'B' Reactor Recirculation Pump
NRC Questions 12/01/04

1) **Regarding the S&L conclusion on operating for another cycle, the S&L report does not currently make a conclusion. The report conclusion should address the probability or likelihood of the shaft cracking leading to shaft failure over the next operating cycle.**

   Response:

   S&L concluded that a change out of the Hope Creek 'B' Reactor Recirculation pump in RF13 is acceptable because it is improbable that there will be a shaft cracking failure over the next two operating cycles. Both the vibration levels and the time in service for the pump are comparable to other reactor recirculation pumps in the industry. The 'B' Reactor Recirculation pump vibration levels also have been stable for the last two operating cycles; therefore, there is no reason to believe that the current level of vibration is causing pump degradation.

   The assumption that the Hope Creek reactor recirculation pump shafts have thermally induced cracking is based on the General Electric (GE) documentation (SIL 459) that all the Byron-Jackson (now Flowserve) reactor recirculation pump shafts that have been inspected have had thermal stress cracks. Therefore, while the Hope Creek reactor recirculation pumps can be expected to have thermal axial stress shaft cracks, no data indicates that the thermal stress cracks will propagate to the point of failure without additional mechanical loads. There is no data to indicate that the Hope Creek reactor recirculation pumps are being subjected to the mechanical loads that would cause the thermal stress cracks to propagate into circumferential cracks and ultimately, shaft failure. See response to Question 3 for further details.

2) **Provide technical basis for the 80,000 hours given in SIL 459.**

   Response:

   The GE SIL 459 recommends several actions for stations to take to improve the monitoring of their Byron-Jackson (now Flowserve) reactor recirculation pumps. Those actions include vibration monitoring and an inspection of the reactor recirculation pump after 80,000 hours of operations or approximately 10 years. Neither GE nor Flowserve have a documented basis for this pump inspection frequency. The 80,000-hour inspection interval was developed in 1987, less than 18 months after the first report of reactor recirculation pump shaft cracking. Both GE and Flowserve believe that this inspection frequency was developed based on the initial theories of the cause of the shaft cracking, and shaft crack

propagation projections based on the very limited empirical data available at the time. After SIL 459 was issued, Flowserve performed several years of extensive testing and analysis into the shaft cracking issue. Their final conclusions on the thermal cracking propagation projections and the mechanical loads required to transition the crack into a circumferential crack were different than the initial theories and projections.

The axial thermal stress cracks are by themselves not detrimental to the operation or reliability of the reactor recirculation pump. However, over time and with increased mechanical loading, the axial thermal cracks transition into circumferential cracks, which can lead to shaft failure. The time between the cracks departure from the expected thermal crack propagation line to the ultimate shaft failure is calculated to be a period of 1-2 years with the higher mechanical loading. (See figure 3.7 of reference 3) There is no empirical data to validate this calculated time period. A survey was performed of stations in the Boiling Water Reactor Owners Group (BWROG), and only one station was found that has performed a shaft inspection since 1987. Due to the high cost of removing a reactor recirculation pump and the high associated radiation doses, the industry practice has been to replace the reactor recirculation pumps with a new pump when either pump performance or life cycle management concerns dictate pump replacement.

3) Is there a fatigue growth calculation for the postulated cracks in the shaft?

Response:

Crack propagation calculations exist for the thermal stress induced shaft cracks. These calculations have been verified by both physical testing at power stations in Japan and by the accumulation of empirical data from reactor recirculation pumps that have been refurbished. They indicate that thermal stress cracks are not expected to propagate to the point of shaft failure.

Crack propagation calculations also exist for the circumferential cracks that develop in higher mechanical load cases. These calculations have resulted in a graph, which correlates the mechanical load on the shaft to the minimum crack depth necessary to propagate the crack into circumferential cracks and ultimately shaft failure. (See figure 3.7 of reference 3) They indicate that for pumps operating close to their design point, the mechanical loads are low enough that the highest expected crack depth for thermal stress cracks will not propagate into circumferential cracks. The Hope Creek Reactor Recirculation pumps

operate in a band of 44,000 - 45,000 gpm with a best efficiency point of 39,000 gpm. This represents a 15% deviation from the best efficiency point and indicates a nominal loading increase on the Hope Creek reactor recirculation pumps.

4) Status of all SIL 459 Recommended actions. Discuss RACs temperature and flow.

SIL 459 12/15/87 (Reference 5)

Recommended Action:

(1) Consider installing shaft vibration probes.
Status: Both 'A' and 'B' Reactor Recirculation pumps have two proximity probes (X & Y directions) located on the pump shaft/coupling. The vibration channels, i.e., H1BB –1BBVT7910A/B1/2, indicate in the control room and contain alarms. (See Reference 12)

(2) Consider monitoring RACs effluent.
Status: The RACs system contains radiation monitor H1SP -1SPRE-2534. (See Reference 13)

(3) Inspect pumps with greater than 80,000 hours of operation.
Status: Hope Creek station does not perform periodic reactor recirculation pump inspections. (See References 9 & 10)

(4) Prepare inspection plan to include the following: (Only if pump inspected)
- Method to examine shaft.
- Criteria for return to service
- Repair methods
- Plans for parts replacement

Status: Recirculation pumps not inspected.

(5) Report the results of pump inspections to GE. (Only if pump inspected)
Status: Recirculation pumps not inspected.

SIL 459S1 03/23/90 (Reference 6)

Recommended Action:

None

SIL 459S2 10/21/91 (Reference 7)

Recommended Action:

(1) Consider installing shaft vibration probes - *Repeat from original.*

(2) Consider monitoring RACs effluent. - *Repeat from original.*

(3) Reduce seal purge flow.
Status: Reactor Recirculation pump mechanical seal purge system flow reduced to 1.5-2.5 gpm. (See Reference 11)

(4) Additional recommendations:
- Improve shaft surface condition
  Status: This will be accomplished when the reactor recirculation pumps are upgraded to the 4th Generation reactor recirculation pumps.
- Improve balance and alignment
  Status: The pump alignment was improved in RF09. (WO 60004748) Pump balance correction was attempted, but there was on imbalance to correct. (WO 60004748, 60018593, 60014335)
- Reduced shaft vibration amplitudes.
  Status: Significant troubleshooting has been performed to improve 'B' Reactor Recirculation pump vibration levels. (See Reference 10)
- Reduce operational transient frequency.
  Status: Operators continue to operate the plant in a manner consistent with plant safety, and transition the plant only when required.

(5) Replace parts with upgraded parts.
Status: The station has purchased two 4th Generation Reactor Recirculation pumps, which are the vendor recommended upgrade to resolve all shaft cracking concerns.

<u>SIL 459S3 08/31/93</u> (Reference 8)

This SIL describes shaft cracking in Sulzer-Bingham pumps, which is not applicable to Hope Creek Station.

5) Does GE concur with the S&L statement (page 15 of report) that we meet SIL 459?

Response:

GE does not consider any of the recommendations in SIL 459 or its supplements to be requirements for which the utility must comply. They do not wish to review utility actions to determine compliance. The recommendations they provided are suggested actions to improve component reliability. If the recommended actions were required for the safe operation of the component the actions would be issued in a Potential Reportable Condition (PRC) report under the guidance of 10CFR21.

6) Does the shaft bow increase the chance of a shaft crack?

Response:

The shaft bow does not increase the chance of the initiation of a shaft crack. The shaft cracks are initiated by thermal (not mechanical) stresses. The bow does contribute to the vibration level of the pump.

GE and Flowserve found in their investigations that the amount of shaft mechanical loading being experienced by any particular pump is difficult to quantify. They have been unable to develop a methodology to quantify shaft mechanical loading. The mechanical loading is a function of the forces on the shaft, which include shaft torque, radial impeller thrust, pump vibrations, and residual loads from initial pump installation tolerances. Of these forces the radial impeller thrust is considered the largest. None of the published shaft cracking technical documents, failure analyses, shaft cracking testing, or industry operating experience has identified elevated vibrations as a significant contributor. The exact cause of the high mechanical loading that lead to the circumferential shaft cracks found in the Grand Gulf reactor recirculation pumps has not been determined; however, elevated vibration were not a factor since the Grand Gulf reactor recirculation pump vibration levels are below the industry average.

7) Are the vibrations in 'B' Reactor Recirculation pump (due to the bow) related to the shaft cracking?

Response:

No, the shaft cracking is initiated due to temperature gradients in the shaft that occur during system operation. The bow likely occurred due to the relaxation of stress imparted into the shaft during initial fabrication.

8) Will the vibrations make the shaft cracking worse?

Response:

No, not at their current levels. Also, see the answer to Question (6)

9) Has the risk significance of the recirculation pump (shaft cracking) been assessed and has the PRA group evaluated it?

Response:

The PRA Group has evaluated the reactor recirculation pump shaft cracking issue. Reactor recirculation pump shaft cracking is not included in the PRA model. The PRA group analyzed the risk significance of reactor recirculation pump shaft cracking and found it to be small. The PRA group assessment is as follows:

Input Data:	Number of cracked shafts in the industry	4
	Number of pump operating years in the industry	1778

Notes:

(1) For the purposes of the shaft cracking evaluation, the industry is defined as the 77 reactor recirculation pumps installed in the 35 plants that are members of the BWROG.
(2) The response to Question 6 concluded that vibration levels are not a significant contributor to the propagation of shaft cracking; therefore, the elevated vibration level of the 'B' Reactor Recirculation pump was not included in this evaluation.
(3) For conservatism, the number of pump operating years was reduced to 25% of the calculated value given above.

A conservative approach is to assume that there were 4 failures in 444 pump years. This will give an estimate of 9.0E-3/year. (i.e., 4/444)

The failure experiences indicate that if a crack is developed while in operation, manual shutdown is the most likely outcome. Using a non-informative prior, the likelihood of the consequences other than manual trip given the shaft failure is about 0.25. It is judged that a turbine (or reactor) trip or a small LOCA is equally likely given the shaft failure and manual trip is not initiated.

To be conservative, the risk analysis assumes the following:

1. Annual shaft failure likelihood is 9.0E-3.

2. Given the shaft failure, there is a 0.125 chance a small LOCA is developed and a 0.125 chance a turbine (or reactor) trip event will occur.

Therefore, the additional annual risk (CDF) of Hope Creek operation with one recirculation pump is the summation of the following:

1. 9.0E-3*0.75* 1.44E-6 = 9.7E-9
2. 9.0E-3*0.125*1.6E-6 = 1.8E-9
3. 9.0E-3*0.125*1.36E-4 = 1.5E-7

   Sum = 1.6E-7

The additional risk of Hope Creek station operation with the vibration is about 1.6E-7/year. This is judged to be small.

The above analysis is sensitive to the assumption of shaft failure likelihood. If the shaft failure likelihood is to be increased by a factor of 10, the estimated CDF is increased to 1.6E-6/year.

The analysis is also sensitive to the assumption of the consequence given the shaft failure. If the shaft failure is assumed to cause a small LOCA, the estimated CDF is increased to 1.1E-6/year.

These two sensitivity cases are judged to represent the upper bound of the estimated risk. If the benefits of significant increase of monitoring devices and high operator awareness are factored into the sensitivity cases, most likely the CDF increase is going to be limited to the high E-7 range.

In conclusion, the CDF increase is in the low E-7/year. The distribution is likely in the range from low to high E-7/year. The risk increase is considered small.

10)What is the risk of shaft failure with the bow and cracked shaft?

Response:

The risk for a shaft failure for the 'B' Reactor Recirculation pump is no greater than any other reactor recirculation pump with the same service life.

The shaft bow increases the pump's vibration levels. Vibrations do increase the mechanical loading on the shaft, but the amount is insignificant. The shaft cracks, which are suspected to be in the Hope Creek reactor recirculation pumps, are axial thermal cracks. Flowserve analysis has determined that axial thermal cracks are not detrimental to the operation or reliability of the reactor recirculation pump without additional elevated mechanical loading. There is a large database of

industry experience of reactor recirculation pumps operating without shaft cracking failures for a significant amount of time beyond the amount of service time that 'B' Reactor Recirculation pump will have by the end of Operating Cycle 13.

11) What is the plant's response to a rapid shaft crack (failure of pump shaft)?

Response:

Reactor Recirculation Pump Shaft Break is described in the Hope Creek UFSAR section 15.3.4.

A rapid shaft failure could result in an automatic trip of the main turbine with resultant reactor scram due to the very rapid decrease in core flow and water level swell in the reactor.

In the event a failure of the reactor recirculation pump shaft does not result in a high reactor level trip of the reactor, the operators would remove the pump from service. Based on the nature of the failure this may be accomplished by varying methods.

If vibration levels trend up significantly prior to the failure, plant staff will remove the pump from service based on prescribed procedural limits. (HC.OP-AR.ZZ-0008(Q)) This limit is currently 21 mils.

Additionally, plant staff would enter the procedure for a tripped reactor recirculation pump should a scram not occur based on the observation of loss of the associated jet pump flow without a corresponding trip of the Recirculation Pump Trip Breakers or Motor Generator Set. Similar abnormal operating procedures would be entered due the negative reactivity insertion associated with the loss of the core flow and unanticipated rise in reactor level.

12) Are vibration levels indicated and alarmed?

Response:

Radial vibration levels are detected on both reactor recirculation pumps by two proximity probes (X & Y directions) located on the pump shaft/coupling. The radial vibration levels are monitored in the control room via H1BB –1BBVT-7910A/B1/2. Axial vibration levels are detected on both reactor recirculation pumps by a velocity meter located on the reactor recirculation pump motor. The axial vibration levels are monitored

in the control room via H1BB –1BBVT-7910A/B4. Increases in either vibration level are detected by digital points D5351 and D5352, which provide a visual and audible alarm to the operators via overhead alarm (OHA) C1-E4 "Reactor Recirc Pump Vib Hi." The setpoint for radial vibration is 11 mils and the setpoint for axial vibration is 7 mils. In the event of an alarm operators receive guidance by Operations Procedure HC.OP-AR.ZZ-0008(Q) which gives direction to lower pump speed to lower vibration levels. If the vibration levels cannot be lowered, the pump is removed from service at 21 mils of radial vibration, or 11 mils of axial vibration.

During RF12, additional vibration instrumentation is being added to 'B' Reactor Recirculation pump motor via DCP 80062466. The new instrumentation includes five accelerometers. Three accelerometers (X, Y & Z directions) will be installed close to the upper motor bearing, and two accelerometers (X & Y directions) will be installed close to the lower motor bearing. This new instrumentation is for component trending purposes and is not alarmed in the control room.

13) **Can you reliably detect the initiation and growth of a crack with our current installed instrumentation?**

Response:

The growth of the crack into a circumferential crack can be seen by two methods. First, when the crack departs from the thermal crack propagation line, the 2X vibration peaks will rise and the phase angle will shift. For this reason vibration data is collected monthly. Data is downloaded monthly into the ADRE system, (MP# HC650050) and evaluated for any pump phase angle changes, any sudden increases in 2X vibration levels, and the condition of the vibration orbital plots for indications of shaft cracking. Flowserve has developed analytical predictions of the crack growth of a circumferential crack. The period of time between the cracks' departure from the thermal axial crack propagation line and shaft failure is 1-2 years. This should provide sufficient opportunity of the vibration trending to detect the transition of the crack.

In the event that the vibration trending does not detect the transition of the shaft crack into a circumferential crack, the reactor recirculation pump vibration levels are continuously monitored and alarmed in the control room. There is limited industry experience on actual cracked shafts to determine the amount of reaction time available to the operating crews. In the case of Grand Gulf in 1989, they received a reactor recirculation pump vibration alarm on May 11th, and performed a controlled shutdown of the

plant on May 15th. There were similar results at Sequoia (Westinghouse PWR), which had a cracked shaft in 2002. Upon initial pump start after a refueling outage, the pump vibrations were immediately high. The station analyzed the condition for several hours with the pump in service prior to securing the pump. From these two examples, the only industry experience available indicates that the operating crews will have sufficient time to react and perform a controlled plant shutdown if required.

14)Has any other plant demonstrated the ability to detect a rapid crack?

Response:

Yes, Grand Gulf had a cracked shaft in 1989. See Reference 12. While raising reactor recirculation flow, operators observed a rise in pump vibration level to 17 mils. They continued the flow increase, and vibration levels rose to 32 mils, and they received the reactor recirculation pump high vibration alarm. (setpoint 20 mils) The operators responded by lowering reactor recirculation system flow, and lowering the pump speed. Reactor recirculation pump vibration levels returned to normal. The station remained in this condition for four days while the condition was analyzed. After four days, a controlled plant shutdown was performed.

15)How much time do the operators have to respond to a recognized pump shaft rapid crack?

Response:

The response to a failed shaft is described above in Question 11 and 13. Hope Creek UFSAR section 15.3.4 analyzed the condition, modeled as an instantaneous failure. Any slower developing failure that increases the vibration level of the pump would result in a reactor recirculation pump alarm (OHA C1-E4). The alarm response, HC.OP-AR.ZZ-0008(Q), directs the operators to lower pump speed or remove the pump from service in accordance with prescribed vibration limits. (21 mils for radial vibrations and 11 mils for axial vibrations) This is not a time-based response. The design basis failure of a recirculation pump shaft would result in an automatic trip of the main turbine on high reactor water level. Reactor water level would be recovered normally with the reactor feed pumps.

16)If the shaft where to fail what are the consequences (damage to recirculation piping, seal failure, etc.)?

Response:

If a thermal axial crack in the reactor shaft were to transition into a circumferential crack, industry experience indicates the operating crews will be able to perform a controlled shutdown of the pump and plant and there would be no consequences. In the event that the shaft was to suddenly shear while in service, no reactor recirculation piping damage is expected. A qualitative review of the reactor recirculation pump indicates that upon shaft shear, the pump impeller would first drive upward due to normal in service upward thrust. This should not increase the normal upward thrust on the motor thrust bearing because there is no force to increase that upward thrust. The impeller would then settle down to the bottom of the pump casing where it is not expected to limit any reactor recirculation loop flow, nor damage the pump casing. The pump shaft will continue to rotate until the operators trip the pump. Once the shaft has sheared, there will no longer be a bearing to restrain the shaft radial movement. The shaft can be expected to encounter the sides of the casing in the thermal mixing region. No casing penetration is expected, but mechanical seal leakage is very possible.

This accident was evaluated in the Hope Creek UFSAR section 15.3.4. The breaking of the shaft of a reactor recirculation pump is considered a design basis accident (DBA). It has been evaluated as a very mild accident in relation to other DBAs, such as a loss-of-coolant accident (LOCA). The analysis was been conducted with consideration to a single or double loop operation. The postulated event is bounded by the more limiting case of a reactor recirculation pump seizure.

A postulated instantaneous break of the pump motor shaft of one reactor recirculation pump will cause the core flow to decrease rapidly, resulting in water level swell in the reactor vessel. When the vessel water level reaches the high water level setpoint, L8, main turbine trip and feedwater pump trip will be initiated. Subsequently, reactor scram and the remaining recirculation pump trip (RPT) will be initiated due to the turbine trip. Eventually, the vessel water level will be controlled by high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) flow.

The severity of this pump shaft break is bounded by the pump shaft seizure event, which is evaluated separately. In either of the two events, the recirculation drive flow of the affected loop decreases rapidly. In the case of the pump shaft seizure event, the loop flow decreases faster than the normal flow coast down, as a result of the large hydraulic resistance introduced by the stopped rotor. For the pump shaft break event, the hydraulic resistance caused by the broken pump shaft is less than that of the stopped rotor for the pump shaft seizure event. Therefore, the core flow decrease following a pump shaft break effect is slower than the pump shaft seizure event. Thus, it can be concluded that the potential effects of the

hypothetical pump shaft break event are bounded by the effects of the pump shaft seizure event.

17)Is the end of useful life modeled for the pump?

Response:

No, there is no specific model for Byron-Jackson (now Flowserve) reactor recirculation pumps to monitor the pump performance and to determine the amount of useful life remaining. There are many indications that a pump is reaching the end of its useful life. The pump performance could degrade due to wear of the wear rings, or the pump vibration levels could rise due to wear of the bearings. In the case of Byron-Jackson reactor recirculation pumps, there have been no reports of pump performance degradation, and no indications of wear in the wear rings of pumps being refurbished. There have been indications of wear in the hydraulic bearing in pumps that were being refurbished. Hydraulic bearing wear of a pump in service could be seen by vibration levels slowing going higher, which is not the case in either Hope Creek reactor recirculation pump.

18)The pumps have cracks has this been evaluated as a non-conformance and what are the conclusions (see report pg 21).

Response:

No cracks in the Hope Creek reactor recirculation pump shafts have ever been physically identified. Industry experience collected by the pump vendor (Flowserve) indicates that reactor recirculation pump shafts develop thermal stress cracking after as little as 500 hours of operation. Since the Flowserve has identified thermal stress cracks on all the non-4th Generation reactor recirculation pumps that is has refurbished, it is possible that both 'A' and 'B' Reactor Recirculation pumps, which are 2nd Generation reactor recirculation pumps, have thermal stress cracks. The thermal stress cracks have never been evaluated as a non-conformance because they are located in an inaccessible portion of the shaft, which can only be identified by pump removal. Neither 'A' nor 'B' Reactor Recirculation pumps have ever been removed for inspection. However, the thermal axial shaft cracking condition has been well evaluated by Flowserve, and it does not affect the operation or the reliability of the reactor recirculation pumps.

19) If thermal stress, not bowing, is the basis for the shaft cracking, why does S&L recommend only replacing 'B' in RF13? Why didn't S&L recommend the replacement of both A and B shafts?

Response:

Replacing the 'B' Reactor Recirculation pump first is recommended because it has experienced more seal failures and has higher vibrations that the 'A' Reactor Recirculation pump. Replacement of the 'A' Reactor Recirculation pump is recommended for RF14 unless there is indications of pump degradation before that. Based on U.S. industry experience, reactor recirculation pump shaft failures due to thermal shaft cracking have not occurred. The number of operating hours that the 'A' pump will have experienced by RF14 is not atypical for reactor recirculation pumps in U.S. plants. Therefore, shaft failures are not expected to occur between now and RF14.

20) Provide a description of the new vibration monitoring program that will be implemented during the next cycle.

Response:

The original vibration program will remain in place for both the 'A' and 'B' Reactor Recirculation pumps. This program includes two proximity probes (X and Y directions) located on the pump shaft/coupling. These probes provide radial vibration data, which is used for engineering trending purposes, and is displayed and alarmed in the control room for continuous operations monitoring. The program also includes one velocity meter located on the top of the motor. It provides axial vibration data, which is used for engineering trending purposes, and is displayed and alarmed in the control room for continuous operations monitoring.

During RF12 additional vibration instrumentation is being installed on the 'B' Reactor Recirculation pump motor only. (DCP 80062466) This additional instrumentation includes two accelerometers (X and Y directions) located low on the motor housing; and three accelerometers (X, Y, and Z directions) located on the top of the upper motor bearing. These detectors will be read at a data collection cabinet outside the containment, but not in the control room. The data will be periodically downloaded to the Stations vibration program for engineering trending purposes.

<u>Description of Drywell Piping Vibration Monitoring System (DCP 80062466)</u>

Vibration monitoring instrumentation is being installed on various piping systems throughout Hope Creek. This instrumentation is required for several reasons:

1) To assess any increases in flow-induced piping vibration that might occur as a result of future power uprate; and
2) To implement Independent Assessment Team recommendations for assessing the magnitude of drywell piping vibrations that may be occurring due to reactor recirculation pump rotational vibration or pump vane pass frequency excitation of those systems. Monitoring will be conducted during plant startup and during the operating cycle.

In the Drywell, vibration instrumentation (100 accelerometers) are being installed to monitor approximately 40 locations. Systems to be monitored will include the recirculation system, RHR, main steam, and feedwater and selected attached components and piping. The specific locations are based on recommendations developed by the Power Uprate project and also the Reactor Recirc Vibration Independent Assessment Team. The drywell instruments will be connected through penetrations to two local data acquisition systems located in rooms 4303 and 4310 adjacent to the drywell.

In the Turbine Building and steam tunnel, vibration instrumentation (24 accelerometers and 20 strain gages) is being installed to monitor approximately 18 locations. Systems to be monitored will include main steam, feedwater, and extraction steam in the Turbine Building and steam tunnel. These instruments will also be connected to one local data acquisition system located in Turbine Building Room 4101.

21) Were any safety related components rendered inoperable due to the vibration?

Response:

Yes. The safety-related components are discussed below:

1BC-HV-F050A, testable check valve in RHR A return line.

This valve (and similar valve 1BC-HV-F050B in RHR B return line) prevents back flow from the RR system in conjunction with containment isolation valves 1BC-HV-F015A/B.

Component degradation to this valve has been limited to external, non-pressure boundary elements, which are utilized only during valve seat testing, conducted during unit outages. These initially included

detachment of the air actuator cylinder, but more recently the apparent missing linkage key and looseness (but no degradation) of the top mounted limit switch power supply. Degradation or the loss of these elements would not allow for testing of the valve during scheduled outage testing. Loss of the power supply and or linkage key would limit the ability of the operator to confirm valve position, which is passively closed during normal RR pressure and open upon RHR initiation.

Hence, the noted conditions would not by themselves be considered as an inoperable condition for the valve to function as designed.

1BC-HV-F060A/B and 1BC-HV-F077, manual gate valves

Valves 1BC-HV-F060A/B are locked open and used to isolate testable check valves F050A/B for maintenance or test. Valve 1BC-HV-F077 is locked open and used to isolate valve F009 for maintenance or test.

Component degradation has been limited to external, non-pressure boundary elements. These components are utilized only for verification that the valves are open (limit switch indication hardware and cables) or to manually put the valves to a closed position (hand wheel, gear box cover, pinion gear and yoke nut).

Hence, the noted conditions would not by themselves be considered as an inoperable condition for the valves to function passively in the open position as intended.

22)The cause of step change in pump vibration at RF11 is not known. How can the statement its okay be made without knowing the cause of the step change (pg 10 of report)?

Response:

The 'B' Reactor Recirculation pump vibration signature prior to the planned outage in March 2003 (on month prior to RF11) had been steady since the pump was placed in service after RF10. The overall magnitude of the vibration was steady at 7-8 mils prior to the outage and steady at 9-10 mils after the outage. The vibrations continued at this magnitude after RF11, and remained steady for all of Cycle 12. The vibration spectrum prior to the outage indicated a 1X vibration magnitude of 5 mils and a phase angle of 269°, and the 2X magnitude of 0.84 mils and a phase angle of 284°. The vibration spectrum after the outage indicated 1X vibration magnitude of 7.5 mils and a phase angle of 292°, and the 2X magnitude of 0.82 mils and a phase angle of 270°. The shift in the 5X

vibration was within the normal data scatter. This change in magnitude and phase angle vibration data indicates a change in the pump coupling stack-up. During the outage, the pump was uncoupled and the vibration probes removed to facilitate the replacement of the mechanical seal. The replacement of the mechanical seal would not affect vibration levels; however, the uncoupling and re-coupling of the pump and motor may affect the vibration levels. During this process the pump hub is removed and reinstalled. Any variation in the position of the pump hub would translate into a change in vibration levels. In addition, the lower rabbit fit of the coupling spacer was oversized to allow proper alignment during RF10. The resulting rabbit fit was 4 mils oversized. In this condition, the pump can be properly coupled, but repeatability of the pump alignment was difficult. With the coupling spacer in a slightly different location, a slight coupling imbalance would affect the vibration readings. The lower rabbit fit of the 'B' Reactor Recirculation pump coupling spacer is being restored during RF12 under WO 60036037. The removal and recalibration of the vibration probes could also result in a step change in vibration signature, but the shift in vibration phase angle indicates that this is not an instrumentation only concern.

As stated in the report, the post-RF11 vibration amplitudes were similar to the pre-RF09 vibration amplitudes and the vibrations are not trending upward. Considering this evidence, the step change in vibration does not indicate pump degradation.

23) What was the affect of moving the probes and is the vibration data consistent? It appears that there are errors in the S&L graph.

Response:

The specific effect of moving the vibration probes cannot be quantified from the available data, since the balance weights were removed about the same time that the probes were moved. However, considering that the pre-RF09 and post-RF11 vibration amplitudes are similar, the effect of moving the probes does not appear to be significant.

There have been two configuration changes made to the Reactor Recirculation pump vibration instrumentation.

During RF10 (November 2001) the X and Y proximity probes were inadvertently switched. (CR 70043098) The two proximity probes are identical. The X and Y locations are arbitrary and only need to be 90° apart to perform their required functions. With the probes switched there is no change in the accuracy of the readings, but the continuity of the data trend will be challenged if the switch was not properly identified. The

switch was identified immediately and the vibration data trend annotated, thus permitting proper continuity in the vibration trend.

During RF10 (November 2001) the X and Y proximity probes were moved via DCP 80036347 from the original location sensing on the lower flange of the coupling spacer to the flange of the pump hub. This was done because it was believed at the time that the OD of the lower flange of the coupling spacer was out of round and giving a false vibration reading. During RF12 (November 2004) the coupling assembly was dimensionally verified under WO 60036037 and both the OD of the lower spacer flange and the OD of the pump coupling hub flange were found to be round to within 0.001 inch. Therefore, the DCP installed in RF10 had no affect on the pump vibration trend. The vibration readings taken before and after the DCP installation are of the same accuracy. Any actual effect of moving the vibration probes cannot be quantified, since the balance weights were removed about the same time that the probes were moved. However, considering that the pre-RF09 and post-RF11 vibration amplitudes are similar, there does not appear to be an effect from moving the probes.

The S&L report was created from vibration data supplied to them from PSEG Engineering. The graph contains overall vibration data in both the X and Y directions. PSEG Engineering independently developed an almost identical graph, and confirmed that the S&L graph in the report is accurate.

24) The pump vibration levels were highest in cycle 10 and lower now. Why is the rate of vibration related components failures increasing?

Response:

The vibration levels in 'B' Reactor Recirculation pump are not related to the failed components in the recirculation system. The vibration levels in the reactor recirculation pumps are measured by proximity probes located on the pump coupling. The 'B' Reactor Recirculation pump vibration levels last cycle were 8-10 mils. The reactor recirculation pump has the following clearances:

Lower pump wear rings	50-52 mils
Upper pump wear rings	25-26 mils
Hydraulic bearing	13-15 mils
Thermal mixing region	25-26 mils

The prominent vibration peak of the reactor recirculation pumps is at 1X running speed. This is a measure of the excess energy in the pump's

rotating assembly caused by inconsistencies in the alignment of the motor bearings, the stack-up of the coupling, and the bow in the shaft. Due to the pump clearances, this energy is not transmitted to the piping system.

This can also been seen in the observed vibration spectrum. The predominant frequency peak in the reactor recirculation system is 5X and 10X running frequency. This is a result of hydraulic forces from the pump's five vane impeller. The 5X and 10X running frequency vibration peaks have an insignificant contribution to 'B' Reactor Recirculation pump's vibration level.

The historical data trend of F060/F077 failures documented in engineering evaluation H-1-BB-CEE-1862 displays that increased pump vibration levels does not correlate to increased number of component failures.

25) Did S&L look at just the current (RF12) small bore ISI results or did they also look at previous data?

Response:

The S&L team reviewed the small-bore ISI results for RF12 and from each of the previous outages at Hope Creek station.

26) Vibration of Large Bore Piping. References 6.2 & 6.3. S&L report just critiques these references but does not come to any conclusion. Please provide copies of References 6.2 & 6.3.

Response:

Copies provided.

27), Is the rigid thermal restraint inducing stresses (Page 12 of report, Hope Creek pump snubber supports are less)

Response:

The Independent Assessment Of Hope Creek Reactor Recirculation System And Pump Vibration Issues dated November 12, 2004 on page 12 states:

"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 somewhat 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."

PSEG has reviewed the current Hope Creek reactor recirculation pipe stress analysis. (C-0142, Revision 10, "Stress Report for Recirc. Loop B" and "C-0141, -Revision 11, "Stress Report for Recirc. Loop A and RHR (Inside Drywell)"). The stress calculations include the pipe struts and the current configuration of snubbers on the motor. The loads on the struts on both loops are significantly below allowables. We have discussed the configuration with General Electric's pipe stress analyst. The analyst states the strut is used on newer Type 4, 5, and 6 reactor designs. He further states the strut is a key part of the recirculation piping support system. A review of the Byron Jackson stress report for the RRP (VTD PN1-B31-C001-0137, Revision 2) showed that the strut load on the RRP casing was not explicitly addressed. The absence of this documentation has been entered into PSEG's corrective action program. (CR No. 70042757)

The RRP rigid restraint configuration is slightly different from the design dimensional tolerances specified in design drawings. (VTD PN1-B31-G003)-0022, Revision 7. The attachment of the restraint to the "A" RRP is 5/8" lower than the lower than the minimum vertical offset from the strut centerline as compared to its opposite end attachment to the biological shield wall. The rigid restraint on the "B" RRP is within drawing tolerance. The disposition of this discrepancy has not been located in historical records. The issue has been entered into PSEG's corrective action program. (Notification 20214905)

28) Is there displacement data for large bore piping (pages 23 and 24 of report)?

Response:

The RR Displacement Acceptance Criteria was specified and is referenced as reference 6.4 (GE Report GENE-000-0027-4832-01, Revision 1 "PSEG Nuclear LLC Hope Creek Generating Station Recirculation & RHR Piping Start-up Test Criteria" (VTD 326534)).

The dynamic test data and derived displacement data, obtained in April 2004, is reference 6.3 (Ref: Calculation HC-06-301, Revision 1, "Hope Creek Recirculation System Vibration Data Reduction" (VTD 326747). Both the testing and the acceptance criteria identified RR large bore piping locations where the piping displacement were evaluated.

29) Provide copy of reference 6.4 – S&L critiques the reference report but does not draw a conclusion.

Response:

Copy Provided

30) There is a 3% reduction in core flow; why and what are we doing? Is it leading to a pump failure?

Response:

The reduction in core flow is on the order of 3% and is not leading to pump failure. The concern of reduced total core flow is being evaluated by reactor engineering. They have identified several possible causes for the reduction in flow, and pump degradation was originally one of the possible causes.

Over time any pump could experience wear ring wear, which could reduce the efficiency of the pump. Industry experience has indicated that the reactor recirculation pump wear rings to not wear while in service. Review of the pump clearances given in the response to Question 24 confirms that pump wear ring wear is not likely. There are numerous reactor recirculation pumps in the industry, which have more service time then the Hope Creek, and they have not identified any pump performance degradation. Flowserve experience refurbishing reactor recirculation pumps has found the pump wear rings to be in like new condition.

Reactor Recirculation pump degradation is no longer being considered a possible cause for the flow reduction.

Reactor Engineering continues to evaluate the exact cause of the reduced core flow, but has concluded that all of the possible causes impact only economic performance and not reactor safety performance.

31)With respect to the Union of Concerned Scientist response to the S&L report, does PSEG plan to respond to the UCS letter?

Response:

No. PSEG does not intend to respond to the UCS letter.

32)Provide copy of plan to address the recommendations in the S&L report.

Response:

The attached Table 1 is a compilation of the independent assessment team recommendations and CAP Notification Numbers.

33)Plan for monitoring the recirculation pump vibrations during the next cycle (operator guidance). Provide a copy of the procedure for operations in the event of a shaft crack indication. Both Hope Creek Reactor Recirculation pumps have vibration levels indicated and alarmed in the control room. In the event of an alarm the operators follow the actions in procedure HC.OP-AR.ZZ-0008 pages (Reference 4).

Response:

See procedure HC.OP-AR.ZZ-0008(Q), supplied with the references.

References:

1) An Advanced Design Main Coolant Pump for BWR Plants, S. Gopalakrishnan, BW/IP International Pump Division, March 1996.
2) Analytical Investigation of Thermal Cracking in Reactor Recirculating Pumps, S. Gopalakrishnan, BW/IP International Pump Division, October 1992.
3) Crack Propagation in Main Coolant Pumps, S. Gopalakrishnan, BW/IP International Pump Division.
4) Evaluation of Main Coolant Pump Shaft Cracking, EPRI, 1992.
5) GE SIL 459, dated 12/15/87.
6) GE SIL 459S1, dated 03/23/90.
7) GE SIL 459S2, dated 10/21/91.
8) GE SIL 459S3, dated 08/31/93.
9) Mark Bezilla letter dated 06/12/97, Ser# GMHC-97-021.
10) H-1-BB-MEE-1878, 'B' Recirculation Pump Vibration Analysis.
11) HC.OP-SO.BB-0002(Q), Reactor Recirculation System Operating Procedure.
12) OE 3351, Grand Gulf Shaft Crack.
13) OE 3557, Grand Gulf Update.
14) OE 3565, Grand Gulf Update.
15) P&ID M-13-1, Reactor Auxiliaries Cooling System.
16) P&ID M-43-1, Reactor Recirculation System.
17) HC.OP-AR.ZZ-0008(Q), Overhead Annunciator Window Box C1, Operations Alarm Response Procedure.

Table 1 – Independent Team Recommendation – PSEG Actions

	Independent Team Recommendation	Page #	Notification #
1	The Reactor Recirculation (RR) pumps speed is limited to 1510 rpm, and at this speed the maximum core flow achieved in the past was 103 million. 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 latter is deemed a significant operational concern, it should be investigated during RF12.	4	20213970
2	The RR pump instrumentation being added is being installed per a temporary modification. 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.	11	20214011
3	Review RR pump instrumentation at other plants, including how the data from the instrumentation is being used, to help verify that the type and amount of instrumentation being added is appropriate.	11	20214011
4	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.	12	20214012
5	For the "B" RR pump, alignment of the coupling and checking alignment when the pump is recoupled during RF12 is recommended.	16	20214013
6	The "B" RR 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.	16	20214013
7	The Hope Creek RR pumps 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)	17	20214014
8	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	17	20214014

	Independent Team Recommendation	Page #	Notification #
9	Considering the age and time in service of the RR pumps, the Station should be prepared to rebuild the RR pumps because of capacity degradation or rapidly increase in vibrations.	17	20214017
10	The replacement "B" and "A" pump rotor on hand should be checked for rotor balance and shaft straightness before installing in the pump casings. New couplings included in the replacement packages should be checked for concentricity, squareness, and balance.	17	20214017
11	The DCP, installation plans, access and rigging plan, and inspection of the replacement parts should be done during or as soon after RF12 for replacement of "B" and "A" pumps.	17	20214044
12	The replacement pump parts on hand do not include seal cartridges. The intent is to rebuild existing seals at the Stations, using 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.	17	20214017
13	Both "A" and "B" RRPs have operated over 130,000 hours and are approaching a perceived end of useful life. 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.	17	20214017
14	More accurately estimate the time "A" and "B" pumps have operated. Collect similar data form other plants. Estimate the remaining life of the "A" and "B" pumps based on data from other plants.	17	20214017
15	The acceptance criteria for March 2004 monitoring 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. 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.	23	20214018

	Independent Team Recommendation	Page #	Notification #
16	The March 2004 vibration monitoring acceptance criteria are in terms of displacement. When higher frequency harmonic excitation is monitored, as is the case with vibrations caused by vane pass frequencies, it is advisable to also establish an acceleration acceptance limits in addition to the displacement limits. This recommendation is applicable to EPU vibration monitoring.	23	20214018
17	For developing the acceptance criteria for the March 2004 monitoring, response spectrum was 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. This recommendation is applicable to EPU vibration monitoring.	24	20214018
18	Vibration measurements at RR pump speeds above 1500 rpm are planned. 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. Thus, more comprehensive monitoring of the RR and RHR piping than planned is warranted for EPU. This recommendation is applicable to EPU vibration monitoring.	26	20214018
19	The susceptible valve component should be included in the EPU vibration monitoring program. The most effective number of sensors required can best be determined from analytical models that provide accurate vibration response characteristics.	26	20214019
20	It is planned to determine the acoustic characteristics of the RR system. In order to benchmark the acoustic model, dynamic pressure data should be collected 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) during power ascension up to the maximum speeds at which the RR pumps will be operated.	26	20214042
21	The acoustic modes predicted by the acoustic model will be strongly dependent on the speed of sound used in the analysis. Means for benchmarking the acoustic velocity used in analytical models should be investigated.	26	20214042

	Independent Team Recommendation	Page #	Notification #
22	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.	27	20214018
23	The vibration 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.	27	20214018
24	Acceptance criteria should be developed for the monitored valve components.	27	20214018
25	Vibration monitoring 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.	27	20144018
26	The analytical finite element model results for the current configurations of the F060A/B and F077 valve operator assemblies have not been assessed to provide 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. .	31	20214019
27	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 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.	31	20214019

	Independent Team Recommendation	Page #	Notification #
28	The stated action to be taken for Operation 0150 is to include sufficient post mod testing to ensure goals are met. At this time what is a "sufficient testing" has not been defined. It is recommended that post mod testing of the manual gate valve top works should include collection of vibration data including collection of data at pump speeds above 1500 RPM.	32	20214019
29	Repetitive failures of yoke nuts have been established (valve F060A, F077). It is recommended that the failure of these components should be addressed in the recommended failure mode assessment.	32	20214019
30	It is recommended that 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, a similar inspection of valve F060B should be conducted. 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	32	20214019
31	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. It is recommended that the monitoring system should be available and/ or the program implemented, as needed, during the full operating cycle.	32	20214020
32	During the spring 2004 outage, modification to repair the failed cylinder of valve F050A included replacement of like for like parts. Initial walk down conducted during RF12 note indications that the actuator cylinder exhibits play. Therefore, there is reason to believe that these components will continue to fail if simply replaced. It is recommended that valve operator should be modified during RF 12.	33	20214019
33	The inspection activity task for inspection of valves similar in design to FA050A should define the specific attributes to be inspected. 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.	33	20214019

	Independent Team Recommendation	Page #	Notification #
34	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.	44	20214041
35	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.	44	20214041

ATTACHMENT 8.5

RF10 'B' Reactor Recirculation Pump Coupling Maintenance / Troubleshooting Summary