configurations for Units 2 and 3 are very similar. From Browns Ferry Reference Drawing No. 2-ISI-0278-C, Sheets 1 and 2 "Recirculation System Support Locations" the following are the support quantities and types:

- Attached to the Pump Motor: 3 Snubbers 2-47B408S0067, 68 and 69
- Attached to the Pump Bowl: 3 Constant Supports 2-47B408S0064, 65 and 66
- Attached to the Pump Bowl: 3 Snubbers 2-47B408S0061, 62 and 63

Hope Creek

The Pump/Motor support configurations provided on Drawing No.762E179, the following are the support quantities and types:

- Attached to the Pump Motor: 2 Snubbers SSA4 and SSA5
- Attached to the Pump Bowl: 3 Constant Supports HA5, HA6 and HA7
- Attached to the Pump Bowl: 3 Snubbers SSA1, SSA11 and SSA12
- Attached to the Pump Bowl: 1 Pump restraint as shown on Drawing Nos.131C7810 and 761E593

The rigid pump restraint that is attached to the pump bowl and the number of snubbers, (2 snubbers vs. 3 snubbers) attached to the pump motor are the only deviations from Dresden and Browns Ferry.

Additionally we also have reviewed Clinton station's Pump/Motor Support Configuration. Following are the support quantities and types from Reference Drawing No: 768E993 Sheets. 1, 3 and 4.

- Attached to the Pump Motor: 3 Snubbers S369, S370 and S371
- Attached to the Pump Bowl: 4 Constant Supports H301, H302, H303 and H304
- Attached to the Pump Bowl: 4 Snubbers S372, S373, S374 and S375
- Attached to the Pump Bowl: 2 Rigid restraints as shown on Drawing B301 and B302

Since Clinton is not a BWR of the same vintage as the above plants, this review was performed only to provide additional comparative information.

APPENDIX D

SELECTED MEASURED VIBRATION DATA FOR RR PUMP AND RR SYSTEM PIPING

APPENDIX E

RR PUMP HISTORY

November 12, 2004

Reactor Recirculation Pump History

Prepared by: Peter Koppel Component Engineering

Date: 10/21/04

Purpose: Provide a summary of the Hope Creek 'B' Reactor Recirculation pump vibration and mechanical seal history, RF12 plans, and issues status.

1) Pump Vibration Levels: (Industry average vibration level 6-8 mils)
1985-2004 - 'A' Rx Recirc. average vibration level 4-5 mils.
1985-2001 - 'B' Rx Recirc. average vibration level 19-21 mils (pre RF10)
2000 (RF09) - Corrected a large motor misalignment with minor reduction in 'B' Rx Recirc. pump vibration levels.
2000-2001 - Installed and moved several balance weights with no positive affect.
2001 (RF10) - Re-machined coupling fits to correct coupling stack-up.
2001 (RF10) - Found the vibration probe was measuring vibration on an 'egg' shaped section of the coupling. The probe was moved to a round section of the coupling. Vibration levels lowered to 8-10 mils, and steady.
2003 - Replaced mechanical seal and vib. levels rose to 10-12 mils, and steady.

2) Reactor Recirculation System "Freight Train" Noise Issue:
1985-2004 - There have been numerous reports of loud system vibrations when the Rx Recirc pumps are at high speed. (above 1529 rpm)
- The belief is the vibration comes from mechanical and acoustic components in Rx Recirc system, which are excited by the pump's five vane impeller.
2004 - The power upgrade project is installing addition instrumentation on the system in RF12 to determine if the power upgrade will adversely affect the system.

3) Mechanical Seal Leakage Issue: (Industry standard seal life 72 months)
1985-1999 - Station utilized original Byron-Jackson Rx Recirc pump mechanical seals with historically low reliability.
1999 (RF08) - Changed to the Flowserve N7500 cartridge mechanical seal.
1999 - After three months of operation, replaced both 'A' & 'B' Rx Recirc pump seals due to improper initial venting.
1999-2006 - 'A' Rx Recirc. seal remains in service to be replaced in RF13 after a projected 79 months of operation.
2000 (RF09) - 'B' Rx Recirc. seal replaced due to high, but not unsatisfactory leak-off.
2001 (RF10) - 'B' Rx Recirc. seal replaced as a precaution.
2003 - 'B' Rx Recirc. seal replaced (Planned Outage) due to excessive leakage.
2003 - Root cause on the reliability of the mechanical seal (70029861) was completed. It concluded that the most likely cause of the poor seal reliability was a bow in the shaft.

4) Pump Possible Bowed Shaft Issue:

2003 - Root Cause 70029861 found that the pump vibration levels and the mechanical seal wear patterns were similar to Quad Cities, which operated with a documented 8 mil bow in their shaft.

- There is no documented measurement of a bow in the 'B' Rx Recirc pump shaft. Additional measurements will be taken in RF12 to verify.

2003 - Flowserve (pump OEM) believes 'B' Rx Recirc. pump shaft is bowed based on their inspection of the February 2003 failed mechanical seal.

- Two replacement Flowserve fourth generation reactor recirculation pumps were purchased in the mid-1980's and are available in Warehouse 13. Their crates have not been opened, but all the shipping data was verified in early 2004 to be two Flowserve reactor recirculation pump rotating assemblies.

2003 - The decision to replace 'B' Rx Recirc. pump in RF13 was made at the June PHPC meeting. This was based on the findings of root cause 70029861. The mechanical seal performance during Cycle 12 has been satisfactory, and we have not physically measured a bow in the shaft; therefore, there was not need to move the pump replacement into RF12.

5) Mechanical Seal Purge System:

2001-2003 - Both 'A' & 'B' purge line relief valves leaked diverting all the purge water away from the mechanical seals. Operated in this condition for approximately 18 months until the relief valves were replaced in April 2003 (RF11)

2003 - The 'B' Rx Recirc. pump failed in mechanical seal in February had indication of debris scratching on the seal faces.

- Operations routinely monitors the relief valve discharge and can isolate the relief valves if they begin to leak again.

6) RF12 Scheduled Actions Owners: Peter Koppel (Eng) & Rich Keenan (Maint)

- Replace 'B' Rx Recirc. pump mechanical seal with new (not rebuilt) mechanical seal assembly.
- Collect shaft measurements inside 'B' Rx Recirc. pump stuffing box to verify the possible bow in the shaft.
- Re-machine section of pump coupling that is 'egg' shaped.
- Install mechanical seal leak-off line modification on 'B' Rx Recirc pump to properly slope the leak-off lines.
- Walkdown 'B' Rx Recirc pump replacement in preparation of RF13.

7) RF13 Proposed Actions - Owner: Peter Koppel (Eng)

- Replace 'B' Rx Recirc pump with a Flowserve fourth generation reactor recirculation pump.
- Replace 'A' Rx Recirc pump mechanical seal.
- Install mechanical seal leak-off line modification on 'A' Rx Recirc pump to properly slope the leak-off lines.