Metzinger

PSEG

Public Service Electric and Gas Company P.O. Box 236 Hancocks Bridge, New Jersey 08038-0236

Nuclear Business Unit

TO: File

FROM: Mark B. Bezilla
General Manager - Hope Creek Operations

SUBJECT: REACTOR RECIRCULATION PUMP SHAFT AND COVER RELATED DECISION

DATE: June 12, 1997
GMHC-97-021

The purpose of this memo is to document my decision inregard to the Hope Creek Reactor Recirculation (RR) Pump Shaft and Cover activities for RFO7.

My decision is to not perform any inspections or replacements of the Hope Creek RR pump shafts or covers during RFO7.

The basis for this decision is as follows:

- No BWR 4 has had a shaft failure (break) to-date
- No BWR has had a confirmed (documented to the industry) cover to cooler leak
- Plants that have had shaft failures reported that they had unusual vibration indication 1 to 3 days prior to failure (e.g., had forewarning)
- Seven (7) "sister" BWR 4's have greater than the number of operating hours that Hope Creek should have accumulated by RFO8
- Vibration instrumentation currently exists to detect unusual behavior of RR pump vibration
- Leak detection exists for both the drywell and the RACS system
- Procedures exist to direct actions regarding RR pump performance problems, and leakage into either the drywell or RACS system

(Reference the attached G. Overbeck to M. Bezilla memo of 6/4/97)

However, based on the items discussed in the attachment I am directing the following item be accomplished:

Fakhar	1.	ISI 10 year Inspection Plan to be updated based on no disassembly of the RR pump shafts or covers in RFO7.
Roberts	2.	Enhanced vibration data gathering (equipment) and monitoring (analysis) be implemented prior to startup from RFO7.
Roberts Wagner	3.	Procedural guidance dealing with: * RR pump problems (including vibration based actions) * Drywell leakage * RACS system inleakage (from contaminated systems) be reviewed and enhanced or strengthened as required, prior to startup from RFO7.
Zerbo (lead) Wagner Roberts	4.	Refresher training be provided to the operators dealing with: * RR Pump Problems * RR Pump Vibration Instrumentation Capabilities * Drywell Leakage * RACS System Inleakage (from contaminated systems) This training shall utilize the enhanced procedures from Item 3 above and shall include applicable operating experiences of a similar nature. This training shall be conducted prior to startup from RFO7.
Roberts (lead) Zudans Anderson Jones	5.	A plan be formulated to use RFO7 as a "data gathering" opportunity to "SCOPE" the drywell for a "future" RR pump shaft and/or cover inspection or replacement. This "Scoping" plan is to be developed by 9/1/97.

Finally, I would like to acknowledge my staff for their efforts in providing me with a thorough "picture" of RR pump shaft and cover issues within the industry. The staff's efforts have allowed me to determine that there are no nuclear safety issues and that my decision is a business decision.

By copy of this memo, I request that Pete Kordziel enter this memo and the requested five (5) Action Items into the Action Request system as a BPAR.

cc:

T. Anderson	M. Headrick	J. Pike
L. Aversa	J. Hilditch	P. Roberts
J. Benjamin	S. Jones	E. Salowitz
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D. Crouch	R. Metzinger	W. Sutton
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G. Englert	J. Miller	M. Trum
A. Faulkner	M. Mohney	L. Wagner
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Nuclear Business Unit

TO:	Mark Bezilla General Manager - Hope Creek Operations
FROM:	Gary Overbeck, P.E. G.J. Overbeck /cs Director - Hope Creek and Component Engineering
DATE:	June 4, 1997
SUBJECT:	Engineering Director's Recommendations Concerning Reactor Recirculation Pumps Shaft and Casing Cracking Inspections for RFO 7
REFERENCE:	E-Mail Memorandum from Deborah Schultz and Carl Fuhrmeister, GE SIL 459 - Reactor Recirc Shaft & Cover Cracking, dated 6/4/97
ATTACHMENT:	Director's Evaluation of the Basis for System Manager's Recommendation

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The purpose of this memorandum is to document the Engineering Director's conclusion concerning expansion of RFO 7 work scope to include inspection of Hope Creek's reactor recirculation pump shafts and covers for thermal fatigue cracking. On 6/2/97 the System Manager and Design Engineer presented the results of their research and evaluation into industry experience and its application to Hope Creek. The results are documented in referenced memorandum and enclosed for clarity

Based upon the referenced information, subsequent discussion, and careful consideration of the System Manager's recommendation, it is concluded that inspection of the reactor recirculation pump shaft and covers could be deferred for the following reasons.

- No BWR 4 reactor recirculation pump shaft failures have been reported (Enclosure 2 of Reference).
- Seven BWR 4 plants have had reactor recirculation pump operate well in excess of Hope Creek's 84,000 hours without shaft or casing inspection or replacement. In addition, the cumulative hours after the next 18 month fuel cycle (approximately 97,000) is below the cumulative hours of six BWR 4 plants (Enclosure 2 of Reference).

- The failure of a reactor recirculation pump shaft is analyzed in the Hope Creek UFSAR.
- Prior to failure of reactor recirculation pump shafts in non-BWR 4, the shafts have exhibited predictive vibration trends which allow mitigating actions to be taken. Plants that have experienced shaft failure (circumferential cracking) also experienced unusual vibration levels from 1 to 3 days prior to the event. All plants were successful at isolating the reactor recirculation pump prior to catastrophic pump failure.
- Hope Creek's normal, abnormal and alarm response procedures provide guidance to the operator to minimize pump vibration and to shut down the pump if vibrations cannot be maintained below acceptable levels.

Recognizing the business risk associated with a reactor recirculation shaft failure, the following should be accomplished to enhance our monitoring and improve our ability to detect and response to an imminent shaft failure.

- Improved pump vibration monitoring
- Enhanced procedural guidance
- Shaft failure refresher training for plant operators
- Development of standing plans to replace the shafts and covers

Since my recommendation and basis differs from the System Manager's, I have attached my evaluation of each point she uses to draw her conclusion. I do not believe there is a safety issue, and her recommendation is too risk adverse given our current condition and no safety issue.

The foregoing engineering management decision was accomplished through the dedicated research and investigation provided by a board spectrum of individuals from system, design, and component engineering, licensing, and operations. The safety significance, basis for regulatory compliance, and component reliability was assessed in a thorough and professional manner and provided a sound basis for our decision.

If you need more information or clarification, please contact me.

cc: L. Storz
E. Simpson

Distribution:

D. Schultz	C. Fuhrmeister	P. Smith	P. Duke
J. Flanagan	C. Morlock	L. Wagner	A. Fakhar
D. Powell	P. Roberts	J. Zudans	J. Hawrylak
G. Englert	M. Cirelly	S. Roche	M. Headrick

ATTACHMENT

Director's Evaluation of the Basis for System Manager's Recommendation

The numbering and order follows those used by the System Manager.

1. *The limited Hope Creek specific information available is not sufficient to support deferral. While the vibration data obtained and reviewed to date indicates no change in orbital shift, this data is not predictive.*

Evaluation

HCGS vibration data, including orbit analysis, of the reactor recirculation pumps show little change since 1992. The most recent orbit analysis data was acquired on 3/17/97 due to a high vibration alarm on 1B-P-201. The cause of the alarm was position change corresponding to pressure changes in the pump seals. The most recent vibration analysis was performed on 11/19/96 during the startup of the reactor recirculation pumps following the mini-outage. This data was compared to similar data acquired on 9/11/92. Data analysis indicates that there has been no significant change in pump response which suggests the mechanical condition of the pumps have not degraded.

Most plants that have experienced shaft failures reported they had experienced unusual vibration levels from 1 to 3 days prior to the event. All plants were successful at isolating the reactor recirculation pump prior to catastrophic pump failure. In addition, our pumps have demonstrated good performance with low vibration levels compared to other plants thus improving our ability to detect imminent failure.

More frequent and enhanced monitoring will improve our ability to detect degrading conditions before failure and, therefore, this is recommended to improve our predictive capability.

2. *The industry data concludes that every BWR 4 that has performed an internal pump inspection has found some degree of cracking. Hope Creek has not maintained RACs temperatures and flows consistent with vendor recommendations or GE assumptions which can adversely affect the crack initiation phenomenon. Therefore, engineering judgment suggests that if we looked right now Hope Creek recirc pump shafts and covers would exhibit cracking. Because the covers are ASME Code Class 1 components with a known defect, we are compelled to inspect them.*

Evaluation

Hope Creek has not maintained RACs temperatures and flows consistent with vendor recommendations which could adversely affect the crack initiation phenomenon. However, once a thermal induced crack is initiated it is self-relieving and will only propagate through mechanical force interaction. A crack would have to transverse axially down the shaft and then circumferentially to cause shaft failure. As stated previously vibration monitoring and detection of excessive vibration has been used to prevent shaft failure during operation.

With respect to the ASME Class 1 casing, a defect is not known. We have no definitive information indicating the casings have a defect. Cover cracking is a potential problem per the GE SIL. The concern described in SIL 459 S1 involves a possible failure of the drilled hole heat exchanger at Dresden 3. Increased levels in the closed cooling water (CCW) surge tank were noted; however, investigations concluded that in-leakage source was from inside the drywell not a confirmed failure of the drilled hole heat exchanger.

Problem reports 950517153, 951110087, and 960512126 document a current condition at Hope Creek where contamination is present in the RACS demineralizer resins. The source of leakage was not identified but was reported to be "a very small heat exchanger leak". Components with potential to leak into RACS are the RWCU pump cooler, RWCU non-regenerative heat exchanger, recirculation pump shell and tube and drilled hole heat exchangers.

In addition, HCGS is not required to inspect the ASME 1 pump casing unless we open the pump for performance of other maintenance (i.e., the Code no longer requires the pump casing to be inspected at a specific interval).

3. *The Hope Creek procedures require a safety evaluation for this issue prior to the conclusion of RFO 7. Specifically, NC.NA-AS.ZZ-0059(Q), Section 3.2.3 identifies the application of Generic Letter 91-18 to degraded or nonconforming conditions affecting SCCs described in the SAR. G.L. 91-18 defines a nonconforming condition as "operating experience or engineering reviews demonstrate a design inadequacy". Interviews with Byron-Jackson engineering personnel and reviews of the technical alerts and 4th generation designs for the shaft and covers confirms that shaft and cover cracking is due to design inadequacies caused by the introduction of seal purge. In addition, Hope Creek ran from 1986 to 1992 with high seal purge rates and is subject to this phenomenon.*

Evaluation

Whether or not a safety evaluation is required prior to the conclusion of RFO 7 is not material to whether or not we need to do an inspection. I do not concur that we have a nonconforming condition. Our pumps are installed per design and we meet our design and licensing bases.

The necessity of a safety evaluation will be referred to Licensing. Nonetheless a draft safety evaluation has been completed should one be necessary.

4. *The changing regulatory environment at this time presents a challenge to the assessment of this issue. SECY paper 97-35 and recent proposed rulemaking suggests that a failure to perform the inspection at the first outage past 80,000 operating hours will be viewed as a "de facto modification" requiring a 50.59 safety evaluation. In addition, it will be difficult to credit compensatory actions to offset the change in probability of occurrence of an accident in the SAR. The fact that we were changing the "probability of occurrence of an analyzed accident in the SAR" made it very difficult to justify that we have not created an unreviewed safety question (USQ) by deferring this activity. While it may not be law to date, it is possible that it could be by the end of this year complicating our startup from RFO 7.*

Evaluation

The SECY paper is not law, and the industry does not concur with the current proposed rulemaking on this very issue of increasing the probability of occurrence ever so slightly. Nonetheless, the existing operator instructions could be used as compensatory measures to offset the increase in probability by preventing the event from occurring. Based on the compensatory measures of shutdown of the recirculation pumps at appropriately vibration levels, the proposal to defer inspection does not increase the probability of occurrence or a malfunction of equipment important to safety previously evaluated in the SAR.

5. *There is not another outage of sufficient duration planned until RFO 14 at which point reactor recirculation pumps will have over 168,000 hours. With the exception of Vermont Yankee, that value is higher than any other BWR 4 plant that has not inspected to date.*

Evaluation

The duration of future outages is a management decision that will be made, in part, on the needs to maintain our operating assets. The duration of RFO 7 is not a compelling argument. Trending of industry data and pump performance will provide input as to time to replace the shafts. It is recommended that we develop standing plans to replace the shafts and covers.

To: Gary Overbeck
From: Deborah Schultz@NUC.HC.TECH
cc: Mark Cirelly@NUC.HC.TECH, Peter Roberts@NUC.HC.TECH, Mark Bezilla@NUC.HC.TECH, David R. Powell@NUC.EPB.LIC, Carl Fuhrmeister@NUC.EPB.MECH, John Zudans@ NUC.NSF.QA
Bcc:
Subject: GE SIL 459 - REACTOR RECIRC SHAFT & COVER CRACKING
Attachment: ~~SORC1.DOC~~, s459data.xls
Date: 6/4/97 11:23 AM

This e-mail and the attached white paper are provided in response to your request of June 2, 1997. The white paper was prepared by the BB Design Engineer (C. Fuhrmeister) and the BB System Manager (me) at the request of plant and engineering management to support a SORC presentation on the deferral of the reactor recirculation pump augmented ISI activities from RFO 7. A contingency 10CFR50.59 safety evaluation has also been drafted to ensure compliance with plant procedures and is located in the reactor recirc system files at Hope Creek.

The following discussion is provided to satisfy your request for the system manager's technical recommendation on this issue. Please note that it is consistent with the position I have maintained since April 22, 1997, when this issue was first identified to Hope Creek station management at an RFO 7 challenge meeting.

I recommend we perform the internal reactor recirculation pump inspections in accordance with the ISI augmented inspection plan. The basis for this recommendation is summarized below:

1. The limited Hope Creek specific information available is not sufficient to support deferral. While the vibration data obtained and reviewed to date indicates no change in orbital shift, this data is not predictive.

2. The industry data concludes that every BWR 4 that has performed an internal pump inspection has found some degree of cracking. Hope Creek has not maintained RACs temperatures and flows consistent with vendor recommendations or GE assumptions which can adversely affect the crack initiation phenomenon. Therefore, engineering judgement suggests that if we looked right now Hope Creek recirc pump shafts and covers would exhibit cracking. Because the covers are ASME Code Class 1 components with a known defect, we are compelled to inspect them.

3. The Hope Creek procedures require a safety evaluation for this issue prior to the conclusion of RFO 7. Specifically, NC.NA-AS.ZZ-0059(Q), Section 3.2.3 identifies the application of Generic Letter 91-18 to degraded or nonconforming conditions affecting SCCs described in the SAR. G.L. 91-18 defines a nonconforming condition as "operating experience or engineering reviews demonstrate a design inadequacy". Interviews with Byron-Jackson engineering personnel and reviews of the technical alerts and 4th

generation designs for the shaft and covers confirms that shaft and cover cracking is due to design inadequacies caused by the introduction of seal purge. In addition, Hope Creek ran from 1986 to 1992 with high seal purge rates and is subject to this phenomenon.

4. The changing regulatory environment at this time presents a challenge to the assessment of this issue. SECY paper 97-35 and recent proposed rulemaking suggests that a failure to perform the inspection at the first outage past 80,000 operating hours will be viewed as a "de facto modification" requiring a 50.59 safety evaluation. In addition, it will be difficult to credit compensatory actions to offset the change in probability of occurrence of an accident in the SAR. The fact that we were changing the "probability of occurrence of an analyzed accident in the SAR" made it very difficult to justify that we have not created an unreviewed safety question (USQ) by deferring this activity. While it may not be law to date, it is possible that it could be by the end of this year complicating our startup from RFO 7.

5. There is not another outage of sufficient duration planned until RFO 14 at which point reactor recirculation pumps will have over 168,000 hours. With the exception of Vermont Yankee, that value is higher than any other BWR 4 plant that has not inspected to date.

HOPE CREEK GENERATING STATION - SYSTEM ENGINEERING
REACTOR RECIRCULATION PUMP SHAFT & COVER CRACKING

PROBLEM STATEMENT:

SORC has been tasked with reviewing the deferral of the reactor recirculation pump shaft and cover inspections from RFO 7 for any potential safety implications.

FACTS:

The reactor recirculation pump shafts and covers are susceptible to thermal fatigue cracking as described in SIL 459S2, SER 20-86S1, and RICSIL No. 003. General Electric and BW/IP recommend inspection of pump shafts and covers when pumps have greater than 80,000 hours of operation. The Hope Creek reactor recirculation pumps will have 84,000 hours of operation at the beginning of RFO 7 and have not been inspected to date.[1]

Funding has not been authorized to support inspection of the recirculation pump shafts and covers during RFO 7. The capital project to support this work was presented to work scope committee in August 1996 at which time funding was approved to purchase upgraded shafts, covers and seals. Funding was not authorized to support inspection activities or installation of the shaft or cover replacements during RFO 7.

Internal commitments currently exist to perform these internal pump inspections during the ISI 10 year inspection plan.[2] These commitments are contained in the ISI long term plan and in CD-921E & 191F which are being tracked to completion via PRs 960912268 and 961008163. The response to PR 960912268, BPCA 01, did not provide the engineering assessment necessary to support deferral of these commitments from RFO 7. Engineering has not generated the necessary technical justification required to remove the augmented inspections from the ISI Long Term Plan. This is due in part to the fact that there is no means currently available to predict the end of reliable shaft and cover life and no on-line crack detection system has been installed. Industry data shows that the currently installed vibration monitoring equipment will indicate eratic vibration for 1-3 days prior to shaft break. Drywell leak detection (unidentified leakage) would provide indication in the highly unlikely event of RCPB breach through the cover. These would provide indication of imminent failure or notification that failure had occurred, but would not be considered predictive.

HCGS recently identified RACs cooling water temperature, pressure & flow issues which can accelerate thermally induced shaft crack growth.

- The jacket cooler flows to the reactor recirculation pump seals were recently determined to be adjusted higher than specified in the vendor manual.[3] This was discussed with BW/IP (G. Sausman) on 4/14/97 who stated that this condition increases the probability of occurrence of thermally induced fatigue cracks, particularly for "B" pump, whose jacket cooler flowrate is 3 times higher than recommended by BW/IP.

- RACs has been operated outside it's original lower thermal design limits as documented in PRs 951206344 and 950224131, with temperatures reported as low as 35°F; however, the analysis provided by Design Engineering and General Electric[4] did not address the potential impact on shaft cracking concerns.

- RACs demineralizer resins were found contaminated as documented in PRs 950517153 and 951110087. The apparent cause was attributed to a very small leak on "one of the contaminated systems cooled by RACs". The exact source was never identified. The seal jacket coolers and heat exchangers are cooled by RACs, are susceptible to thermally induced fatigue cracking as described in SIL 459 S1, and are a potential source for this contamination.

- RACs relief valve 1EDPSV-2557B was tested and inspected on 4/18/94 and showed evidence of excessive valve wear due to cycling.[5] This valve was rebuilt and returned to service. This relief valve is on the supply line from RACs to the B reactor recirculation pump and could be subjected to higher than normal pressures in the event of a through wall cover crack. However, pressure relief valve cycling could also have been caused by inappropriate filling and venting practices.

INDUSTRY EXPERIENCE

Chinshan 1, a BWR-4, exhibited significant circumferential cracking at 74,000 hours of operation.

52% of the BWR-4 plants have changed out their shafts and covers to address SIL 459.

50% of these BWR-4 plants have reported shaft cracking.

To date, a complete shaft break event has not occurred at a BWR 4.

Most plants that have experienced shaft failures reported they had experienced unusual vibration levels from 1 to 3 days prior to the event. With the exception of two plants (a PWR & BWR 6) that had also experienced hydrostatic bearing failures, all plants were successful at isolating the reactor recirculation pump prior to catastrophic pump failure. It should be noted that a number of plants replaced shafts at the 10 year inservice inspection and although they had no negative vibration information they found circumferentially cracked shafts.

For example, the System Manager at Perry stated that the removed shaft from one of their pumps had circumferential cracks 300 degrees around the shaft, approximately 0.91" deep but displayed no evidence of unusual vibration levels. However, he also stated that both pumps ran normally with 25 mils vibration at the time. Likewise, the Davis Besse plant removed their shafts for inspection in 1989 and observed circumferential shaft cracking in one pump with no abnormal vibration levels noted.

Cover cracking was noted in nearly all covers removed and inspected as described in SIL 459, Supplement 1. However, Dresden 3 was the only plant to report potential through wall leakage from reactor coolant pressure boundary (RCPB) into RACs. Dresden 3 identified potential leakage by disconnecting the CCW, pressurizing the recirc pump and observing a very small amount of leakage from the drilled hole heat exchanger. Repeat pressurization did not confirm initial leakage observations. CCW was isolated from the drilled hole heat exchanger, the drilled hole heat exchanger outlet was piped to the identified leakage sump and the pump was returned to service. (It should be noted that this abnormal cooling line up to the pump seal could have resulted in seal failure if CRD had been lost.)

Impeller bolting failures have been limited to BWR 6 and PWRs. This phenomenon has not been observed to date in BWR 3, 4, or 5's, possibly due to the differences in bolting arrangements.

BACKGROUND:

The initiation of shaft cracks is due to the mixing of cold mechanical seal injection water with hot system water near the bottom of the thermal barrier of the reactor recirculation pump shafts. In this turbulent mixing region the material surfaces are exposed to rapidly varying water temperatures which cause alternating stresses. The frequency of this dynamic action can vary from about 1 Hz to 25 Hz. Therefore, a large number of stress cycles can occur in a relatively short operating time and fatigue cracking of the material surface can result.[6]

General Electric performed detailed destructive examination of the KKL-Leibstadt (BWR 6) shaft and heat exchanger. Examination of cracked surfaces of both shaft and heat exchanger by scanning electron microscopy showed fatigue striation spacings indicative of relatively large magnitude, low frequency stress cycles on both the shaft and the heat exchanger. Because there are essentially no mechanical loads on the heat exchanger it was concluded that thermal loading alone, possibly similar to that suspected at La Salle 1, produced the bulk of crack propagation observed in both shaft and heat exchanger at KKL.[7]

Metallography performed at Liebstadt, Grand Gulf, Chinshan and La Salle indicated that deeper circumferential cracks did not arrest and stated that shaft cracking must be "assumed to progress to eventual shaft failure". Their conclusions also supported the observations that crack depths are generally larger with increased pump service hours.[8]

BW/IP provided updated plant data on shaft performance at a meeting with PSE&G on 4/10/97. This data lists 23 - BWR 4 plants. Of those BWR - 4 plants, 12 have replaced shafts and covers (52%). Of the 12 plants that replaced parts, 6 reported cracks (50%). The remaining 6 plants either did not inspect for cracking or did not report their findings to BW/IP.

The Hope Creek initial response to this issue (10/87) documented in CD-921E, states: "there is no immediate need to inspect the pump shaft and cover for thermal cracks. This type of cracking requires a number of years to initiate and propagate (>20,000 hours). Plant Hatch inspected after approximately 42,000 hours of operation and found some shaft cracks which they machined out. Cracks propagate axially down the shaft from the thermal mixing region until they reach the upper slide ring to shaft weld. At this point they propagate circumferentially at a faster rate since they are driven by the pump itself.

The shaft to impeller bolting arrangement concern at Crystal River 3 is different than Hope Creek's arrangement. Crystal River had a multiple bolting arrangement and Hope Creek has a single bolt/lock key arrangement. Therefore, the impeller to shaft bolt failure concerns attributed to the multiple bolting arrangement are not directly applicable to Hope Creek. This is discussed in RICSIL No. 003."[9]

A BWROG Meeting Summary dated 9/26/90 indicates that several BWR pumps had been disassembled and inspected and all observed thermal fatigue cracking. In addition, the cracks were deeper than originally projected (0.3 inches) which was likely caused by low frequency fluctuations or mechanical loads. The failure impact was identified to be as follows: cover cracking could result in penetrating CCW (RACs) and shaft cracking would result in reduced structural margin, less remaining life and eventual shaft failure.

Three conditions exist which complicate Hope Creek's assessment of this issue:
First, the cooling water flowrate to the seal heat exchangers (jackets coolers) is set higher than specified in the vendor manual.[10] For "A" pump (1A-P-201) reactor auxiliary cooling water flowrate is set at 17 gpm (1.4 times recommended) and for "B" pump (1B-P-201) the flowrate is 35 gpm (3 times recommended). This condition was discussed with George Sausman, BW/IP, on 4/10/97 and 4/14/97 at which time he indicated that higher differential temperatures in the thermal barrier region would tend to initiate more cracks initially and could accelerate crack growth.

The second concern is that the RACs Demineralizers have been reported contaminated. The reactor recirculation pump seal coolers are a potential source for this radioactive contamination.

The third concern involves the relief valves on the seal cooler RACs supply line. The 1EDPSV-2557B was inspected via WO 921001327 completed 4/18/94 and exhibited extensive internal valve damage indicative of excessive cycling. The 1EDPSV-2557A is scheduled for inspection and testing during RFO 7. Excessive pressure on this line could be indicative of a breach of the reactor coolant pressure boundary as would be expected if indications have developed in the recirculation pump cover cooler.

REGULATORY ISSUES:

Operating experience reports have been issued since 1984 identifying potential design deficiencies in Byron Jackson reactor coolant and reactor recirculation pump shafts, covers, and impeller bolting. These design deficiencies were said to have resulted from the introduction of seal purge systems which introduced thermal stresses in the thermal barrier region and beyond, depending on the seal purge flow rate. The result of thermal cyclic stresses are axial "crazing" cracks exhibiting a turtle back pattern in the thermal barrier region of the shafts.

GE Sil 459 recommends "scheduled inspections of pumps with greater than 80,000 hours of pump operation to detect cracking and to implement corrective actions before excessive crack growth can occur". Hope Creek reactor recirculation pumps will have 84,000 operating hours at the commencement of RFO 7 in September 1997.

Two 10CFR Part 21's have been issued on shaft and cover cracking issues to date and one ISI exemption request has been submitted to NRC in this regard. Details of these issues are being acquired by Nuclear Licensing.

NC.NA-AS.ZZ-0059(Q), Rev. 0, Section 3.2.3, discusses the requirements for application of Generic Letter 91-18 to degraded or nonconforming conditions affecting SSCs described in the SAR. It states that "delay or partial completion of scheduled corrective actions should be treated as a change and should be reviewed for 50.59 applicability."

Generic Letter 91-18, Section 2.4, defines a nonconforming condition as follows:
"A condition of an SSC in which there is a failure to meet requirements of licensee commitments. Some examples of nonconforming conditions include the following:

3. Operating experience or engineering reviews demonstrate a design inadequacy."

The following operational transients or design basis events described in the HCGS UFSAR are considered applicable to the issues described in SIL 459 and supplements:

- 15.3.1 "Reactor Recirculation Pump Trip"
- 15.3.3 "Reactor Recirculation Pump Shaft Seizure"
- 15.3.4 "Reactor Recirculation Pump Shaft Break"
- 15.6.5 "Loss-of-Coolant Accident Resulting from the Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary Inside Primary Containment".
- 9.2.1 "Station Service Water System" (for RACS contamination)
- 9.2.8 "Reactor Auxiliaries Cooling System".(for RACS contamination)

HOPE CREEK GENERATING STATION - SYSTEM ENGINEERING
REACTOR RECIRCULATION PUMP SHAFT & COVER CRACKING

SHAFT CRACKING:

HCGS has the Byron-Jackson reactor recirculation pumps with seal purge systems, the source of which is Control Rod Drive (CRD). Elevated seal purge flow rates were used at HCGS from 1986 through 1992. Therefore, crack initiation in the form of axial cracks (crazing) should be assumed to be present in the Hope Creek reactor recirculation pumps.

Cracks must propagate and change orientation from axial to circumferential in order for catastrophic pump shaft failure to occur. In a report dated April 12, 1992, BW/IP concluded that "thermal cracks can reach circumferential orientation and in this configuration can be driven by mechanical loads". A review of industry information suggests that pump vibration levels will increase anywhere from 1 to 3 days prior to shaft failure.

Reactor recirculation pump trips and failures are described in the Hope Creek Updated Final Safety Analysis Report (UFSAR) Chapters 15.3.1, 15.3.3, and 15.3.4.

COVER CRACKING & HEAT EXCHANGER LEAKAGE:

The concern described in SIL 459 S1 involves a possible failure of the drilled hole heat exchanger at Dresden 3. Increased levels in the closed cooling water (CCW) surge tank were noted. Investigations conducted by plant personnel indicated the in-leakage source was from inside the drywell. Although cracking had been observed in other recirculation pump covers during in-service inspections at other GE BWRs, this is the first occurrence at any GE BWR of possible leakage through the pressure boundary between the reactor coolant and the CCW flow passages."

Problem reports 950517153, 951110087, and 960512126 document a current condition at Hope Creek where contamination is present in the RACs demineralizer resins. The source of leakage was not identified but was reported to be "a very small heat exchanger leak". Components with potential to leak into RACS are: RWCU pump cooler, RWCU non-regen heat exchanger, recirc pump shell and tube and drilled hole heat exchangers.

Contamination of RACS by primary coolant leakage into the pump heat exchanger is addressed by UFSAR Section 9.2.1 "Station Service Water System" and 9.2.8 "Reactor Auxiliaries Cooling System".

IMPELLER BOLTING

Degraded impeller-to-shaft bolting was first identified in INPO SER 89-84 which concerns "extensive pump damage in an event at Palisades on September 16, 1984". INPO SER 20-86R2 followed and discussed the reactor coolant pump failure at Crystal River 3 and identified design inadequacies related to the impeller bolting in addition to the shaft and cover cracking concerns. The HCGS response to SER 20-86 (CD-921E) stated that visual /LP inspections would be performed during the 10 year ISI as augmented examinations.

Hope Creek recirc pump impellers are attached to the shafts by a single left hand thread impeller nut (part 3-2). The nut is locked to the shaft by a lock screw (part 3-3, required torque 150 ft.-lbs.). The screw is retained by a lock pin in a drilled hole at the interface of the lock screw and the impeller nut. The lock pin is covered with a plug weld.

A crack in the plug weld will not allow the screw to loosen. The plug weld would require complete severance from the lock pin hole and then the lock pin would have to fall out for there to be a potential for the lock screw to loosen. In addition, the left hand impeller nut threads in combination with the counter-clockwise rotation of the pump would prevent loosening of the impeller nut.

The recirculation pump mechanical failures are discussed in UFSAR Chapter 15.3.3 & 15.3.4. Impeller bolting degradation could lead to catastrophic pump failure as described in SERs 20-86 and 89-84. However, it should be noted that both Palisades and Crystal River reactor coolant pumps had multiple bolting arrangements which differs in design from the single impeller bolt arrangement on the reactor recirculation pumps at Hope Creek. Therefore, this potential deficiency is not directly applicable to Hope Creek Generating Station.

ASME CODE ISSUES:

The reactor recirculation pump shaft is not an ASME component and is not required to be inspected by ASME Section XI. The pump cover is an ASME Code Class 1 component and is subject to ASME Section XI requirements.

"Structural integrity" is not defined in the Tech Specs, 10CFR50, ASME Section III or ASME Section XI The term "structural integrity" as used in Tech Spec 3/4.4.8 implies that applicable code components meet the requirements of ASME Section XI. The reasoning is as follows: LCO 3.4.8 states, "The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.8."

Specification 4.4.8 states, "No requirements other than Specification 4.0.5."

Specification 4.0.5 requires inservice inspection and testing be performed in accordance with the applicable year and Addenda of ASME Section XI.

The Hope Creek ISI Long Term Plan complies with the 1983 edition, 1983 summer addenda of ASME Section XI. This edition/addenda requires recirculation pump casing inspection once per inspection interval. The current inspection interval ends at the completion of RFO7. The recirc pump casing has not yet been inspected, and in accordance with the frequency of inspection in the 1983 Code, the pump casing would be inspected in RFO7.

10CFR50a(g)(4)(iv) allows the use of portions of later editions or addenda that have been approved by the NRC and identified in 10CFR50.55a(b). The 1989 ASME Code Section XI requirements for examination of the recirc pump have been incorporated into the Hope Creek ISI Program. The resulting inspection requirement is that pump inspection is required only when the pump is disassembled for maintenance, repair, or volumetric examination. If these activities are not required for RFO7, then the pump will not require inspection.

The recirculation pump shaft is not required to be inspected in accordance with ASME Section XI. The pump cover, an ASME Class 1 pressure retaining part, would be inspected as part of the pump internal inspection.

This inspection, in accordance with the 1989 Code, is required once per inspection cycle if the pump is opened for performance of other maintenance, i.e. the Code no longer requires the pump inspection to be performed at a specific interval.

Since the ASME Section XI inspection is not currently scheduled to be performed, a determination of structural integrity does not appear to be explicitly required.

GE SIL 459S1 identifies that cracks generated in the pump cover have the potential for breaching the pump internal pressure boundary between reactor coolant and RACS, causing leakage of reactor coolant into RACS.

A possible scenario where structural integrity would require evaluation would be where cover cracks penetrate the drilled hole cooler in the pump cover. This would lead to RACS contamination from the higher pressure reactor coolant entering the drilled hole(s) and therefore RACS, through the crack(s). If investigation revealed that the source of RACS contamination was the recirculation pump(s) drilled hole cooler, structural integrity would not be maintained by virtue of the crack penetrating the interface between the ASME Class 1 reactor coolant pressure boundary and the ANSI B31.1 RACS. The action of Tech Spec 3/4.4.8 would have to be followed in this case.

OPERATONS IMPACT:

The Operations Department procedures concerning abberrations of the reactor recirculation pumps were reviewed to assess the potential impact of SIL 459. Some changes should be considered and training should be provided to highten operator awareness to the change in probability associated with the reactor recirculation shaft and cover cracking phenomenon.

SHAFT CRACKING ISSUE:

The abnormal operating procedure for recirculation pump trip (HC.OP-AB.ZZ-0112(Q) δ5.4) states that "the instantaneous recircularion pump shaft seizure or break are **extremely unlikely events.** If either were to occur, the recirculation drive flow of the affected loop will decrease rapidly and can result in a high vessel level (L8) trip of the Main Turbine and Feedwater Turbines. Subsequently, a reactor scram occure from the trip of the Main Turbine."

There is no reference to SIL 459 or associated Cds.

The normal system operating procedure for reactor recirculation (HC.OP-SO.BB-0002(Q), Caution 5.2.6.d) states that "neither recirculation pump **should** be operated above the danger setpoint. These setpoints are as follows:

	Alert	Danger
Recirc Pump A	10.0 mils	10.5 mils
Recirc Pump B	12.5 mils	13.5 mils

There is no reference to SIL 459 or associated Cds.

The alarm response procedure for reactor recirculation pump high vibration (HC.OP-AR.ZZ-0008(Q), Attachment E4, pages 127 and 128) contains the following operator actions:

1. Monitor vibration (CRIDS Points A2601/A2602/A2603/A2604)
2. REDUCE reactor recirculation pump speed in an attempt to reduce vibration.
3. Do not operate pumps above the danger vibration setpoint.
4. If vibration continues to increase, REDUCE pump speed to minimum and trip the pump as directed by the NSS.

This procedure should contain guidance to address vibration aberrations other than strictly increasing data. In addition, it should provide some guidance regarding decreasing pump speed and decreasing vibration such that the operator must seek vibration assistance from the vibration engineer or Maplewood labs prior to restoring recirc pump speed to avoid cycling a pump with a potentially cracked shaft.

COVER CRACKING ISSUE:

The operations procedure for RACs system malfunction (HC.OP-AB-ZZ-0123(Q) δ4.11, contains the following guidance:

If a RACs head tank high level alarm or RM-11 Liquid Process Monitor Alarm is received, then perform the following:

1. Ensure makeup valve SV-2616 I closed
2. Notify Chemistry to sample RACS & SSW to identify source of inleakage
3. IF possible, identify source of inleakage, Secure affected equipment and isolate RACS to that component.
4. If entry was RM-11, check for detector failure and/or power failure.
5. If radioactive inleakage is occurring and can not be located and isolated, then isolate RACS from the rest of SSW..."

This procedure contains no reference to SIL 459 or associated Cds. This procedure is not consistent with the UFSAR which states that RACS is isolated upon receipt of the RACs radiation monitor alarm. This CAQ is documented in PR 960809087. CRCA #1 is due on 6/1/97 to address this issue.

An increase in unidentified leakage would be expected in response to a complete breach of the recirculation pump cover/casing . This would be observed as increased leakage into the drywell floor drain system. Operations procedure HC.OP-GP.ZZ-0005(Q) needs some improvement to address reactor recirculation pump *cooler* leak detection. The procedure considers motor cooler issues, but not specifically the pump jackets and heat exchangers which come off another RACs supply line.
In addition, two prerequisites must be satisfied to implement this procedure which has been a point of contention between System Engineering and Operations in the recent past. First, the trended drywell leakage data must show it is increasing above a *baseline leakage rate*. Second, the Operations manager must authorize it's implementation. One possible suggestion is for a baseline leakage rate band to be provided to Operations each cycle for floor drain and equipment drain leakage to preclude uncertainties and delays in implementing this procedure, particularly in light of SIL 459 concerns.

OPTIONS:

1. Perform the shaft, cover & impeller bolt inspections on both pumps during RFO 7 in accordance with existing commitments and the ISI long term plan.

Challenges:

The capital project must be reinstated & DCP prepared
Funding must be pursued with SERB, PRC & WSC (~5 million)
Planning must be accelerated to support Drywell disassembly
Special tooling must be purchased and delivered
Motor refurbishment should be reconsidered

Benefits:

Zero risk assûmed.
Perception of responsiveness to industry information.

2. Perform the shaft, cover and impeller bolt inspection on only 1B-P-201 during RFO 7.

Challenges:
The capital project must be reinstated and DCP prepared.
Funding must be pursued with SERB, PRC & WSC (~5 million)
Planning must be considered to support equipment hatch disassembly
Special tooling must be purchased and delivered
Motor refurbishment should be reconsidered
Perform engineering analyses to support this option
Develop contingency plan for 1A-P-201 including improved vibration monitoring system
Safety evaluation required after RFO 7 to satisfy Generic Letter 91-18

Benefits:
Supports keeping 1A-P-201 available for full core offload/decay heat removal
The inspection of 1B-P-201 may support taking no action on 1A-P-201.
The capital project cost should be reduced.
The Drywell Personnel Airlock won't need disassembly avoiding historical assembly and LLRT issues

3. Perform no internal reactor recirc pump inspections during RFO 7.

Challenges:
Increased probability of shaft/cover failure event
Increased probability must be offset by compensatory actions that "far outweigh"
Resources required to install transient vibration monitoring system
Potential perception of a lack of responsiveness to long standing industry issues
Resources required to verify adequacy of existing operations procedures
Safety evaluation required after RFO 7 to satisfy Generic Letter 91-18

Benefits:
Improves outage cost and schedule
Supports keeping 1A-P-201 & 1B-P-201 available which supports alternate decay heat removal strategy for full core offload.

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[1] Memo from P. Steinhauer to A. Fakhar dated March 27, 1997 - SPE 97-052

[2] Memo from R. Brandt to P. Walzer dated June 28, 1988 - HMT-88-0025

[3] Reference PR 970306232

[4] Reference PN1-B31-C001-0265

[5] Reference WO 921001327

[6] BW/IP Technical Service Bulletin 8701-80-005 dated February 6, 1987

[7] System Interaction Study - Reactor Recirculation Pump Shaft Cracking dated April 25, 1989, OG9-423-55

[8] Memo to BWROG Committee Members dated 9/14/90 OG90-786-55,56

[9] CD 921E dated 11/19/87

[10] Problem Report 970306232 CREV 1 disposition dated 4/4/97

HOPE CREEK GENERATING STATION
SYSTEM ENGINEERING

BWR 4 REACTOR RECIRCULATION PUMP
SIL 459 PLANT DATA

Plant	GMWe	Ops Hours To Date or when Replaced (x 1000)	Shaft Replaced	Cover Replaced	Cracks	Inspected
Browns Ferry 1	1098	58	N	N		
Browns Ferry 2	1098	75	Y	Y		
Browns Ferry 3	1098	58	N	N		
Brunswick 1	849	116	N	N		
Brunswick 2	849	122	N	N		
Chinshan 1	636	74	Y	Y	Y	Y
Chinshan 2	636	70	Y	Y	Y	Y
Cooper 1	800	114	Y	Y		
Duane Arnold	563	96	Y	Y	Y	Y
Fermi 2	1154	69	N	N		
Fitzpatrick	850	138	N	N		
Hatch 1	813	86	Y	Y		N
Hatch 2	822	80	Y	Y	Y	Y
Hope Creek 1	1118	80	N	N		N
Limerick 1	1092	62	N	N		N
Limerick 2	1092	60	N	N		N
Muhleberg		107	Y	Y	Y	Y
Peach Bottom 2	1098	92	Y	N	Y	Y
Peach Bottom 3	1098	100	N	N		
Susquehanna 1	1085	101	N	N		N
Susquehanna 2	1085	94	N	N		N
Vermont Yankee	537	178	N	N		