596 pgs total

Exelon Nuclear

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RS-05-005

January 5, 2005

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Dresden Nuclear Power Station, Units 2 and 3
Renewed Facility Operating License Nos. DPR-19 and DPR-25
NRC Docket Nos. 50-237 and 50-249

Quad Cities Nuclear Power Station, Units 1 and 2
Renewed Facility Operating License Nos. DPR-29 and DPR-30
NRC Docket Nos. 50-254 and 50-265

Subject: Submittal of Information Regarding Extended Power Uprate Vulnerability Reviews

Reference: Letter from K. R. Jury (Exelon Generation Company, LLC) to U. S. NRC, "Commitments and Plans Related to Extended Power Uprate Operation," dated May 12, 2004

Following implementation of extended power uprate (EPU) at Dresden Nuclear Power Station (DNPS) and Quad Cities Nuclear Power Station (QCNPS), increased steam and feedwater (FW) flows associated with the higher power levels resulted in increased main steam line (MSL) component vibrations. Subsequently, these increased vibrations produced equipment problems, mainly associated with accelerated component wear, that were determined to be a result of operation at EPU. Exelon Generation Company, LLC (EGC) performed assessments to ensure that vulnerable components were identified, monitored, and modified as necessary to support acceptable performance for a complete operating cycle at full EPU power levels, and to ensure that no potentially affected components experience failure or unacceptable degradation due to vibration.

In addition to the vibration assessment, EGC conducted an extent of condition (EOC) review for EPU-related issues. The objectives of this review were to identify and mitigate the potential vulnerabilities associated with operating the DNPS and QCNPS units at EPU levels, to fully understand the differences between the industry norms for uprated plants and the design and operating margins for DNPS and QCNPS, to development and implement recommended corrective actions, and to reevaluate EPU as a safe and reliable operating strategy for DNPS and QCNPS.

In the referenced letter, EGC committed to meet with the NRC technical staff to: (a) discuss results of the reevaluation of previous assessments of the impact of flow-induced vibration under EPU conditions on reactor vessel internals, steam and FW systems and components, including an evaluation of previous evaluation deficiencies; (b) describe how the data collected

was used to assess the dynamic loading on plant components other than the steam dryer, and (c) provide results of the review to identify potential EPU-related equipment vulnerabilities. On September 23 and 24, 2004, EGC met with the NRC technical staff to discuss the results of our reevaluations. Following the meeting, the NRC requested that EGC provide details regarding information that was discussed. The requested information is provided in the attachments to this letter.

Attachment 1 provides the requested Engineering Change Evaluations performed for EPU-related issues at DNPS, Units 2 and 3, and QCNPS, Units 1 and 2. Components included in the vibration assessments for DNPS Unit 3, and QCNSP Units 1 and 2, are tabulated in the ECs. However, because DNPS Unit 2 operated continuously during periods when the other units were evaluated, a qualitative assessment of DNPS Unit 2 components was performed. Attachment 2 contains a comprehensive report for the EOC review, including scope and depth of the evaluation, a list of individuals involved in the review, review methodology/approach, findings, and associated recommendations and corrective actions. In addition, this attachment provides EOC reports compiled for main steam, FW, reactor internals, and recirculation systems. Attachment 3 contains tables that outline specific recommendations identified during the EOC review, including modifications and enhanced inspections, strategic operational and maintenance initiatives, and ongoing analyses, studies, and reviews to address EPU-related system vulnerabilities. Attachment 4 provides a root cause report and operability evaluation for the Target Rock Safety/Relief Valve issues at QCNPS. The root cause report is currently being revised to reflect the latest available information. Attachment 5 contains the root cause and operability evaluation related to FW sample probes at DNPS.

One (1) CD-ROM is included in this submittal. The CD-ROM labeled, "Dresden and Quad Cities EPU Reviews," contains the following file:

001	September NRC Meeting.pdf	9,985,893 bytes	Publicly Available

Should you have any questions concerning this letter, please contact Mr. Thomas G. Roddey at (630) 657-2811.

Respectfully,

Patrick R. Simpson

Patrick R. Simpson
Manager – Licensing

Enclosure:
(1) CD-ROM, "Dresden and Quad Cities EPU Reviews"

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Dresden Nuclear Power Station
NRC Senior Resident Inspector – Quad Cities Nuclear Power Station
Illinois Emergency Management Agency – Division of Nuclear Safety

ATTACHMENT 1

Engineering Change Evaluations – Vibration Assessments for Dresden and Quad Cities Nuclear Power Stations

1A) Engineering Change Evaluation 348693, Revision 1

1B) Engineering Change Evaluation 346515, Revisions 0, 1, 2 and 3

1C) Engineering Change Evaluation 348316, Revision 2

1D) Engineering Change Evaluation 347001, Revision 1

1E) Engineering Change Evaluation 346402

ATTACHMENT 1A

Engineering Change Evaluation 348693, Revision 1
"Evaluation (EVAL) of Dresden Unit 2 for Potential Vibration Effects at Extended Power Uprate (EPU) Power Levels"

Engineering Change (EC) 348693, Revision 1
Evaluation (EVAL) of Dresden Unit 2 for Potential Vibration Effects at Extended Power Uprate (EPU) Power Levels

Reason for Evaluation / Scope:

Background

During the Quad Cities Unit 1 forced outage in November 2003 (i.e., Q1F51), the 3B Electromatic Relief Valve (ERV) actuator was discovered with significant damage. ERV actuators were refurbished and vibration instrumentation was added to the ERVs and other Main Steam Line (MSL) components. Quad Cities EC-EVAL 346515, "Evaluation of Quad Cities Unit 1 Main Steam Line (MSL) Vibrations at Extended Power Uprate (EPU) Power Levels," concluded that the ERV actuators were vulnerable to accelerated aging due to the increased vibration levels from EPU. Therefore, it was necessary to determine the extent of condition of the observed indications. The instrumentation of Dresden Unit 3 components, EC EVAL 347006, "Evaluation of Dresden Unit 3 Main Steam Line (MSL) Vibrations at Extended Power Uprate (EPU) Power Levels," and this evaluation are part of the extent of condition review from the Quad Cities issues.

Purpose

This EC EVAL provides a qualitative technical evaluation to support continuous operation of Dresden Unit 2 up to a power level of 2957 megawatts thermal (MWth). This evaluation confirms the conclusions reached for component acceptance after work performed during the Fall 2003 Dresden Unit 2 refueling outage (i.e., D2R18), prior to the discovery of the Electromatic Relief Valve (ERV) degradation on Quad Cities Unit 1.

Approach

Component and system responses for full EPU power operation were assessed using vibration data taken during the EPU power ascension in December 2001, component failure history, laboratory testing, inspection results from preventive maintenance (PM) and analytical modeling. Unit 2 work orders and other PM information are compared to the conditions found at Quad Cities. Vibration data is compared to data measured at Quad Cities Unit 1 during original EPU power ascension. This allows comparison of data taken at similar locations. The results of the extensive evaluations for Quad Cities Unit 1 (i.e., EC EVAL 346515) are then used for justification of the acceptability of the Dresden Unit 2 components. The purpose of this evaluation is to provide assurance that full EPU power operation will pose no threat to continued equipment operation throughout a 24-month fuel cycle.

Results

All components are found to be acceptable for full EPU power operation. Required actions are included to ensure acceptable equipment performance beyond the current cycle, which culminates in the Dresden Unit 2 refuel outage scheduled for November 2005. Actions are tracked under AR 203507.

Engineering Change (EC) 348693, Revision 1
Evaluation (EVAL) of Dresden Unit 2 for Potential Vibration Effects at Extended Power Uprate (EPU) Power Levels

Required Actions:

Prior to D2R19

1. Evaluate the current PM scope and frequency for all components included in this evaluation. The expectation is that some MSL components will need increased PM frequencies for inspections, rebuilds, refurbishments and/or replacements until sufficient EPU operational experience proves that accelerated aging is not occurring. (Assignee: A8330NESTP, due date November 21, 2004)
2. Replace current ERV actuator Catalog Identifications for the springs, bracket with bushing and guide rods with the upgraded parts as utilized at Quad Cities prior to rebuilding any additional ERV actuators. (Assignee: A8322MM, due June 30, 2004)
3. Create a standard PM task for walkdown of potentially vulnerable components. For scope, see EC EVAL 346402 (walkdowns performed for Unit 3). (Assignee: A8330NESTP, due January 15, 2005)

During D2R19 Outage

4. Inspect the components internal to the ERV actuators. (Assignee: A8351NESPR, due date December 15, 2005)
5. Inspect one main steam isolation valve (MSIV) during the next refuel outage, D2R19, to confirm that no degradation has occurred, especially in the area of disk to stem. If degradation is found, evaluate the need for expanding inspection to other MSIVs. (D2R19– Assignee: A8351NESPR, due date December 15, 2005)

Detailed Evaluation:

Comparison of Vibration Values

Dresden Unit 2 does not currently have vibration monitoring equipment installed on specific components like Dresden Unit 3 and the Quad Cities units. Therefore, the original EPU power ascension data was reviewed for this evaluation. A comparison of the magnitudes and frequencies was performed for three units (Attachment 1). This comparison shows that there is no conclusive evidence that Dresden Unit 2 exhibits any difference in vibration response from the other two units. One reading in excess of 1.0 g maximum is considered as non-consequential due to the high frequency at which it occurs (150 Hz) and the small displacements that would occur. The high frequency, high g value (>1.0) responses were not found to have any negative impact on the evaluated components for the other three units. The comparison concluded that the evaluations performed for Dresden Unit 3 and Quad Cities Unit 1 are applicable to Dresden Unit 2. Additional comparisons are discussed for critical components below and are based principally on the as-found conditions of the components as identified during D2R18. All other components are considered acceptable since evaluations for the other units identified no vulnerabilities. These other

evaluations are documented under EC EVAL 347006 for Dresden Unit 3 and EC EVAL 346515 for Quad Cities Unit 1

As-found Component Condition Summary (Dresden Unit 2)

Work orders for inspection and refurbishment of the four Unit 2 ERVs were reviewed. Step 4.6 contains specific steps to inspect and replace, as necessary, the plunger springs, guide rods, and bushings (on support brackets). All four Unit 2 actuator internals were found in acceptable condition with no parts requiring replacement. Since these parts are the components of concern, and since Unit 2 had operated at full EPU power (except for normal down power evolutions) for over 700 days, degradation effects due to EPU are deemed negligible. It is recommended that the upgraded parts be utilized during any future rebuilds of the actuators. This will be accomplished by retiring the existing part Catalog Identifications and replacing them with the revised parts, as was done for Quad Cities.

Acceptability of ERV Component Operation at EPU Levels

The four ERVs have virtually identical assemblies, which consist of the main ERV valve body, pilot valve, and solenoid actuator. The pilot valve is connected to the ERV by means of a turnbuckle (a threaded pipe coupling arrangement) and a pilot valve tube. Each valve has small diameter leak off piping that is routed back to the ERV discharge line.

Significant testing of an ERV assembly has been completed at Wyle Laboratories. Although the focus of this testing was to validate the modifications for Quad Cities, they are directly applicable to Dresden. Documentation of the testing and results is included in the documents related to EC (DCP) 343933, calculation SIR-04-023. The results of the testing, which confirmed the vulnerable internal actuator components, and inspections of those components performed during D2R18, are that the ERVs in their current configuration are adequate for full cycle operation. It is recommended that during any subsequent actuator rebuilds, the hardened parts be utilized for the Dresden actuators.

Evaluation of MSIVs

An evaluation of the MSIVs was performed for Quad Cities and Dresden Unit 3 by comparing measured values to seismic aging qualification test data for the Quad Cities Unit 1 actuators. Measured and extrapolated values for MSIVs were well bounded by this aging evaluation and were therefore determined to be acceptable for continuous full EPU power operation. Since the EPU vibration data is similar for Dresden and Quad Cities, the evaluations already performed are considered bounding for Dresden 2. In addition, as discussed below, the as-found condition of the MSIVs, after more than 700 days of EPU operation, showed no deleterious effects due to vibration. Therefore, there are no concerns with the acceptability of the MSIVs for EPU power operation. Continued

inspections during the Dresden Unit 2 refueling outage scheduled for Fall 2005 (i.e., D2R19), and when warranted due to leak rate failures, will ensure that any degradation is found early and prompt corrective actions taken.

The work performed during D2R18 on the MSIVs was reviewed for possible effects of vibration. That review and the work performed is summarized below as reported by the responsible plant engineer.

- During D2R18 the following work was performed. No indications were noted on the MSIVs or any of the sub-components that would be indicative of EPU related vibration degradation. Any issues noted on the MSIVs are considered normal.

  1. The MSIV liners were proactively replaced on the 1A, 1B, 1C and 2C MSIVs. No indications of abnormal degradation on the valve, liners or plug.
  2. During D2R18 the total MSIV minimum path leakage was determined to be 94.25 standard cubic feet per hour (scfh) (A MSL – 0.15 scfh, B MSL – 1.1 scfh, C MSL – 57.1 scfh and D MSL 35.9 scfh). This value is in excess of the Technical Specifications (TS) allowable value of 46 SCFH as required by TS Surveillance Requirement 3.6.1.3.10. As a result, the 1D and 2D were also repaired. The cause of the excessive leakage was determined to be normal wear.
  3. During disassembly of the 1D MSIV, the four lower liner welds were identified cracked. This valve still had the original liner configuration. This is an issue that was not unexpected, and a contingency modification with the new liner design was installed. The 2D lower liner welds were intact. Other than the cracked lower liner welds, there were no indications of abnormal degradation on the valve, liners or plug.
  4. The air supply manifolds were replaced on the four inboard MSIVs per the normally scheduled PM. No abnormal degradation was noted.
  5. The 10% close limit switches were adjusted on all MSIVs.
  6. A leak test was performed on the inboard MSIV actuators. Two were noted to have minor leakage, and were replaced.
  7. A non-destructive examination was performed on the inboard air supply line connections to the accumulators both at the beginning of the outage to evaluate the as-found condition, and at the end of the outage to verify that no damage had been induced as a result of maintenance. No indications were noted. In addition, snoop checks were performed at the supply line connection to the accumulator and to the manifold for the outboard valves. No abnormal degradation was noted.

Evaluation of Target Rock Safety/Relief Valves (S/RVs)

The Target Rock S/RV as found setpoint after the first EPU cycle was found low by 3.6%. Evaluation of the impact of this value is documented under IR #225880. Investigation and action, both completed and ongoing, documented

under AR 215874 for the quad Cities valve will also provide documentation for the Dresden units of the final approved configuration. Final resolution will be documented under EC EVAL 350693. No actions are specifically required under this EC for the target rock valve.

Other MS Components

The following MS supports were inspected during D2R18 and found acceptable.

Support Number
M-569 SHT 21
M-564G SHT 2
M-564G SHT 3
M-569 SHT 9

The following MS snubbers were tested during D2R18 and found acceptable.

M-564G Sht 2, M-564G Sht 3, M-564E Sht 10

Summary of Other MS Work performed during D2R18

Following the issues discovered at Quad Cities and the decision to perform a detailed inspection of Dresden Unit 3 during Maintenance Outage 10 (i.e., D3M10), concerns were raised about Dresden Unit 2. Dresden Unit 2 began EPU operations near the end of December 2001. Up to the start of the D2R18 outage, Dresden Unit 2 had operated nearly 2 full years. During the D2R18 outage, the System Manager performed routine inspections of the MS system. Also the standard PMs and program inspections were performed. No adverse conditions were identified that were indicative of abnormal wear that resulting from increased vibrations as seen at Quad Cities. The Electrical Maintenance Department first line supervisor that has had experience at Quad Cities on overhauling their ERV actuators stated that the D2 ERV actuator overhauls performed during D2R18 indicated only normal wear. The four main steam safety valves that were removed all passed their +/-1% setpoint test at NWS Technologies.

Conclusions / Findings:

This EC EVAL provides an engineering evaluation supporting operation up to 2957 MWth. It has been concluded that this full EPU power operation will not result in imminent failure or unacceptable degradation levels of any components. The conclusion is based on evaluation of the empirical data, including operational and work history data, measured vibration data, endurance test reports, comparison to Dresden Unit 3 and Quad Cities Units 1 and 2 evaluations, and ERV testing.

Engineering Change (EC) 348693, Revision 1
Evaluation (EVAL) of Dresden Unit 2 for Potential Vibration Effects at Extended Power Uprate (EPU) Power Levels

Attachments:

Attachment 1: Chart of EPU MSL Vibration Levels – Dresden Unit 2 versus Dresden Unit 3 and Quad Cities Unit 1.

References:

1. EC 347006, Evaluation for Dresden Unit 3
2. EC 346515, Evaluation for Quad Cities Unit 1
3. TODI CC2004-9993, Main Steam Line Post EPU Material Condition Documentation From D2R18

Independent Review:

MPR Associates, Incorporated, has performed an independent review of this evaluation and they have concurred with the conclusions. Their letter is attached below.

MPR
ASSOCIATES INC.
ENGINEERS

April 22, 2004

Ms. Sharon Eldridge
Lead Project Engineer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

Subject: Review of Vibration Assessment for Dresden Unit 2 (EC-EVAL 348693)

Dear Ms. Eldridge:

Exelon recently provided MPR with draft EC-EVAL #348693, "Evaluation of Dresden Unit 2 For Potential Vibration Effects at EPU Power Levels," for review. The EC-EVAL examined the potential for damage to occur to main steam line components as a result of higher vibration levels during operation at extended power uprate (EPU) power levels. MPR has completed a review of the draft EC-EVAL and concluded that the Dresden Unit 2 emergency relief valves (ERVs), main steam isolation valves (MSIVs), and Target Rock safety relief valves are unlikely to experience accelerated vibration related aging effects prior to the next refueling outage (D2R19). These components were identified as being particularly susceptible to damage based on inspections of main steam line components, especially ERVs, at Quad Cities Unit 1 (QC1) in November 2003. Other Dresden Unit 2 components susceptible to elevated vibration damage that were inspected recently also should maintain their integrity through D2R19. In summary, MPR concurs with the conclusion in EC-EVAL #348693.

The conclusion in the EC-EVAL is based primarily on the absence of observed vibration damage to components that were inspected in the most recent Dresden Unit 2 refueling outage. At the time of the inspections, the plant had operated for approximately 700 days at EPU power levels (less occasional reductions in power). Significant damage to QC1 ERV components was noted after only 5 months of operation at less than full EPU power levels. Although a limited amount of vibration data suggests that the vibration levels at lower power levels are similar between Dresden Unit 2 and QC1, MPR agrees that the operating experience from the previous cycle indicates that accelerated component degradation will not occur in the current operating cycle.

Our review of EC-EVALs for other Exelon units (Quad Cities Unit 1 and Dresden Unit 3) is ongoing. We will forward you our comments or conclusions shortly. Please call me or Bill

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McCurdy if you have any questions related to our review efforts or this letter.

Sincerely,

Phillip J. Rush, P.E.

EC# 348693
ATTACHMENT 1
Summary of Vibration Comparison Data

Dresden Unit 2				Dresden Unit 3				Quad Cities Unit 1			
MSL	Direction	Max gs	Hz	MSL	Direction	Max gs	Hz	MSL	Direction	Max gs	Hz
B-1	X	0.01	237	B1	EW	0.12	153	D	NS	0.34	139
B-1	Z	0.07	159	B1	VERT	0.21	139	D	VERTICAL	0.05	139
B-3	VERTICAL	0.11	157	B1	NS	0.04	156	D	EW	0.15	139
A-1	Z	1.06	150	B2	EW	0.06	142	B	NS	0.06	139
				B2	VERTICAL	0.04	153	B	VERTICAL	0.07	139
				B2	NS	0.05	157	B	EW	0.13	139
				D	NS	0.10	11.5				
				D	VERTICAL	0.11	153				
				D	EW	0.15	163				

ATTACHMENT 1B

Engineering Change Evaluation 346515, Revisions 0, 1, 2, and 3 "Evaluation of Quad Cities Main Steam Line Vibrations at Extended Power Uprate (EPU) Power Levels"

EC # 346515
Evaluation of Quad Cities Main Steam Line Vibrations at Extended Power Uprate (EPU) Power Levels

Reason For Evaluation / Scope:

Background

During Q1F51, the 3B Electromatic Relief Valve (ERV) actuator was discovered with significant damage. ERV actuators were refurbished, reinforcing plates were installed (tieback) and vibration instrumentation was added to the ERVs and other Main Steam Line (MSL) components. Exiting Q1F51 and Q1F52 (outage to repair monitoring equipment), Unit 1 was limited to pre-EPU power levels until data could be collected and evaluations performed to conclude that components were not vulnerable to accelerated aging due to the increased vibration levels that would challenge equipment operability or near term structural failures. The components evaluated are listed in the table in the "**Summary: Long Term Component Responses to EPU Vibration Levels**. Also evaluated were Main Steam Piping lines, including branch lines. The result of these evaluations was that all components were found acceptable for full cycle operation, with the exception of the ERVs. Therefore, the ERV vulnerability defined the recommendation for limited operation.

Purpose

This EC EVAL document provides a technical evaluation for allowing operation of Quad Cities Unit 1 up to a continuous power level of greater than 2511 megawatts (MWth) up to 2957 MWth for a maximum of 3500 hours, with an intermediate limitation of 1600 hours pending actions described below. This value is chosen to ensure sufficient time elapse prior to any shutdown to provide some indication of wear. This would facilitate a determination of longer-term acceptability of the ERVs for full power operation. This is also predicated on acceptable operational history for the ERV actuators during the 7355 hours completed in 2003 above 2511 MWth, combined with the understanding that accelerated wear / degradation is the only area of concern. Some levels of degradation of some components will be expected, specifically the ERV actuators. A unit shutdown prior to the end of this period will be required to inspect for wear/degradation and/or to install modifications.

Approach

Vibration data throughout the range of power operation, component failure history and analytical modeling were used to assess component and system responses for full EPU power operation. Vibration data were obtained during the ramp up to and at pre-EPU power level of 2511 MWth (nominal) and then during the ramp up to and at 912 MWe (2910 MWth) on December 30, 2003. This data was extracted and then utilized in evaluations. The purpose of these evaluations was to provide assurance that full EPU power operation will pose no threat to continued equipment operation throughout the remainder of the current fuel cycle or to make specific actions required prior to allowing full cycle operation.

Results

Required actions are included to ensure acceptable equipment performance beyond the 1600-hour limitation to the 3500-hour maximum approved period.

Additional actions are also provided to be completed during the subsequent shutdown and prior to the end of the fuel cycle for Q1R18. Detailed required actions for further testing, evaluations and potential plant changes are also detailed below and the actions required to complete these items are tracked under AR 194877:

Required Actions:

To Remove 1600 hour limitation:

1. Perform detailed testing on the ERV configuration, with Wyle Labs, to confirm analytical results, project wear rate, and to confirm proposed additional modification adequacy. Testing to be completed prior to installing ERVs in Unit 2. (Assignee: A8064MW-DR, due date: 2/23/04)
2. Complete analytical evaluation of the 3E ERV and comparison of results of all four ERVs to confirm Engineering Judgment determination of acceptable operational period of 3500 hours. (Assignee: A8064MW-DR, due date 2/11/04)

Prior/During Planned Outage After 3500 Hours of Operation

3. Obtain a vibration data set monthly and after return to full power operation after any down power of greater than 10% and assess for variation/deviation from the analyzed data and any negative impacts. (Assignee: gather data A8426CMO, due date 5/30/04, Analysis assignee: A8064MW-DR, due date 5/30/04)
4. Install accelerometers on the 1C MSIV during the planned shutdown following 3500 hours of operation. Evaluation of data obtained will be used to define the need for the inspection required in Action 16 below. (Assignee: A8426CMO, due date 5/28/04)
5. Install accelerometers on the 3C ERV pilot valve to confirm model accuracy. (Assignee: A8426CMO, due date 5/28/04)
6. Install modification to the ERVs to reduce pilot valve response to vibration. (Assignee: A8452DEM, Due Date 5/30/04)
7. Inspect the components identified during Q1F51 as degraded. These include the ERV actuators, HPCI 4 valve operator and snubber mounting brackets. (Assignee: A8451NESPR, due date 5/30/04)
8. Monitor weekly individual MSL flows to provide any anomalies that indicate MSIV degradation. (A8452RRT, due date Q1R18, 3/22/05)

Prior to Q1R18

9. Obtain a full set of vibration data when the unit is moved to full thermal power of 2957 MWth for the first time. This data will be used to confirm the assumptions made on expected maximum vibration levels. (Assignee: A8426CMO, due date 5/28/04; Analysis action assignee: A8064MW-DR, due date 6/11/04) (This is also an unverified assumption in Reference 4 calculation, ATI 194877-08)
10. Evaluate the current PM scope and frequency for all components included in this evaluation. The expectation is that some components

will need increased PM frequencies for rebuilds, refurbishments and/or replacements. (Assignee: A8430NESSC, due date 5/21/04)

a. Consideration should be given to perform a post-EPU baseline inspection during the Q1R18 outage.

11. Perform testing on the NAMCO limit switches using the Quad specific vibration predominate frequencies and amplitudes. This testing to be performed prior to approval of full cycle operation. (Assignee: A8064MW-DR, due date 5/20/04)
12. Perform testing on the Limitorque actuator type and size SMB-2-80 or equivalent using the Quad specific vibration predominate frequencies and amplitudes. This testing to be performed prior to approval of full cycle operation (Assignee: A8064MW-DR, due date 5/20/04)
13. On the HPCI 4 valve, create and utilize an analytical model to evaluate the failure analysis results to determine the vulnerable sub-components and to design an effective modification. Install the modification of the 4-rotor design to remove the degradation vulnerability during Q1R18. Closely trend normal IST test data until that outage to ensure that any deleterious degradation is identified prior to challenging valve functionality. (Assignee: three separate tasks will be assigned, due date 6/1/04)
14. Evaluate the snubber inspection plan to ensure that sufficient MSL snubbers are functionally tested during Q1R18 to ensure that degradation is not occurring. Inspections should include the previously degraded snubbers, 1-66 and 1-71. Adjust inspection plan as necessary. (Assignee: A8451NESPR, due date 6/1/04)
15. Complete comparison of Quad Cities Unit 1 to Dresden Unit 3. This comparison may provide further insights into changes that can be made for Quad Cities to minimize the measured vibration level responses such that accelerated component degradation does not occur. This evaluation includes the following elements: (Assignee: A8064MW-DR, due date 6/25/04)
    a. Expansion of the ongoing MSL circuit analyses and/or scale model testing to include the frequency range up to 180 Hz. This may provide insight into the source of the measured 139 and 157 Hz predominate frequencies.
    b. Detailed configuration differences in all MS line branch line connections between Dresden and Quad.
    c. Detailed evaluation of the ERV configuration installed at Dresden.
    d. A comparison of steam flows between Dresden and Quad on MSL and all branch connections during EPU power level operation.

<u>During Q1R18</u>

16. Inspect the 1C MSIV; during the next refuel outage (Q1R18) to confirm that no degradation has occurred. If degradation is found, evaluate the need for expanding inspection to other MSIVs. (Q1R18 – Assignee: A8451NESPR, due date 3/22/05)

17. Perform any additional inspections as determined by evaluations performed in actions 7, 8, 9 and 11. (Assignee: A8450EM, due 3/22/005)
18. Install modification to the HPCI -4 valve rotor. (Assignee: A8452DEE, due date Q1R18, 3/22/05)

Detailed Evaluation:

Component Damage Summary (Quad Cities Unit 1)

The below information is summarized based on information contained in CRs written during the subject forced outage (Q1F51).

CR No.	Brief Description
186979	Documents as found condition of 3B ERV actuator and drain pipe
188050	Walkdown of drywell, MSIV room and top of torus for structural and support issues, summary of three issues found
188052	Walkdown results from heater bay and Unit 1 turbine deck
188128	Equipment issues identified during walkdown of drywell first level and MSIV room
188185	Issues with HPCI 2301-4 valve MOV found during walkdown of MOVs inside containment
188202	Discrepancies noted during drywell walkdown of second, third and fourth levels.

ERV/Solenoid Historical Failure Summary:
ERV Actuator refurbishment history back to 1990 indicates that ERVs have experienced wear similar to that noted in Q1F51, although not as severe, prior to EPU operation. ERV issues identified in Q1F51 are included in the table below, with repairs made as stated. A formal Root Cause is documented under AT 186979-13. More details on the historical problems with ERV actuator are provided in the root cause report referenced here. This information was used in part as input into the current evaluations.

EPN	As Found Actuator Conditions	As Found Visual and PT Exam results	As Left Actuator Conditions	As Left Visual and PT Exam results
1-0203-3B	▪ Left actuator spring protrudes through bushing ▪ Cover welds broken ▪ Guide Rods grooved from spring wear ▪ Limit switch arms broken and/or missing	▪ Pilot drain line broken ▪ Other welds had no recordable indications ▪ Cold spring load noted during attachment of pilot drain line flanges after valve replaced	Actuator refurbished	▪ Drain line replaced to first flange ▪ Cold spring removed
1-0203-3C	▪ General condition sat, springs just beginning to wear into bushings	▪ 1 inch weld on pilot drain line weld slightly under-filled ▪ Cold spring noted	Actuator refurbished	▪ No action taken - under-filled weld acceptable ▪ Cold spring removed

1-0203-3D	▪ Trip plate pivot point sloppy	▪ 1 inch line at ERV (between ERV and pilot) had 1/8 inch puddle mark in toe of weld at ERV (acceptable) ▪ Slight undercut indication at toe of weld near ERV all the way around	Actuator refurbished	▪ Valve replaced ▪ Results on replacement valve were sat (visual and PT) ▪ No cold spring noted
1-0203-3E	▪ Cracking of spring plate noted at shorting bar ▪ Grooves worn into guide rods ▪ Springs just beginning to wear into bushings	▪ 1-inch pilot drain line has small pitting filled with paint. Pits indicated to be less than 1/8 inch. ▪ No cold spring noted	Actuator refurbished	▪ Flapped the area of the drain line near the pilot valve and re-inspected

ERV Pilot Drain Line Historical Failure Summary:
Drain line breakage has occurred prior to EPU conditions, attributable to both cyclic fatigue, as well as personnel interaction issues.

Snubber Issues discovered during walkdowns in Q1F51:
- TS 1-66 (for 3E ERV discharge piping) - clamp found loose and rotated. A nearby clamp on same pipe was tight. Snubber was tested satisfactorily. The loose clamp was most likely the result of improper installation.
- TS 1-71 (support for C MSL) - load pin was nearly pulled out. This was likely the result of snubber proximity to workers and worker catching PCs on the pin. The pin will be replaced by a stud in the upcoming refuel outage, preventing recurrence.

Steam Dryer Issues
Steam Dryer issues have impacted station performance. Repairs to the dryer have been completed. A formal root cause is underway under CR 188129 and will be completed in January 2004.

Other vibration related issues identified on Unit 1:
- Vibrations in HPCI piping resulted in LLRT test tap removal
- Small Bore Piping Failures in MS, FW, CD Systems (but issue primarily on Unit 2).
- Miscellaneous vibration-related clamp looseness/damage issues identified in Q1F51 (moisture separator supports, MSL low point drain line tiebacks) - these items were corrected.
- Damage noted to 1-2301-4 limit switch rotors in Q1F51, but EPU vibration tie not yet established.

Conclusions
- **Damage, particularly that on the ERVs, generally can be characterized as fretting, wear and piece part fatigue.** This damage can be considered relatively long-term (months), and would not lead to rapid, catastrophic failure.
- With the exception of 3B ERV (Failed Drain Line), QC-1 ERVs were operational.

Effects of Increase to Full EPU Power 2957 MWth

Based upon a review of the thermal modeling for the Unit 1 Reactor heat balance and operating data from the summer, the maximum steam velocity through the MSLs at full power (2957 MWt) was determined to be 226 ft/sec for the Unit 1 "A" steam line. The 'A' line is evaluated here because it is the shortest line and therefore has the highest steam velocities. This velocity represents a 23% increase over the corresponding maximum velocity pre-EPU (at 2511 MWt) of 184 ft/sec for the "A" line, which is where Unit 1 is currently operating. The ramp up to 912 MWe on Unit 1 required a reactor thermal power of approximately 2910 MWt, which would result in increased flow velocities in the main steam piping to 222.6 ft/sec. This conclusion agrees with the expected response of the system to such conditions. Since the mass flow rate through the main steam piping is approximately proportional to the thermal power, operation at a thermal power approximately 1.5% less than 2957 MWt would be expected to result in a steam flow rate of approximately 1.5% less. The impact of the future 1.5% increase in thermal power will be increases in the measured vibration levels.

All of the component evaluations utilized the measured data obtained during the December 30, 2003 ramp up to 912 MWe. These values represent the plant response to 2910 MWth. At some point in the future, power levels are anticipated to reach full licensed power of 2957 MWth. For most of the components, excepting 3C ERV, the predicted increase can be calculated by plotting the available data. For 3C ERV the prediction for full power vibration values are that the magnitude will not significantly change. This is predicated on the small change seen from 2488 MWth nominal to 2910 MWth (4.14 grms to 4.17 grms). Prediction graphs are included in Attachment 1. The accuracy of these predictions is influenced by the fact that the change is a function of both flow rate and proximity of the vortex shedding frequency to the acoustic frequency. Each component evaluation includes a discussion of any potential effects of increased vibration levels as the unit approaches full thermal power. Actual vibration levels will be measured when the unit stabilizes at full power to confirm the assumptions made.

Acceptability of ERV Component Operation at EPU Levels

The four ERVs have virtually identical assemblies, which consist of the main ERV valve body, pilot valve, and solenoid actuator. The pilot valve is connected to the ERV by means of a turnbuckle and a pilot valve tube. A tie back support has recently been added to all four valves. Each valve has small diameter leakoff piping that is routed back to the ERV discharge line. Detailed finite element analyses, using the finite element program ANSYS, have been completed for ERV 3B, 3C, and 3D and are documented in Reference 2.

The finite element analyses determined natural frequencies and their corresponding mode shapes. The natural frequencies and mode shapes were

compared to the frequency content of the measured vibration data, and it was determined that a pilot valve pendulum mode is the primary cause of the excessive solenoid wear.

Correlation of Bump Test Data to Analytical Models

Bump test data for 3B and 3D ERV and dynamic data were obtained during plant start-up on December 22nd to 100% power (2488 MWth) and power ascension to EPU power level (2910 MWth) on December 30th. Two ERVs, 3B and 3D, were subjected to impact tests while recording the accelerometers on the ERV and the pilot valve. The magnitude of the data was not controlled, so the spectra plots were assessed for frequency content only. A review of the bump test data shows the frequencies of the 3B and 3D ERV models, with the tie back supports, compares well to the bump test data. Thus, the analytical models are assessed to be representative of the installed ERVs and can be used to evaluate the vibration loads and response, and effect of the tie back support.

Evaluation of Vibration Test Data

The vibration data was reviewed for amplitude versus power level to identify key frequencies and to observe amplitude versus power level trends. Two key frequencies showed up at all accelerometer locations: 139 Hz and 157 Hz. These two frequencies had very sharp ramps (similar to AC spikes) indicating low system damping. While the magnitude varied from channel to channel, these two frequencies were dominant on all channels and the magnitude usually increased proportional to operational power level.

The measured vibrations at 139 Hz and 157 Hz are acoustic frequencies that are systemic with the steam pipe system, since they were detected at all accelerometer locations. The 139 Hz frequency is observed at all recorded power levels and amplitude increases are proportional with power level (with the exception of 3C ERV). The 157 Hz frequency is <u>not</u> observed at 2212 MWth and appears at power levels above 2384 MWth. In addition to the vibration at 157 Hz, the vibration magnitude at the ERV 3B inlet flange increases at 139 Hz, as the power increases to 2910 MWth, whereas, the vibration magnitude on the ERV 3C inlet flange increases rapidly only at 139 Hz. As noted above, the two discrete frequencies are very sharp.

Effect of the Tie Back Support

Evaluations were performed to assess the effect of the tieback support on the ERV assembly and drain piping. The primary benefit of the tie back support is to reduce amplification of accelerations at low frequencies (i.e., less than 50 Hz). Modal analyses were performed using the ERV 3B finite element model with and without the tie back support. The analytical results were found to compare well to the bump test results. Thus, the analytical model was used to predict the natural frequencies of the ERV with and without the tie back support. A

comparison was made of the calculated natural frequencies, mode shapes and modal participation factors near the predominant vibration frequencies of 139 Hz and 157 Hz for both the ERV 3B with and without the tie back support. The results of the comparison show that the addition of the tie back support does not change the dynamic characteristics of the ERV near 139 Hz and 157 Hz. Thus, at the predominant vibration frequencies of 139 and 157 Hz, the tie back support provides no significant improvement; however, there is also no negative impact.

Wear and Degradation

Historical data shows that ERV 3E actuator typically sustains more actuator wear, during pre-EPU, than the ERV 3B actuator. This was not the case in outage Q1F51 after approximately 7355 hours of operation at power levels above 2511 MWth. The ERV 3B actuator experienced significant wear and the valve actuator became inoperable. During the same outage, the ERV 3E actuator saw significant wear but the valve was still operable. The key difference was that the actuator spring wore through the bushing on ERV 3B, whereas the ERV 3E actuator springs were just beginning to wear into the bushings. In addition to the significant actuator wear found on ERV 3B, the drain pipe connection to the pilot valve was broken. The failure of the drain pipe was attributed to socket weld quality and cold spring.

Configuration and Turbulent Flow Assessment – ERV 3E and 3B

Main Steam Line (MSL) B has two ERV branch lines and a HPCI branch line. The ERV 3E branch line is just downstream of a MSL bend radii, ~2.5 diameters (MSL outside diameters, 20 inches) past the bend radii. The HPCI branch line is downstream about 5.2 diameters past the bend radii, and the ERV 3B branch line is about 9.6 diameters past the bend radii. The ERV pilot valve orientations are the same, and the pilot valve turnbuckle is normal to the MSL.

All three valves are subjected to the same steam flow rates and the turbulent flow caused by the two acoustic frequencies at 139 and 157 Hz. These acoustic modes are a result of turbulent flow separation and structural coupling with the geometry of the branch lines. The flow turbulence is approximately the same magnitude for all valves in this MSL segment, since the turbulent flow does not attenuate due to the high flow rate (typical 203 ft/sec) and the low steam density (2.24 lbs/ft^3). Therefore, all three-valve branch lines experience essentially the same vibration levels.

ERV Modal Assessment – Drain Line Intact

ERV 3B was modeled to determine its modal characteristics. The results of the modal analysis determined the key mode shape that could cause this type of wear. For this mode shape, the frequency coincided with one of the predominant vibration frequencies that was identified as a pilot valve pendulum mode at 157 Hz. Since ERV 3E has similar geometry and orientation on MSL B, the modal

displacements would be similar. The natural frequency near 157 Hz for ERV 3C shifts slightly and is attributed to the differences in the drain pipe routing.

ERV 3B Modal Assessment – Drain Line Failed

ERV 3B was modeled with the drain line detached to simulate the failure of the drain line connection to the pilot valve found during the Q1F51. The modal response of this valve configuration (i.e., the pilot pendulum mode) dropped to 140 Hz, which was very close to the 139 Hz acoustic mode. A dynamic assessment of this configuration shows a greater dynamic motion for this pilot valve (due to its increased flexibility) than ERV 3B with an intact drain line. Based on the measured vibration for the ERV 3B inlet flange, the vibration is 0.52 g rms at 139 Hz and 0.22 g rms at 157 Hz. Since the drain line failure causes a drop in the natural frequency of the actuator to 140 Hz, the actuator would be subjected to higher vibration levels than prior to the drain line failure.

ERV Design Modification

Based on the results of the dynamic analysis, correlation to the test data, and review of the actuator degradation, the long term recommendation is to replace the tie back support with a pilot valve modification (a reinforced plate bolted or welded to both ends of the pilot valve – see Figure 1 or some alternative which results in a significant relative stiffness change). This design modification will not eliminate the cause of the vibration but is considered economically feasible, since eliminating the acoustic vibration would entail significant main steam and branch line changes.

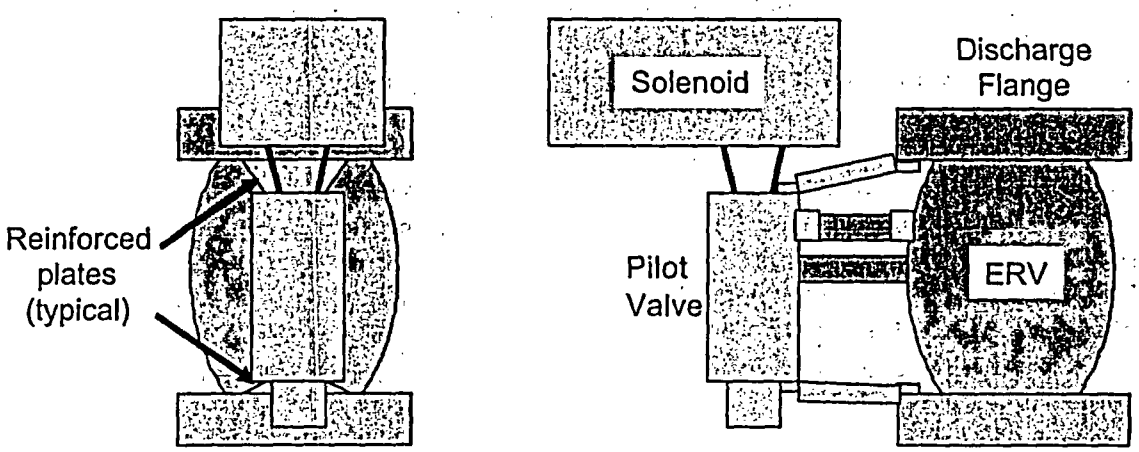

Figure 1. Plates Supporting the Ends of the Pilot Valve

Operation at 2910 MWth and Higher

The ERV 3B actuator degradation was found to be greater than ERV 3E during Q1F51. A comparison of the measured vibration for ERVs 3B and 3E shows that

the vibration of both valves has predominant vibration at 139 Hz and 157 Hz. In addition, the measured vibration magnitudes at the inlet flanges are almost the same. Since their measured vibration and valve geometries are similar, their modal responses are expected to have similar pilot valve modes. Therefore, their dynamic responses are expected to be similar. This would also imply that their wear rates/degradation should also be similar. A dynamic analysis of ERV 3E should be performed to confirm its modal characteristics.

ERV Risk Assessment For ERV 3B and 3E

Additionally, the ERV 3B modal analysis indicates that the valve has increased flexibility with a failed drain line and the dynamic response is much greater (10-20% greater). It is this additional motion that results in greater wear on ERV 3B than ERV 3E. If the drain line had not failed, the wear is expected to have been similar to ERV 3E.

Bushing wear/degradation is a function of time. The wear starts out slow and increases as time increases. As the wear increases, the dynamic response moves larger distances and creates larger dynamic impacts (due to limit cycle motion), which increases the wear rate. For operation up to 1600 hours, the wear at the end of this period is expected to be less than ¼ of that observed during outage Q1F51. Additionally, since the drain line has been repaired, the accelerated rate attributed to the increased pilot valve flexibility would not be a factor. Therefore, plant operation for 1600 hours at 2957 MWth would present a very low risk of wear degradation that would challenge the operation of the actuator.

Risk Assessment For ERV 3C

The ERV 3C actuator has similar modal behavior as the ERV 3B actuator. The key difference is that the ERV 3C actuator observed wear is historically low and the usual amount of wear was found in Q1F51. Additionally, the vibration data indicates low acceleration magnitudes at 157 Hz. The high vibration for ERV 3C at 139 Hz is along the Y-axis (vertical direction). This acoustic frequency would not excite the pilot pendulum mode at 157 Hz, and the vibration is normal to the mode shape motion. Therefore, this mode shape would not respond. This is the reason the ERV 3C has not seen excessive wear. Therefore, plant operation for 1600 hours at 2957 MWth would present an insignificant risk of wear degradation that would challenge the operation of the actuator.

Risk Assessment For ERV 3D

ERV 3D has a similar configuration to the other ERVs, but the pilot valve is parallel to the main steam flow. ERV 3B, 3C and 3E are all normal to the main steam flow. The pilot valve pendulum mode occurs at 159 Hz, but since the valve is oriented parallel to the main steam line, the dynamic response is not as great, which does not permit ERV 3D to wear as much. Therefore, plant

operation for 1600 hours at 2957 MWth would present a very low risk of wear degradation that would challenge the operation of the actuator.

ERV Assessment Summary

Since QC1 has operated at EPU for at approximately 9 months (7355 hours at >2511 MWth), and the ERVs were demonstrated to operate in their degraded condition, QC1 can continue to operate at EPU power levels (up to 2957 MWth) for a limited time. Based on the fact that this degradation is a result of the acoustic mode vibration at 157 Hz and that this damaging frequency is still present, then ERV actuator wear is expected to continue. The wear phenomenon is time dependent, therefore, the longer the plant operates, the more wear/degradation will be observed.

For operation at 2957 MWth (slightly higher power level than the 2910 MWth level attained on December 30th), it is expected that the vibration levels would increase in proportion to main steam flow velocity squared. From the vibration trend plots, Reference 1, the 157 Hz frequency appears to have reached a plateau, whereas, the response at 139 Hz seems to be increasing. Therefore, it is projected that the response at 139 Hz will continue to increase.

If QC1 continues to operate at 2910 or 2957 MWth, the ERV solenoid and pilot valve vibration levels are expected to continue. Based on the analysis that has been performed to-date, Reference 2, it is expected that ERV 3B (with an intact drain line and a tie-back support) would wear no more than ERV 3E (with a tie back support) at 2910 or 2957 MWth. The wear/degradation of ERVs 3C and 3D would not be worse than the wear found during Q1F51 due to the lower measured vibration levels at these valves.

ERV Recommendations

The following recommendations are made:

- Inspect the ERVs at the next outage to quantify the wear.
- Replace the current tie back support with supports that restrain the pilot valve pendulum motion (See Figure 1). The final configuration should be determined by testing and analysis.
- During the next available plant shutdown, install accelerometers on the ERV 3C pilot valve.
- Perform vibration testing of the ERV. This should include the tie back support and any proposed modification to confirm the analytical results.
- Perform a modal analysis of ERV 3E.

Evaluation of Piping

Piping models were generated; see Reference 3, for main steam line A, B and C. These models include the main steam piping from the reactor vessel nozzle to

the drywell penetration and include the relief valve discharge piping to the drywell penetration.

In addition, piping models were generated using the computer program, PIPESTRESS, for the ERV assembly and associated leak off piping. The leak off piping is very similar for the four valves. Therefore, the valve assemblies and the connecting drain piping were evaluated by applying the highest measured acceleration on each ERV for the appropriate main steam line.

The maximum peak accelerations at 2910 MWth as shown in Attachment 4 Table 1 were applied to each piping model and stresses were calculated. The analysis was done by the response spectrum modal superposition method, with the peaks conservatively broadened by ±10% to account for uncertainty. Use of peak rather than rms values conservatively assumes that all vibration cycles are at the peak value, whereas the majority of the cycles are at a much lower level.

Stresses calculated for the piping were compared to the endurance limit as specified by ASME OM Standard Part 3. The allowable stress is 3846 psi, based on the OM-3 criteria. The maximum calculated stress for the ERV piping was 1959 psi, about 50% of the OM-3 allowable. It should be noted that the OM-3 criteria contains a number of safety factors which when taken together, produce a very conservative result.

The evaluation for the effect of the ERV vibration on the main steam piping has been done by analyzing the ERVs on main steam lines A, B, and C using the PIPESTRESS program. MSLB is the limiting line because it contains two ERVs, plus the HPCI tie-in is located between the ERVs. The piping model includes the large bore main steam line B piping, the safety valves, ERVs 3B and 3E, the ERV discharge lines, and the HPCI line. Response spectra analyses were performed applying the maximum measured peak response at the 3B ERV pilot valve to the entire piping. Although limited acceleration data is available on the main steam lines, the accelerations along the pipe are expected to be significantly less than at the pilot valve based on geometry differences. The maximum stress is 2396 psi, at the connection of the ¾" drain line to the 3E pilot valve. All of the stresses are within the OM-3 allowable. The high stress points are all associated with the ERVs, the safety valves or the tees to those valves. The stress at the connection to the HPCI line is only 607 psi.

A second piping model was created for main steam line A (MSLA). MSLA contains a tie-in to the RCIC system. MSLA does not contain an ERV (it has the Target Rock SRV), thus, the Target Rock 3A measured accelerations were conservatively applied to all of the piping. The maximum stress is 1317 psi, at the branch end of the tee to safety valve 4E. The same point on the branch to safety valve 4A has a similar stress. All of the stresses are within the OM-3 allowable. The high stress points are all associated with the safety valves, or the tees to those valves. The stress at the connection to the RCIC line is 1137 psi.

A third piping model was created for main steam line C (MSLC). Response spectra analyses were performed by applying the maximum measured peak response at the ERV 3C to the entire piping. The maximum stress is 1405 psi, at the connection of the ¾" drain line to the 3C pilot valve. All of the stresses are within the OM-3 allowable.

Thus, the vibrational stresses in the piping are below the OM-3 allowable so operation at EPU power levels will not affect the integrity of the piping.

Evaluation of MSIVs

The evaluation of the MSIVs was performed by comparing measured values to seismic aging qualification test data for the Unit 1 actuators. (See Attachment 3) The seismic aging testing consisted of sinusoidal variation of frequency between 25 and 200 Hz at 0.75 g applied to the valve bonnet in accordance with IEEE-344 for vibration endurance testing. Measured values for the B MSIV (inboard) and extrapolated values for the A and D MSIVs were well bounded by this aging evaluation and were therefore determined to be acceptable for continuous full EPU power operation (Attachment 3).

The extrapolated values for the C MSIV, which were based on measured data taken at the ERV inlet flange, were outside the bounds of the aging evaluation. This extrapolation is very conservative because it uses localized data obtained at the ERV inlet flange and extrapolates to the MSIV (inboard). Continued operation through the remaining operating cycle is deemed acceptable because of the following:

1. The 1C and 2C valves were reworked during Q1R17, which included general inspection of valve internals. New trim and liners were installed at that time. No indication of stem/disk wear was noted in the as found condition.
2. The measured vibration levels at the 3C ERV inlet flange are not significantly increased at EPU power versus 2511 MWth level (see Attachment 1). This means that the vibrations being experienced by the MSIVs are most likely very close to those seen during the first twenty years of unit operation. There is no history of stem/disk problems on the 1C MSIV.
3. During Q1F51 the MSIVs were closed to allow for MSL plug installation. There were no problems noted during this evolution.

Operation up to Q1R18 is approved, with the following required actions to confirm acceptable performance.

1. Perform a full stroke test of the inboard C MSIV after several months of full EPU power operation. This action will be performed during the planned shutdown after 3500 hours of operation.

2. Install accelerometers on the 1C MSIV when in a planned outage to obtain actual data on that MSIV. Action to be completed during planned shutdown after 3500 hours.
3. Monitor individual MSL flows for any change that could be indicative of degradation of an MSIV.
4. Inspect the C inboard MSIV during Q1R18 for any signs of wear related degradation.

Evaluation of HPCI 4 Valve Operator

Based on walkdown data, the limit switch (4 rotor design) internal to the HPCI 4 valve operator was found degraded. This limit switch was replaced during Q1F51 and the old component was sent out for failure analysis. Preliminary results of this analysis confirm that the limit switch continued to meet its electrical requirements in its as found condition. This information combined with the evaluation documented in Attachment 4, confirm the acceptability of this valve operator for full cycle operation. Inspection of the operator during the planned outage after 3500 hours is required.

Additional analyses of the operator assembly will also be performed to determine the susceptible subcomponents and to develop a potential modification to eliminate the vulnerability. These actions will be completed to support modification during Q1R18.

Summary: Long Term Component Responses to EPU Vibration Levels

An assessment of the component response to EPU power levels has been performed and is summarized in the following table:

Component	2910 MWt Assessment Results	Expected long term EPU Performance	Comments
MSIV Actuators, except on 1C valve	Sinusoidal aging test @ 0.75 g's from 25 to 200 Hz -	Acceptable at EPU levels based on existing margin with the exception of the 1C valve. Recommendations include Internal inspection during Q1R18, full stroke testing or validation of acceptable vibration levels required for full cycle qualification.	Attachment 3
Inboard and Outboard MSIV Limit Switches	Seismically rugged 21.06 g's, 29 Minute test at 7.5 g's – Acceptable margin.	Acceptable at EPU levels based on existing margin, testing required validating acceptability for long-term operation at EPU measure vibration levels.	Attachment 2
Inboard and Outboard MSIV	Passive Item, very rigid subjected to small G's	Acceptable - no wear or fretting after 1 year at EPU	Attachment 2

Temperature Elements	acceptable	levels	
MSIV/RCIC LLRT Taps	Very low frequency based on analysis – no significant excitation measured at low frequency Acceptable for EPU levels	Acceptable – No indication of degradation after 1 year at EPU levels	Attachment 2
Safety Valve Acoustic Monitors	Sine beat test accelerations 6.0 g's compared to 4.226 g's	Acceptable at EPU levels based on existing margin	Attachment 2
Safety Valve Temperature Elements	Pipe clamp style thermocouple, passive element without any extended masses.	Acceptable - no wear or fretting after 1 year at EPU levels	Attachment 2
RCIC Inboard and Outboard PCI Operator and Limit Switch – Limitorque Operator	Tested at values between 8.25 g and 37 g for 351 minutes. Measured values 3.608 (SRSS peak)	Acceptable as no rotor wear noted after 7355 hours of operation at >2511 MWth	Attachment 2
Pressure Switches	Seismic qualification tested @ 20 g's subjected to less than 0.2 g's Acceptable based on margin	Acceptable – No indication of degradation after 1 year at EPU levels	Attachment 2
ERVs valves, actuators, drain lines, etc	Analyzed using 2910 MWth vibration data and Finite Element Model (FEM)	All components acceptable for EPU levels based on margins and operating experience for 1 year except actuators. At EPU levels for 1 year w/o tieback actuators experienced wear degradation but were operable. 3B ERV exception caused by drain line damage due to weld deficiency and pipe cold spring. Further testing and modification required.	Attachment 4
Safety valves	Seismic qualification 11 g horizontal and 9 g vertical compared to ERV measured values of 1.52 g's horizontal and 4.17 g's vertical. Acceptable based on margin	Acceptable – Based on existing margin and no indication of degradation after 1 year at EPU levels	Attachment 4
HPCI 4 and 5 valve, operators and limit switches	Seismic qualification 6 g horizontal and 4 g vertical compared to 0.395 grms horizontal and 0.589grms vertical. Acceptable based on margin and power labs preliminary test results showing full electrical functionality	Acceptable – Based on existing margin. CR 188185 and 188945 detail as found condition. Modification of the limit switch may be desirable to prevent future degradation.	Attachment 4
Target Rock (3A)	Compared to qualification information on new type 2-stage valves.	There is some potential for pilot leakage and thread wear on the main piston/stem joint.	Attachment 3

		Pilot valve leakage would be detectable by increased tailpipe temperatures.	
MSL Drain valves 1-0220-1 and 2, operators and limit switches	Estimated life duration based on the seismic testing information and extrapolated measured vibration levels is 38 years.	Acceptable for full power operation. Valves are closed after turbine warm-up and left closed during normal operation, hence position change is only required during a unit shutdown.	Attachment 2

Operating Experience at EPU Power Levels

Between 12/1/02 and 11/11/03, Quad Cities Unit 1 operated for approximately 8300 hours at EPU conditions. The table below summarizes the run time at various power levels above pre-EPU 100% power (corresponding to ~85% power post-EPU): all data was taken from PI that utilizes the unit process computer (data point QDC01V_D601)

QC-1 12/1/02 to 11/11/03	Run Hours	% Of Total Run Hours
Total	8300	100
> 85% Power	~7355	89
> 90% Power	~7279	88
> 97% Power	~7041	85

The damage observed during November 2003, which led to this evaluation, was found after approximately ½ of the normal two-year operating cycle or 7000 hours at greater than 97% thermal power. The amount of service time that resulted in the observed degradation provides a level of assurance, based on empirical data, analyses and Engineering Judgment that the 3500 hours of full power operation approved by this EC will not be detrimental to equipment functionality. A temporary limitation of 1600 hours of operation is imposed pending completion of the additional analyses and testing to confirm the Engineering Judgment determinations. Some levels of degradation of the components as discussed above will be expected. A unit shutdown at or prior to the end of this period, 3500 hours, will be required to inspect for wear/degradation and/or to install modifications.

Conclusions / Findings:

EC # 346515
Evaluation of Quad Cities Main Steam Line Vibrations at Extended Power Uprate (EPU) Power Levels

This EC EVAL provides an Engineering Evaluation supporting operation above 2511 MWth up to 2957 MWth for a maximum of 3500 Hours, with an interim limitation of 1600 hours pending additional analyses and testing. It has been determined that this operational period will not result in imminent failure or unacceptable degradation levels of any components. The conclusion is provided by evaluation of the empirical data, including both operational data and ERV testing during the outage to determine natural frequencies. Based on these evaluations, the ERV actuator and the HPCI 4 valve limit switch are identified as having some potential for wear or fatigue degradation at EPU power levels.

Attachments:

Attachment 1: Summary of Current MSL Vibration Levels – Quad Cities U1 – for complete data information see QDC-11Q-302 (Calculation)

Attachment 2: Evaluation of Components for Vibration Effects at 2910 MW

Attachment 3: "Evaluation of Quad Cities Unit 1 MSIV and SRV Vibration," General Electric Report No. DRF-0000-0023-4260, dated January 9, 2004

Attachment 4: Structural Integrity Associates Vibration Assessment

References:

1. QC-11Q-302, Revision 0, Quad Cities Unit 1 Main Steam Line Vibration Data Reduction.
2. QC-11Q-301, Revision 0, ERV Finite element Analysis and Vibration Evaluation.
3. QC-11Q-303, Revision 0, Analysis of Main Steam Piping
4. QDC-0200-M-1360, Revision 0. Evaluation of Components for Vibration Effects.

EC# 346515
ATTACHMENT 1
Current MSL Vibration Levels – Quad Cities U1

		12/1/2003 2143 MW 5 to 200Hz Bandwidth			12/1/2003 2385MW 5 to 200Hz Bandwidth			12/1/2003 2474 MW 5 to 200Hz Bandwidth			12/22/2003 2488 MW 5 to 200Hz Bandwidth			
Tape Deck	Channel	grms	gmax	gmin	grms	gmax	gmin	grms	gmax	gmin	grms	gmax	gmin	
QCTD	1										0.0862	0.3609	-0.3474	3B ERV Inlet Flange
	2										0.2954	0.9251	-0.9113	3B ERV Inlet Flange
	3										0.0824	0.3291	-0.3548	3B ERV Inlet Flange
	4	0.191	0.822	-0.872	0.5673	5.9633	-6.7601	0.776	3.36	-3.56	0.6857	2.6619	-2.6698	3B Pilot Valve
	5	0.109	0.489	-0.449	0.4019	1.5194	-1.5979	0.4903	1.7914	-1.8004	0.8734	2.3504	-2.4321	3B Pilot Valve
	6	0.062	0.28	-0.304	0.1905	0.6431	-0.6183	0.213	0.7613	-0.7041	0.2198	0.7517	-0.7564	3B Pilot Valve
	7	0.031	0.15	-0.14	0.0808	0.2844	-0.2904	0.0853	0.2944	-0.3283	0.0782	0.3167	-0.2837	3E ERV Inlet Flange
	8	0.088	0.424	-0.425	0.3502	1.0789	-1.0627	0.3769	1.1807	-1.176	0.3952	1.179	-1.2683	3E ERV Inlet Flange
	9	0.04	0.166	-0.168	0.0588	0.2531	-0.2611	0.0636	0.3045	-0.277	0.0637	0.2773	-0.2536	3E ERV Inlet Flange
	10	0.054	0.275	-0.283	0.1354	0.5594	-0.5665	0.1565	0.6205	-0.6085	0.1365	0.4955	-0.4989	HPCI 4 Valve
	11	0.061	0.381	-0.331	0.1835	0.6909	-0.6478	0.2108	0.7675	-0.7392	0.2088	0.7313	-0.6828	HPCI 4 Valve
	12	0.136	0.582	-0.61	0.2597	1.0411	-1.1193	0.311	1.1931	-1.3048	*N/A*	*N/A*	*N/A*	HPCI 4 Valve
	13										0.6437	2.4767	-2.5224	x- 3B pilot valve (ch 4)
	14										0.8361	2.415	-2.242	y- 3B pilot valve (ch 5)
	15										0.285	0.8907	-0.8604	y- 3B inlet flange (ch 2)
		grms	gmax	gmin	grms	gmax	gmin	grms	gmax	gmin	grms	gmax	gmin	
DTD	1	0.163	0.672	-0.618	0.2028	0.945	-0.881	0.2439	1.0157	-0.9589	0.2338	0.8405	0.6898	3A Target Rock
	2	0.209	1.06	-0.976	0.293	1.2877	-1.2718	0.335	1.4581	-1.3957	0.3347	1.3052	-1.3434	3A Target Rock
	3	0.04	0.184	-0.175	0.0608	0.332	-0.3769	0.0718	0.4305	-0.4455	0.0771	0.3895	-0.4521	3A Target Rock
	4	0.042	0.185	-0.183	0.0626	0.2884	-0.2642	0.0704	0.2849	-0.303	0.0694	0.2874	-0.288	3D ERV Inlet Flange
	5	0.098	0.441	-0.445	0.1696	0.7825	-0.8738	0.2714	1.1818	-1.0618	0.2693	0.9522	-0.9889	3D ERV Inlet Flange
	6	0.074	0.313	-0.306	0.0952	0.4193	-0.3978	0.1029	0.4482	-0.4472	0.1062	0.4738	-0.4491	3D ERV Inlet Flange
	7	0.163	0.745	-0.75	0.2253	1.1724	-1.0387	0.2725	1.1664	-1.1864	*N/A*	*N/A*	*N/A*	3D Pilot Valve
	8	0.134	0.602	-0.585	0.1974	1.0602	-0.8578	0.3261	1.2	-1.2472	0.3408	1.5371	-1.4911	3D Pilot Valve
	9	0.349	1.685	-1.346	0.447	2.0637	-1.8479	0.4579	2.5693	-1.7693	0.223	0.9032	-0.9272	3D Pilot Valve
	10	0.052	0.211	-0.255	0.3472	0.8093	-0.8744	0.4471	0.9609	-1.0192	*N/A*	*N/A*	*N/A*	3C ERV Inlet Flange
	11	0.992	3.6	-3.55	2.8248	7.3697	-7.5412	3.2098	7.7175	-7.7526	4.1449	8.3377	-8.065	3C ERV Inlet Flange
	12	0.121	0.428	-0.464	0.337	0.885	-0.919	0.385	0.9354	-0.9682	0.4798	1.2177	-1.1456	3C ERV Inlet Flange
	13	0.036	0.155	-0.207	0.0586	0.2465	-0.2509	0.0665	0.2972	-0.2586	0.0605	0.2529	-0.2451	1B MSIV
	14	0.029	0.188	-0.239	0.0395	0.2327	-0.2758	0.0438	0.2079	-0.2939	0.0427	0.2304	-0.2643	1B MSIV
	15	0.044	0.222	-0.205	0.0743	0.3382	-0.346	0.0827	0.3717	-0.3425	0.0865	0.3407	-0.4032	1B MSIV

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Current MSL Vibration Levels – Quad Cities U1

Tape Deck QCTD Channel		12/30/2003 2662 MW 5 to 200Hz Bandwidth			12/30/2003 2799 MW 5 to 200Hz Bandwidth			12/30/2003 2910 MW 5 to 200Hz Bandwidth		
		grms	gmax	gmin	grms	gmax	gmin	grms	gmax	gmin
1	3B ERV Inlet Flange	0.0972	0.3428	-0.2449	0.0803	0.3382	-0.2799	0.0808	0.318	-0.278
2	3B ERV Inlet Flange	0.4014	1.2277	-1.2291	0.4915	1.8521	-1.8079	0.7028	2.2031	-2.2076
3	3B ERV Inlet Flange	0.114	0.417	-0.4456	0.1374	0.4986	-0.5151	0.181	0.6176	-0.6824
4	3B Pilot Valve	0.8884	3.6393	-3.5527	1.1795	4.9836	-5.1776	1.5159	5.4604	-5.3618
5	3B Pilot Valve	1.3299	3.1803	-3.3652	1.8113	4.2011	-4.2987	2.0423	5.2175	-5.3952
6	3B Pilot Valve	0.3421	1.1008	-1.1297	0.4723	1.5155	-1.4756	0.5606	1.8762	-1.7225
7	3E ERV Inlet Flange	0.12	0.4221	-0.416	0.1435	0.4995	-0.4929	0.1812	0.5846	-0.7291
8	3E ERV Inlet Flange	0.5545	1.7307	-1.9407	0.6889	2.304	-2.2608	1.0183	3.3266	-3.0005
9	3E ERV Inlet Flange	0.0882	0.4014	-0.3936	0.1009	0.4229	-0.4205	0.1263	0.5337	-0.5158
10	HPCI 4 Valve	0.2011	0.8248	-0.8172	0.2662	1.17	-1.1081	0.3951	1.2292	-1.2465
11	HPCI 4 Valve	0.3061	1.0753	-1.074	0.4066	1.4902	-1.4673	0.5894	1.6807	-1.6645
12	HPCI 4 Valve	*N/A*	*N/A*	*N/A*	*N/A*	*N/A*	*N/A*	*N/A*	*N/A*	*N/A*
13	x- 3B pilot valve (ch 4)	0.8341	3.3587	-3.3628	1.1038	4.9286	-4.6938	1.4317	5.0637	-5.1765
14	y - 3B pilot valve (ch 5)	1.2766	3.2187	-3.072	1.7332	4.2203	-4.1865	1.9669	5.3074	-5.0964
15	y - 3B inlet flange (ch 2)	0.3851	1.1479	-1.182	0.4708	1.7417	-1.758	0.6729	2.1343	-2.0979
DTD		**grms**	**gmax**	**gmin**	**grms**	**gmax**	**gmin**	**grms**	**gmax**	**gmin**
1	3A Target Rock	0.2041	0.846	-0.7696	0.2087	0.8857	-0.8394	0.218	1.0149	-0.9143
2	3A Target Rock	0.3302	1.2178	-1.2339	0.3543	1.2967	-1.4076	0.363	2.8195	-3.2408
3	3A Target Rock	0.1068	0.4698	-0.5916	0.1324	0.5563	-0.5929	0.1505	0.6191	-0.7107
4	3D ERV Inlet Flange	0.0969	0.4632	-0.3749	0.1084	0.4244	-0.4141	0.1253	0.4681	-0.474
5	3D ERV Inlet Flange	0.4885	1.6036	-1.6242	0.5667	1.7102	-1.6355	0.6668	1.9484	-1.8942
6	3D ERV Inlet Flange	0.1348	0.5548	-0.5959	0.1456	0.6574	-0.6553	0.1797	0.8865	-0.8028
7	3D Pilot Valve	*N/A*	*N/A*	*N/A*	*N/A*	*N/A*	*N/A*	*N/A*	*N/A*	*N/A*
8	3D Pilot Valve	0.5269	1.8207	-1.9001	0.6146	2.0496	-2.0567	0.7492	2.2836	-2.2708
9	3D Pilot Valve	0.404	1.3387	-1.2944	0.4707	1.3665	-1.3795	0.5637	1.4758	-1.5444
10	3C ERV Inlet Flange	*N/A*	*N/A*	*N/A*	*N/A*	*N/A*	*N/A*	*N/A*	*N/A*	*N/A*
11	3C ERV Inlet Flange	3.2056	7.4117	-7.3796	3.5196	8.1761	-8.1169	4.1679	8.6697	-8.7637
12	3C ERV Inlet Flange	0.4534	0.9994	-0.9304	0.4642	1.0187	-1.1047	0.466	1.1198	-0.966
13	1B MSIV	0.0835	0.3284	-0.3687	0.0988	0.4104	-0.4331	0.1253	0.476	-0.4684
14	1B MSIV	0.0531	0.232	-0.307	0.0606	0.2553	-0.2375	0.0725	0.353	-0.4046
15	1B MSIV	0.1161	0.4251	-0.4419	0.1374	0.5252	-0.5323	0.1689	0.6616	-0.6346

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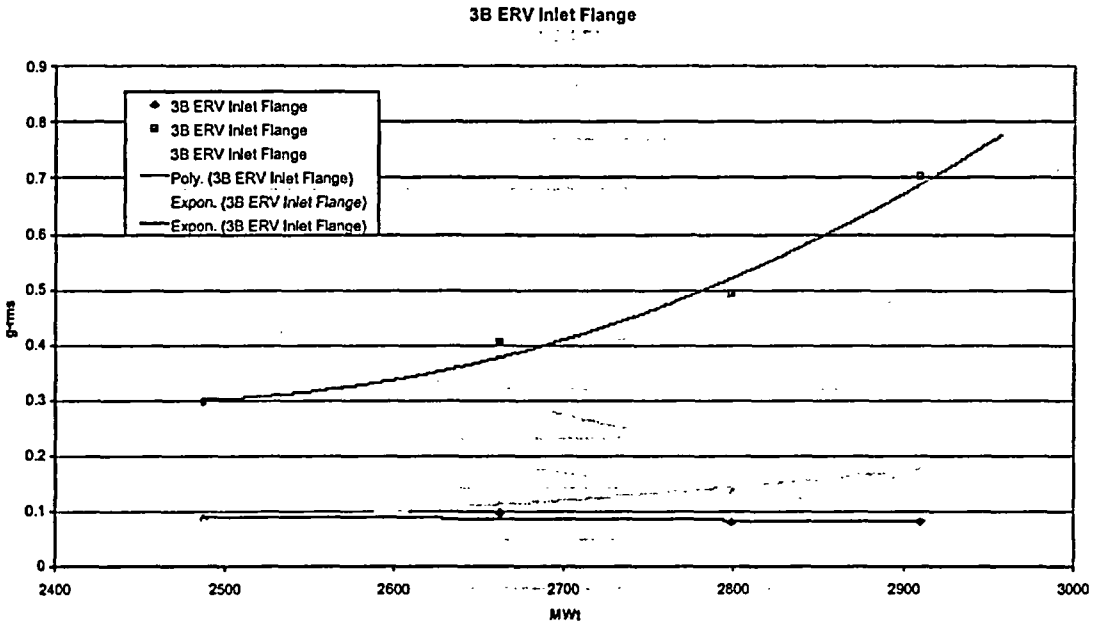
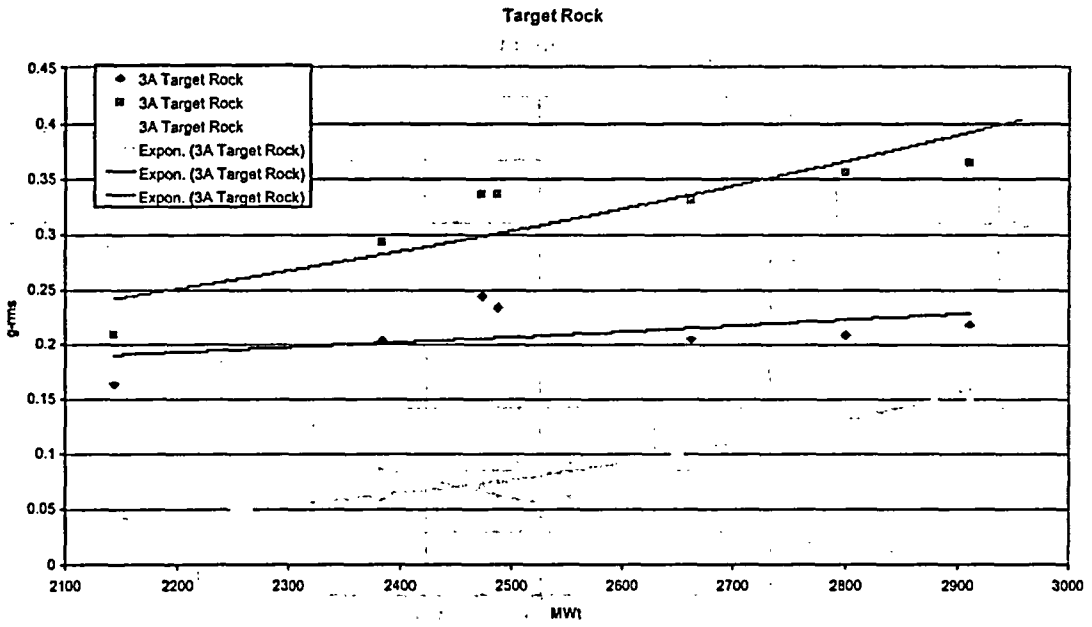
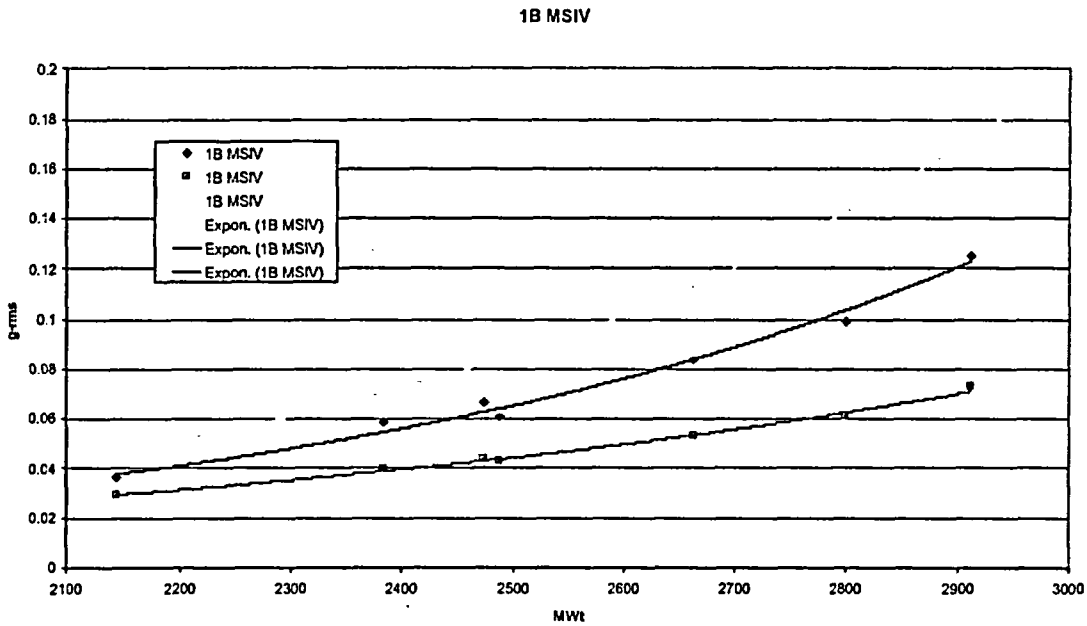
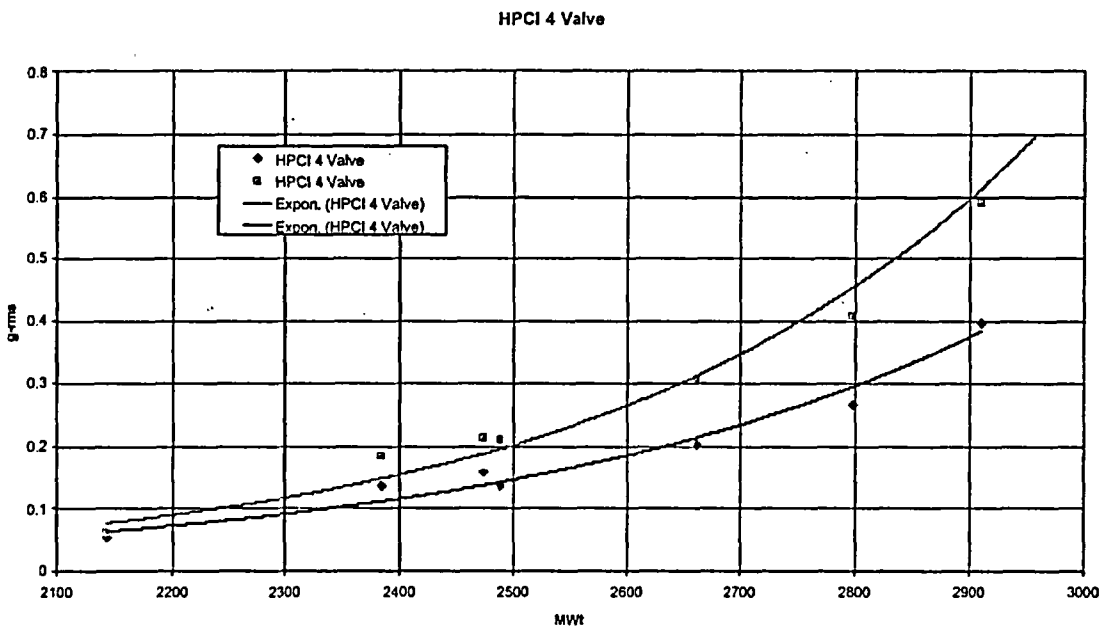
ATTACHMENT 1
Current MSL Vibration Levels – Quad Cities U1

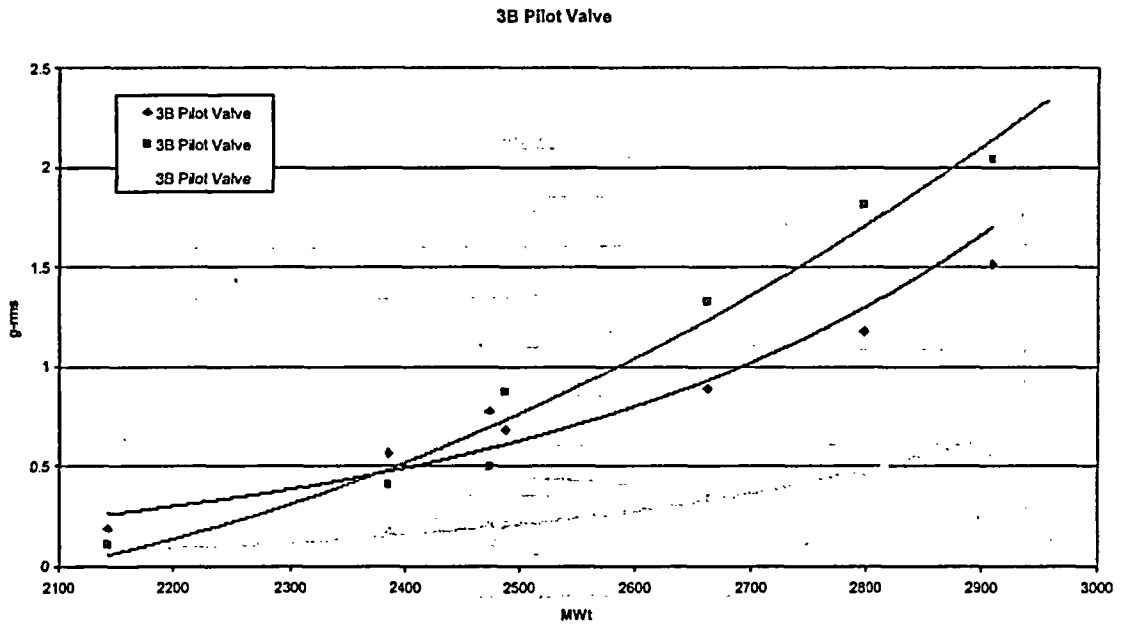


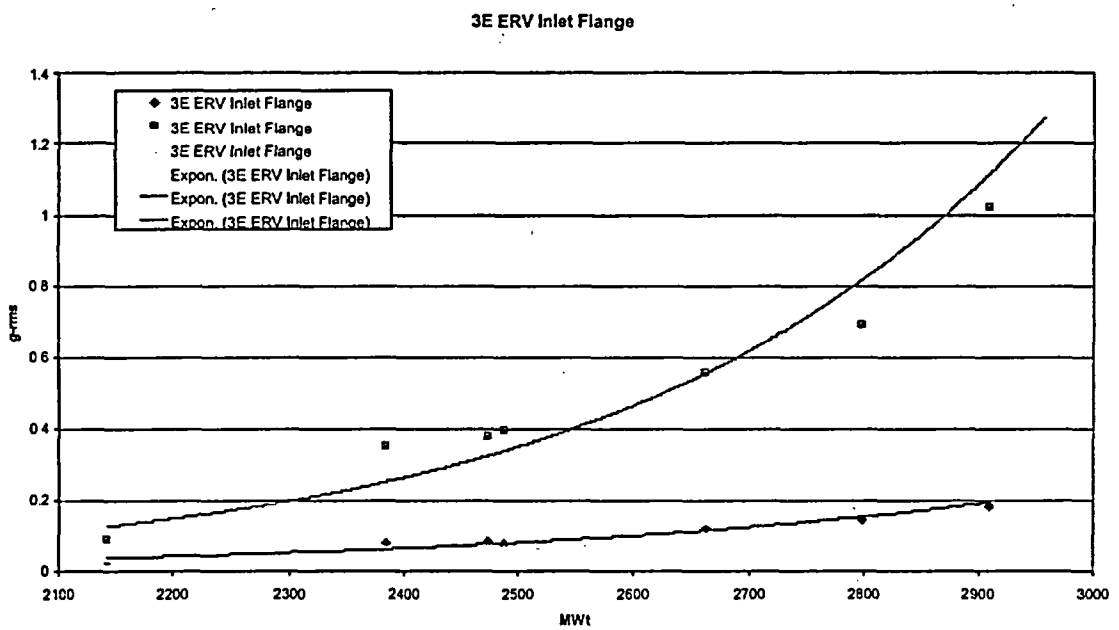


EC# 346515
ATTACHMENT 1
Current MSL Vibration Levels – Quad Cities U1






3B Pilot Valve
3B Pilot Valve
3B Pilot Valve
3B Pilot Valve
g-rms
MWt
0
0.5
1
1.5
2
2.5
2100
2200
2300
2400
2500
2600
2700
2800
2900
3000


3E ERV Inlet Flange
3E ERV Inlet Flange
3E ERV Inlet Flange
3E ERV Inlet Flange
Expon. (3E ERV Inlet Flange)
Expon. (3E ERV Inlet Flange)
Expon. (3E ERV Inlet Flange)
g-rms
MWt
0
0.2
0.4
0.6
0.8
1
1.2
1.4
2100
2200
2300
2400
2500
2600
2700
2800
2900
3000

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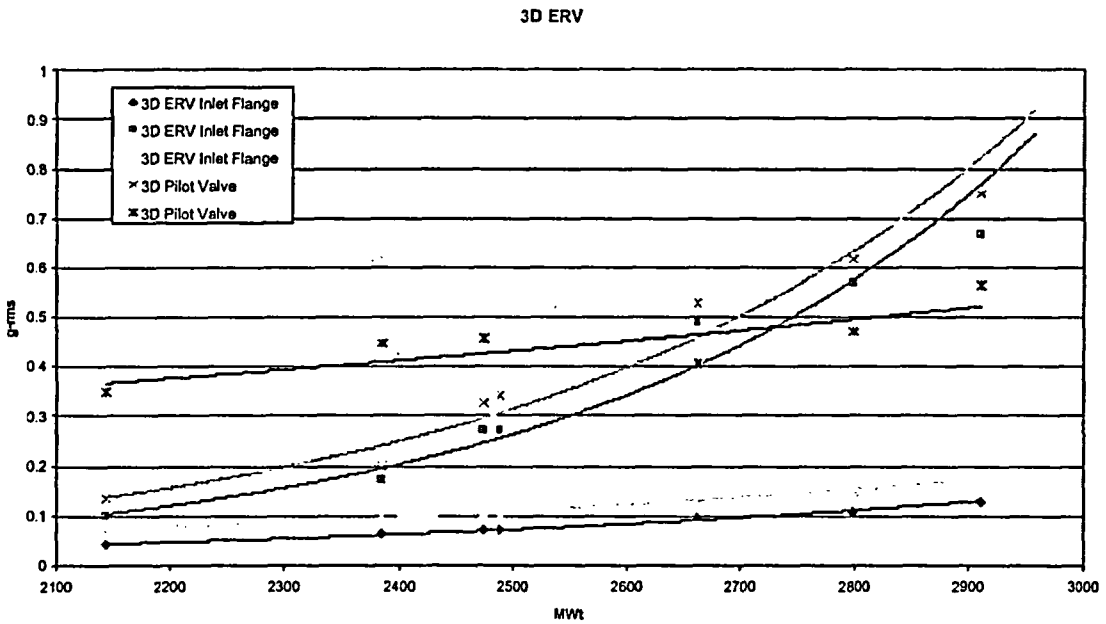
ATTACHMENT 1
Current MSL Vibration Levels – Quad Cities U1



Summary of Evaluation (Calc. No. QDC-0200-M-1360)

Component	Manufacturer	Vibe Points	Methodology / Conclusions
Inboard and Outboard MSIV Limit Switches	NAMCO EA180	DTD 13, 14, 15	a) SRSS of peak accelerations measured = 0.988g < SRSS of peak accelerations from seismic qualification test (Ref. 1) = 21.06g. b) Based on a test duration of 30.5 minutes and input acceleration varying from 7.34g to 15.07g (Ref. 1), the equivalent duration, based on a fatigue exponent of 4.3 and using the SRSS of the measured RMS accelerations (0.24g), is 12.8 years. The limit switches are mounted on a bracket; therefore, an amplification factor of 2 was applied to the RMS acceleration based on computed frequencies of the bracket. c) Using IEEE-344 as guidance, 10 cycles of strong motion effect per fifteen-second duration is considered. Using a fatigue exponent of 4.3, based on 30.5 minutes of test at input acceleration varying from 7.34g to 15.07g (Ref. 1), will yield an equivalent number of cycles of 4.1×10^9 for the SRSS of the measured RMS accelerations of 0.24g. The limit switches are mounted on a bracket; therefore, an amplification factor of 2 was applied to the RMS acceleration based on computed frequencies of the bracket. For typical steels this number of cycles is in the flat region of the S-N curve. Comparing the measured accelerations with the qualified accelerations provides reasonable assurance that at such a low level of acceleration, the corresponding alternating stress intensities will be below the fatigue endurance limit.

Component	Manufacturer	Vibe Points	Methodology / Conclusions
Inboard and Outboard MSIV Temperature Elements	Gordon 30DTCGG072B	DTD 13, 14, 15	This thermocouple is a passive element consisting of a 4.5" long X 3/16" diameter stainless steel tube containing two thermocouple wires. It is attached with a spring-loaded quick connect bayonet style cap. The lead wires are protected with fiberglass and stainless steel overbraid (20 gauge stranded). The extended mass of the thermocouple is very small (less than 1 ounce) and the cantilevered length is only 3.5". Considering the low level of accelerations (SRSS RMS acceleration = 0.24g) the alternating stress intensities in the thermocouple components will be well below their fatigue endurance limit.
MSIV LLRT Taps	_	_	See "Potential Effect of Revised Vibration Readings on Piping" attached.
Safety Valve Acoustic Monitors	NDT Inter. Inc. NDT-838-1	DTD 1, 2, 3	a) SRSS of peak accelerations measured = 4.226g < sine beat test accelerations from seismic qualification test (Ref. 2) = 6.0g. b) The SRSS of the measured RMS accelerations is 0.457g. Comparing this acceleration value with the qualified acceleration provides reasonable assurance that at such a low level of acceleration the corresponding alternating stress intensities will be below the fatigue endurance limit.

Component	Manufacturer	Vibe Points	Methodology / Conclusions
Safety Valve Temperature Elements	Gordon 72XTBGG312A0035	DTD 1, 2, 3	This pipe clamp style thermocouple is a passive element without any extended masses. The entire sensor lead is protected with fiberglass and stainless steel overbraid (20 gauge stranded). This type of construction is highly damped and no appreciable dynamic response is expected. The SRSS RMS acceleration of 0.457g is found acceptable based on the above.
RCIC Inboard and Outboard PCI Operator	Limitorque SMB-00-10	QCTD 10, 11, 12	a) SRSS of peak accelerations measured = 3.608g < SRSS peak accelerations from dynamic qualification test (Ref. 3) = 37g. b) Based on a test duration of 351 minutes and input acceleration varying from 8.25g to 37g (Ref. 3), the equivalent duration, based on a fatigue exponent of 4.3 and using the SRSS of the measured RMS accelerations (1.247g), is 3.66 years. c) Using IEEE-344 as guidance, 10 cycles of strong motion effect per fifteen-second duration is considered. Using a fatigue exponent of 4.3, based on 351 minutes of test at and input acceleration varying from 8.25g to 37g (Ref. 3) will yield an equivalent number of cycles of 1.16×10^9 for the SRSS of the measured RMS accelerations of 1.247g. For typical steels this number of cycles is in the flat region of the S-N curve. Comparing the measured accelerations with the qualified accelerations provides reasonable assurance that at such a low level of acceleration, the corresponding alternating stress intensities will be below the fatigue endurance limit.

Component	Manufacturer	Vibe Points	Methodology / Conclusions
RCIC Inboard and Outboard PCI Limit Switch	Limitorque SMB-00-10	QCTD 10, 11, 12	See evaluation for the operator above.
RCIC LLRT Taps	_	_	See "Potential Effect of Revised Vibration Readings on Piping" attached.
Pressure Switches	Barksdale B2T-A12SS B2T-M12SS	Outside Drywell QC1-OD-MS-1 (CV-3) QC1-OD-MS-1 (CV-4)	a) SRSS of peak accelerations measured = 0.206g < SRSS of peak accelerations from the seismic qualification test (Ref. 4) = 12.273g. b) Comparing the measured accelerations with the qualified acceleration provides reasonable assurance that at such a low level of acceleration the corresponding alternating stress intensities will be below the fatigue endurance limit.

Component	Manufacturer	Vibe Points	Methodology / Conclusions
MS Drain Valves Operator	Limitorque SMB-000	DTD 13, 14, 15 Note: DTD 13, 14, 15 are on MSIV. These values are increased by a factor of 3 for margin.	d) SRSS of peak accelerations measured = 2.964g < SRSS peak accelerations from dynamic qualification test (Ref. 3) = 37g. e) Based on a test duration of 351 minutes and input acceleration varying from 3.29g to 37g (Ref. 3), the equivalent duration, based on a fatigue exponent of 4.3 and using the SRSS of the measured RMS accelerations (0.72g), is 38 years. f) Using IEEE-344 as guidance, 10 cycles of strong motion effect per fifteen-second duration is considered. Using a fatigue exponent of 4.3, based on 351 minutes of test at and input acceleration varying from 3.29g to 37g (Ref. 3) will yield an equivalent number of cycles of 1.23×10^{10} for the SRSS of the measured RMS accelerations of 0.72g. For typical steels this number of cycles is in the flat region of the S-N curve. Comparing the measured accelerations with the qualified accelerations provides reasonable assurance that at such a low level of acceleration, the corresponding alternating stress intensities will be below the fatigue endurance limit.
MS Drain Valves Operator's Limit Switch	Limitorque SMB-000	DTD 13, 14, 15	See evaluation for the operator above.

References

1. Wyle Report No. 47824-2

2. Wyle Report No. 45638-1 (EQ-01Q)
3. SDRC Report No 10959
4. Report No. 1S009.0, Rev. 0 (QDC-0261-S-1296)

Potential Effect of Revised Vibration Readings on Piping

Main Steam Stop and Control Valve Small Bore Piping

Calculation QDC-3000-M-1241 prepared by Sargent and Lundy evaluates the small bore piping attached to the Main Steam Stop Valve (MSV) and Control Valve (CV). The small bore piping attached to these valves is generally only vertically supported and typically has very low frequency structural modes. The Main Steam header also has low frequency modes of the same order. The small bore piping was evaluated by exciting at each line through a forced vibration of the header pipe. The forcing vibration in each case was taken to be one or more of three significant low-frequency modes from the header piping analysis. The calculation addressed EPU vibration levels. The vibration levels that are required to cause a stress problem in the branch piping would likely cause serious overstresses in the header long before the branch. These lines are therefore expected to be acceptable based on a review of the work performed in the calculation.

HPCI LLRT Piping

Sargent and Lundy previously evaluated HPCI LLRT line 1-2369-3/4"-B. Calculation QDC-2300-M-1309 evaluates this piping. The small bore piping was evaluated by exciting the branch line through a forced vibration of the header pipe. The branch fundamental frequency is approximately 9 Hz. Based on the analysis, displacements of over ½" at the branch end in both the X- and Z-directions would be required to stress the branch above its endurance limit at this frequency. This calculation was performed for the revised configuration of the LLRT line for EPU conditions, and resulting displacements result in a stress margin of over 75%. This line is therefore expected to be acceptable when the modified configuration is installed during Q1R18.

Main Steam Drain Line Piping

The Main Steam Drain Line piping was evaluated in calculation QDC-3000-M-1275, Rev. 0 and Rev. 1. The piping in these calculations was evaluated using a broadband spectrum of 0 to 17 Hz. (Pre-EPU data showed no significant vibration above 15 Hz.) Results showed that pipe

vibration stresses with the tieback supports installed are approximately half of the level without the tieback supports installed. This piping is expected to be acceptable, provided tieback supports are properly installed.

DRF 0000-0023-4260

January 9, 2004

To: Sharon Eldridge
From: PD Knecht

Subject: Evaluation of Quad Cities Unit 1 MSIV and SRV Vibration Data

References:

1. "Assessment Scope for Potential Vibration Induced Problems Quad Cities Unit 1", Exelon Test Plan, November 26, 2003
2. "Quad Cities Nuclear Power Station Units 1 and 2 Environmental Qualification Report - MSIV Actuator", NEDC-31886P, Class III, December 1990
3. "Quad Cities Unit 1 Main Steam Line Vibration Data Reduction" , Structural Integrity Associates, January 7, 2004
4. "Quad Cities Unit 1 Assessment of high Frequency Acoustic Vibration on MSL Components", GE-NE 0000-0008-1763-02, December 2002.
5. "Target Rock Relief Valve failure to Fully Open", SIL 646, December 20, 2002
6. "Hope Creek Generating Station Unit 1 Environmental Qualification Report, Book No. S20, NEDC 30942, November 1985

Background

Quad Cities Unit 1 is currently licensed to operate at an extended power uprate value of 2957 MWt. Following two hood failures of the steam dryer assembly at Unit 2 and a hood failure and observed failure of the 3B ERV at Unit 1, operation of Unit 1 at greater than the original licensed power level (about 85% CLTP) has been limited. Unit 1 has operated for about 12 months at full generator electrical output (912 MWe) before the second failure on Unit 2 called into question whether there were other vulnerabilities at Quad Cities that required further evaluation.

Due to this concern, vibration data on various components external to the reactor pressure vessel at Quad Cities Unit 1 were taken on December 1, 2003 and evaluated at approximately 72% current licensed thermal power (CLTP), 80% and 83.7% CLTP. Due to some suspicious data additional data was obtained at about 84.1% CLTP on December 22 following replacement of certain vibration sensors. Finally on December 30, data were obtained at 90%, 94.7% and 98.7% CLTP. Following collection of these data, the unit was returned to about 85% CLTP until the data could be more closely examined.

GENE was asked in Reference 1 to evaluate the inboard MSIV on the "B" Main Steam line and the Target Rock Safety Relief Valve vibration data and provide

recommendations regarding long term operation at CLTP. This letter provides the GENE response to this request.

Methodology

Vibration data for MSIVs were compared against seismic aging qualification test data for the Unit 1 MSIV actuators (Reference 2). Engineering judgment was used to evaluate the extension of the test data to CLTP power for purpose of data collection. The seismic aging test consisted of sinusoidal variation of frequency between 25 Hz and 200 Hz at 0.75g applied to the valve bonnet. No evidence of damage was observed during these qualification tests.

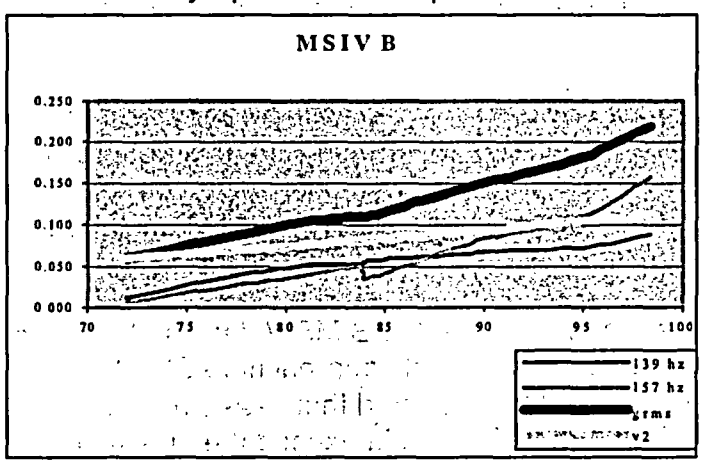
No similar data exists for the 3-stage Target Rock SRVs. However, it was assumed that seismic wear aging would have been successful for these valves at the standard IEEE 382-1980 test value of 0.75g. Furthermore, there are other BWRs that have operated at above original rated power that have not reported evidence of wear or stuck open valve events.

Evaluations

1. Main Steam Isolation Valves

B MSIV

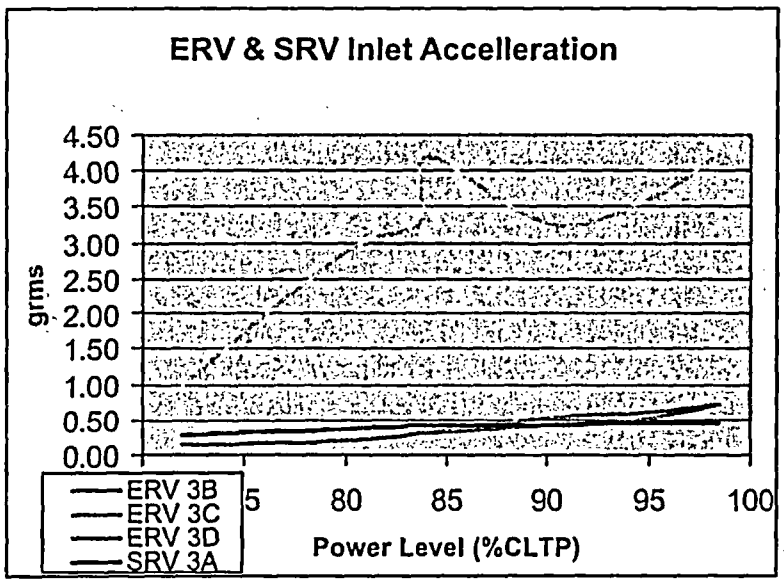
Square Root Sum of Squares (SRSS) values of the three monitored vibration axes corresponding to the peak acceleration values at the "B" MSIV stem are shown in the figure below for the dominant 139 Hz and 157 Hz frequencies during power ascension. Also shown is the total grms value from reference 3. For reference the velocity squared relationship also is shown.



As can be seen, the amplitude of the vibrations increases with power and pronounced peaks are evident above the 80% power level for the 139 Hz and 157 Hz frequencies. In all cases, the amplitudes are relatively low in comparison with the qualification basis of 0.75g and the displacement velocities are below 0.1 in/sec. No damage is expected for these conditions.

A, C and D MSIVs

Because no vibration data was obtained for the A, C or D MSIVs, the total vibration amplitudes at the ERV and SRV inlets was reviewed as shown in the figure below to determine if the other MSIVs would have a similar response to the B MSIV.



As can be seen the amplitude in the C ERV (C MSL) is substantially higher than the other 3 lines, which are quite consistent. Due to the similarity of amplitudes and geometry of the piping, the A and D MSIV likely will have a similar response and be well within the 0.75g wear aging criterion and displacement velocity will be low.

Because no significant evidence of increased wear was observed in the walkdowns, it is possible that the vibration sensor is not providing reliable data. However, it was checked prior to the data collection on December 22 and data before and after are not significantly different. It is concluded that this data could be credible, but unexplained at this time.

Based on observation of the date, the B MSIV data is about a factor of 3 lower than the B ERV data for power levels approaching CLTP. Therefore if the C ERV data were correct, it would be expected that peak rms acceleration at the C MSIV could be greater than 1 g, which would exceed the qualification criterion for the actuator. The corresponding displacement velocity at 139 Hz would be about 1.8 in/sec.

Discussion

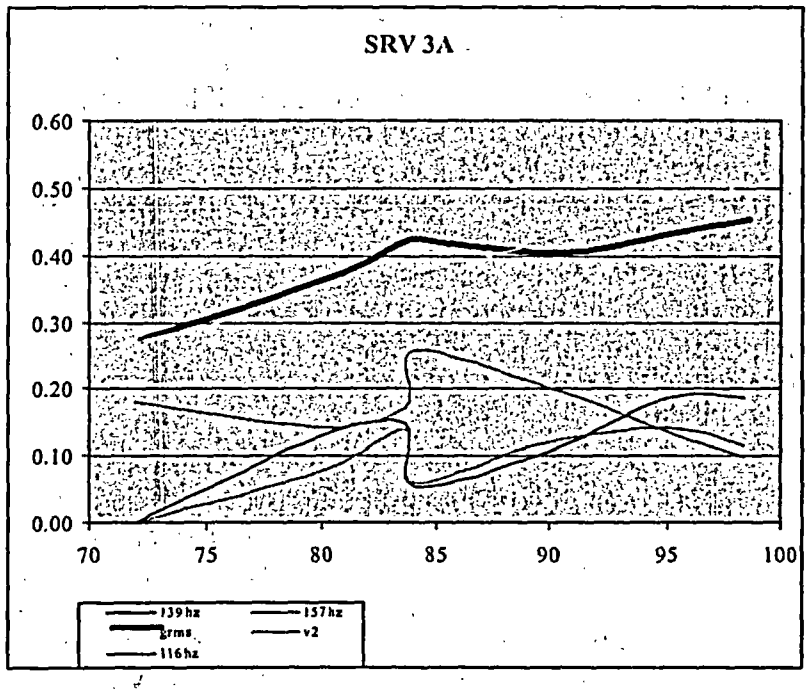
It should be noted that the wear aging criteria does not imply that the components are qualified for the full five-year recommended maintenance life of the components with that steady state vibration level. If this amount of vibration occurs, there can be long term degradation of the valve internals. The higher steam flow rate can cause agitation of the internal parts causing accelerated fretting corrosion wear in comparison with pre-EPU conditions (refer to SIL 568). Accelerated wear of other subassemblies on the MSIVs such as pneumatic control components, limit switches, electrical leads and fasteners also may not be identified though external inspection. Because this failure mode cannot be detected through external inspection, an internal inspection should be made at the next refueling outage to assess the rate of wear.

In Reference 4, the impact of MSL vibrations was discussed. Because the main disk is free to move, high vibration could potentially result in degradation of the lower disk guide liner. If significant, this could slow or possibly prevent valve motion. Although this was considered a low to medium risk in Reference 4, a mid-cycle full-stroke closure of at least the C MSIV should be considered to confirm that degradation is not occurring. This test will not provide an indication of a degradation trend, but it would confirm operability of the valves given the lack of internal inspection results.

Because no degradation of the MSIVs was shown during the walkdowns and to date operability has been demonstrated during the most recent outage, it is concluded that operation at CLTP is acceptable. However, it is recommended that individual MSL velocities be monitored during the operating cycle to detect any variations between steam lines that could be due to MSIV degradation. A full-stoke closure of the C MSIV should be considered at least once prior to the next outage to confirm operability before an internal wear inspection can be conducted. .

2. Target Rock Safety Relief Valves

Square Root Sum of Squares (SRSS) values of the three monitored vibration axes corresponding to the peak acceleration values at the "3A" Target Rock valve are shown in the figure below for the dominant 116 Hz, 139 Hz and 157 Hz frequencies during power ascension. Also shown is the total grms value from reference 3. For reference the velocity squared relationship also is shown.



The data indicates that the vibration levels at the Target Rock valve continue to increase with power although the relative strength from different measured frequencies changes. The 116 Hz frequency is unique to the A MSL and is likely the result of harmonic response with the SRV internals.

Discussion

The Target Rock valve at Quad Cities is a 3-stage design that has been replaced with a 2-stage design in later BWRs. No qualification data exist for these valves because they were qualified through the Seismic Qualification Users Group (SQUG) methodologies that did not address wear aging. Qualification aging of 2-stage valve has been successful with wear aging at the same 0.75 g acceleration in a similar manner to that which has been applied to the MSIV actuators. In comparison with the 2-stage target rock, the actuators on the 3-stage valve are lighter and it should display higher natural frequencies. Reference 6 identified natural frequencies of 128, 152 and 160 Hz in a 2-stage Target Rock valve. Based on the similarity of design and the existing Quad Cities experience at CLTP without observed degradation, it is concluded, based on engineering judgment, that there is only a low to medium risk of degradation occurring at CLTP and continuous operation is acceptable subject to monitoring for evidence of degradation.

As discussed in Reference 4 and 5, vibration could cause thread wear on the main piston/stem joint. In addition vibration could cause the pilot valve to begin to leak. Extended operation with leaking pilot valves has been shown to cause steam cutting and eventual inadvertent opening of the SRV. Because of these

concerns, the downstream thermocouples on the SRVs should be closely monitored during the next operating cycle for evidence of leakage. The location of the thermocouples should be confirmed to comply with SIL 196 supplement 11.

EC 346515
Attachment 4
SIA Assessment

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January 15, 2004
SIR-04-001
KKF-04-001

Ms. Sharon Eldridge
Lead Project Engineer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

Subject: Quad Cities Unit 1 Main Steam Vibration Assessment

Dear Sharon:

This letter report contains the Quad Cities Unit 1 main steam vibration assessment for the Electromatic Relief Valves (ERVs), main steam piping (in particular the RCIC and steam supply HPCI tie-in), the safety valves, and the HPCI-4/5 valves due to operation at Extended Power Uprate. The assessment is contained in Attachment 1.

If you have any questions, please call me.

Prepared by

Karen K. Fujikawa, P.E
Associate

Reviewed by

Kevin J. O'Hara
Associate

Approved by

Paul Hirschberg
Associate

kkf
Attachment: 1. Evaluation of Quad Cities Unit 1 Main Steam Line Vibration

Attachment: 1. Evaluation of Quad Cities Unit 1 Main Steam Line Vibration

cc: P. Hirschberg (SI)
K. J. O'Hara (SI)
M. Qin (SI)
G. L. Stevens (SI)

Evaluation of Quad Cities Unit 1 Main Steam Line Vibration

During Q1F51, the ERV 3B actuator was discovered with significant damage. Prior to this outage, Quad Cities Unit 1 (QC1) had operated at increased power (approximately 17% higher than pre-Extended Power Uprate (EPU)) for approximately 9 months at EPU. Due to damage to ERV 3B and other components, accelerometers were installed on various components of the main steam system and monitored during a variety of subsequent power levels.

Vibration measurements were taken at 30 accelerometer locations on the QC1 main steam piping at various operating levels up to 2910 MWth. Most of the accelerometers were located on the four ERVs and two adjoining pilot valves, which were expected to be the locations of high vibration. The vibration time histories were converted to frequency domain spectra and g rms (root-mean-square) values. The time histories were then filtered and maximum peak values determined. Reference [1] contains details of the vibration data reduction, and Table 1 summarizes the rms and peak accelerations for the ERVs and the Target Rock safety relief valve at 2910 MWth. Where redundant instruments were installed, the highest values are shown.

Table 1. ERV Measured Accelerations at 2910 MWth

Location	Acceleration (g)					
	X peak	Y peak	Z peak	X rms	Y rms	Z rms
ERV 3B Inlet (QCTD Ch 1, 2, 3, 15)	0.32	2.20	0.68	0.08	0.70	0.18
ERV 3B Pilot (QCTD Ch 4, 5, 6, 13, 14)	1.88	5.40	5.46	0.56	2.04	1.52
ERV 3E Inlet (QCD Ch 7, 8, 9)	0.73	3.32	0.53	0.18	1.01	0.13
ERV 3D Inlet (DTD Ch 4, 5, 6)	0.47	1.95	0.89	0.13	0.67	0.18
ERV 3D Pilot (DTD Ch 7, 8, 9)	na	2.28	1.54	na	0.75	0.56
ERV 3C Inlet (DTD Ch 10, 11, 12)	na	8.76	1.12	na	4.17	0.47
TRV 3A Inlet (DTD Ch 1, 2, 3)	1.01	3.24	0.71	0.22	0.36	0.15

Notes:
1. X is perpendicular to the main steam header, Y is vertical and Z is parallel to the steam flow.
2. na - instruments were bad
3. QCTD – Quad Cities Tape Deck
4. DTD – Dresden Tape Deck

The vibration data obtained at EPU conditions (2910 MWth) was used to evaluate the ERVs (1-0203-3B, -3C, -3D, and -3E), the main steam piping inside primary containment and its associated branch lines, HPCI-4/-5 valves, and the safety valves. The following summarizes the results of the vibration evaluation.

Evaluation of the ERV Assemblies

The four ERVs have virtually identical assemblies, which consist of the main ERV valve body, pilot valve, and solenoid actuator. The pilot valve is connected to the ERV by means of a

turnbuckle and a pilot valve tube. A tie back support has recently been added to all four valves. Each valve has small diameter leakoff piping that is routed back to the ERV discharge line. Detailed finite element analyses, using the finite element program ANSYS [3], have been completed for ERV 3B, 3C, and 3D and are documented in Reference [2].

The finite element analyses determined natural frequencies and their corresponding mode shapes. The natural frequencies and mode shapes were compared to the frequency content of the measured vibration data, and it was determined that a pilot valve pendulum mode is the primary cause of the excessive solenoid wear.

Correlation of Bump Test Data to Analytical Models

Bump test data for 3B and 3D ERV and dynamic data were obtained during plant start-up on December 22nd to 100% power (2488 MWth) and power ascension to EPU power level (2910 MWth) on December 30th. Two ERVs, 3B and 3D, were subjected to impact tests while the response was recorded at the accelerometers on the ERV and the pilot valve. The magnitude of the impact was not controlled, so the spectra plots were assessed for frequency content only. A review of the bump test data shows the frequencies of the 3B and 3D ERV models, with the tie back supports, compares well to the bump test data. Thus, the analytical models are assessed to be representative of the installed ERVs and can be used to evaluate the vibration loads and response, and effect of the tie back support.

Evaluation of Vibration Test Data

The vibration data was reviewed for amplitude versus power level to identify key frequencies and to observe amplitude versus power level trends. Two key frequencies showed up at all accelerometer locations; 139 Hz and 157 Hz. These two frequencies had very sharp ramps (similar to AC spikes) indicating low system damping. While the magnitude varied from channel to channel, these two frequencies were dominant on all channels and the magnitude generally increased proportionally to operational power level.

The measured vibrations at 139 Hz and 157 Hz are acoustic frequencies that are systemic with the steam pipe system, since they were detected at all accelerometer locations. The 139 Hz frequency is observed at all recorded power levels and amplitude increases are proportional with power level (with the exception of 3C ERV). The 157 Hz frequency is <u>not</u> observed at 2212 MWth and appears at power levels above 2384 MWth. The vibration magnitude at the ERV 3B inlet flange increases at both 139 Hz and 157 Hz, as the power increases to 2910 MWth, whereas the vibration magnitude on the ERV 3C inlet flange increases rapidly only at 139 Hz. As noted above, the two discrete frequencies have a very narrow bandwidth

Effect of the Tie Back Support

Evaluations were performed to assess the effect of the tieback support on the ERV assembly and drain piping. The primary benefit of the tie back support is to reduce amplification of accelerations at low frequencies (i.e., less than 50 Hz). Modal analyses were performed using the ERV 3B finite element model with and without the tie back support. The analytical results were found to compare well to the bump test results. Thus, the analytical model was used to

predict the natural frequencies of the ERV with and without the tie back support. A comparison was made of the calculated natural frequencies, mode shapes and modal participation factors near the predominant vibration frequencies of 139 Hz and 157 Hz for the ERV 3B both with and without the tie back support. The results of the comparison show that the addition of the tie back support does not change the dynamic characteristics of the ERV near 139 Hz and 157 Hz. Thus, at the predominant vibration frequencies of 139 and 157 Hz, the tie back support provides no significant improvement; however, there is also no negative impact.

Wear and Degradation

Historical data shows that ERV 3E actuator typically sustains more actuator wear, during pre-EPU, than the ERV 3B actuator. This was not the case in outage Q1F51. The ERV 3B actuator experienced significant wear and the valve actuator became inoperable. During the same outage, the ERV 3E actuator saw significant wear but the valve was still operable. The key difference was that the actuator spring wore through the bushing on ERV 3B, whereas the ERV 3E actuator springs were just beginning to wear into the bushings. In addition to the significant actuator wear found on ERV 3B, the drain pipe connection to the pilot valve was broken. The failure of the drain pipe was attributed to socket weld quality and cold spring.

Configuration and Turbulent Flow Assessment – ERV 3E and 3B

Main Steam Line (MSL) B has two ERV branch lines and a HPCI branch line. The ERV 3E branch line is just downstream of a MSL long radius elbow, ~2.5 diameters (MSL outside diameters, 20 inches) past the bend radii. The HPCI branch line is downstream about 5.2 diameters past the bend, and the ERV 3B branch line is about 9.6 diameters past the bend. The ERV pilot valve orientations are the same, and the pilot valve turnbuckle is normal to the MSL.

All three valves are subjected to the same steam flow rates and the turbulent flow associated with the two acoustic frequencies at 139 and 157 Hz. These acoustic modes are a result of turbulent flow separation and structural coupling with the geometry of the branch lines. The flow turbulence is approximately the same magnitude for all valves in this MSL segment, since the turbulent flow does not attenuate due to the high flow rate (typical 203 ft/sec) and the low steam density (2.24 lbs/ft^3). Therefore, all three valve branch lines experience essentially the same vibration levels.

ERV Modal Assessment – Drain Line Intact

ERV 3B was modeled to determine its modal characteristics. The results of the modal analysis determined the key mode shape that could cause this type of wear. For this mode shape, the frequency coincided with one of the predominant vibration frequencies that was identified as a pilot valve pendulum mode at 157 Hz. Since ERV 3E has similar geometry and orientation on MSL B, the modal displacements would be similar. The natural frequency near 157 Hz for ERV 3C shifts slightly and is attributed to the differences in the drain pipe routing.

ERV 3B Modal Assessment – Drain Line Failed

ERV 3B was modeled with the drain line detached to simulate the failure of the drain line connection to the pilot valve found during the Q1F51. The modal response of this valve configuration (i.e., the pilot pendulum mode) dropped to 140 Hz, which was very close to the 139 Hz acoustic mode. A dynamic assessment of this configuration shows a greater dynamic motion for this pilot valve (due to its increased flexibility) than ERV 3B with an intact drain line. Based on the measured vibration for the ERV 3B inlet flange, the vibration is 0.52 g rms at 139 Hz and 0.22 g rms at 157 Hz. Since the drain line failure causes a drop in the natural frequency of the actuator to 140 Hz, the actuator would be subjected to higher vibration levels than prior to the drain line failure.

ERV Design Modification

Based on the results of the dynamic analysis, correlation to the test data, and review of the actuator degradation, the long term recommendation is to replace the tie back support with a pilot valve modification (a reinforced plate bolted or welded to both ends of the pilot valve – see Figure 1 or some alternative which results in a significant relative stiffness change). This design modification will not eliminate the cause of the vibration but is considered economically feasible, since eliminating the acoustic vibration would entail significant main steam and branch line changes.

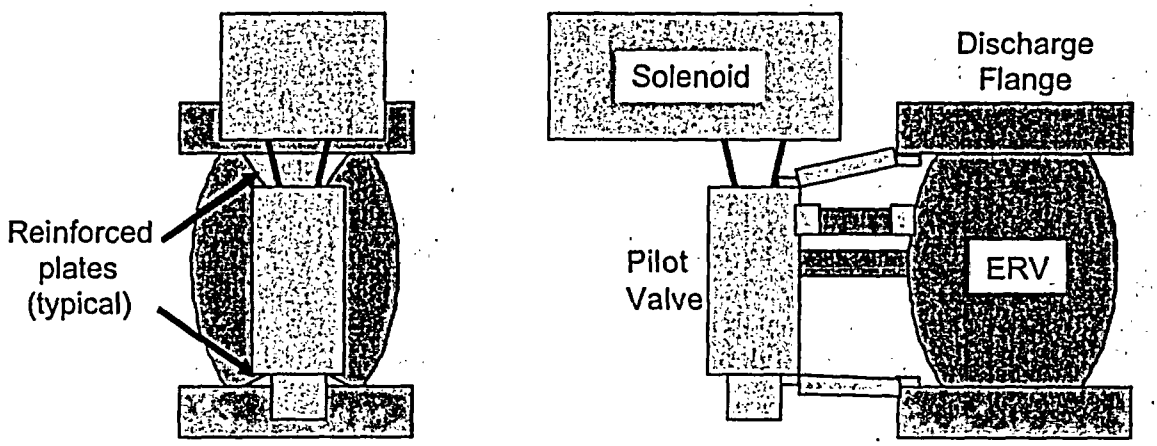

Figure 1. Plates Supporting the Ends of the Pilot Valve

Operation at 2910 MWth and Higher

The ERV 3B actuator degradation was found to be greater than ERV 3E during Q1F51. A comparison of the measured vibration for ERVs 3B and 3E shows that the vibration of both valves have predominant vibration at 139 Hz and 157 Hz. In addition, the measured vibration magnitudes at the inlet flanges are almost the same. Since their measured vibration and valve geometries are similar, their modal responses are expected to have similar pilot valve modes. Therefore, their dynamic responses are expected to be similar. This would also imply that their wear rates/degradation should also be similar. A dynamic analysis of ERV 3E should be performed to confirm its modal characteristics.

ERV Risk Assessment

The ERV 3B modal analysis indicates that the valve has increased flexibility with a failed drain line and the dynamic motion is much greater (10-20% greater). It is this additional motion that resulted in greater wear on ERV 3B than ERV 3E. If the drain line had not failed, the wear is expected to have been similar to ERV 3E.

Bushing wear/degradation is a function of time. The wear starts out slow and increases as time increases. As the wear increases, the dynamic response moves larger distances and creates larger dynamic impacts (due to limit cycle motion), which increases the wear rate. For operation up to 1600 hours, the wear at the end of this period is expected to be less than ¼ of that observed during outage Q1F51. Additionally, since the drain line has been repaired, the accelerated rate attributed to the increased pilot valve flexibility would not be a factor. Therefore, plant operation for 1600 hours at 2957 MWth would present a very low risk of wear degradation that would challenge the operation of the actuator.

Risk Assessment For ERV 3C

The ERV 3C actuator has similar modal behavior as the ERV 3B actuator. The key difference is that the ERV 3C actuator observed wear is historically low and the usual amount of wear was found in Q1F51. Additionally, the vibration data indicates low acceleration magnitudes at 157 Hz. The high vibration for ERV 3C at 139 Hz is along the Y-axis (vertical direction). This acoustic frequency would not excite the pilot pendulum mode at 157 Hz, and the vibration is normal to the mode shape motion. Therefore, this mode shape would not respond. The measured vibration in the horizontal direction is lower in magnitude. For these reasons, the ERV 3C has not seen excessive wear. Therefore, plant operation for 1600 hours at 2957 MWth would present an insignificant risk of wear degradation that would challenge the operation of the actuator.

Risk Assessment For ERV 3D

ERV 3D has a similar configuration to the other ERVs, but the pilot valve is parallel to the main steam flow. ERV 3B, 3C and 3E are all normal to the main steam flow. The pilot valve pendulum mode occurs at 159 Hz, but since the valve is oriented parallel to the main steam line, the dynamic response is not as great, which does not permit ERV 3D to wear as much. Therefore, plant operation for 1600 hours at 2957 MWth would present a very low risk of wear degradation that would challenge the operation of the actuator.

ERV Assessment Summary

Since QC1 has operated at EPU for at approximately 9 months (6400 hours at >2511 MWth), and the ERVs were demonstrated to operate in their degraded condition, QC1 can continue to operate at EPU power levels (up to 2957 MWth) for a limited time. Based on the fact that this degradation is a result of the acoustic mode vibration at 157 Hz and that this damaging frequency is still present, then ERV actuator wear is expected to continue. The wear phenomenon is time dependent, therefore, the longer the plant operates, the more wear/degradation will be observed.

For operation at 2957 MWth (slightly higher power level than the 2910 MWth level attained on December 30th), it is expected that the vibration levels would increase in proportion to main steam flow velocity squared. From the vibration trend plots [2], accelerations at the frequency of 157 Hz appears to have reached a plateau, whereas the response at 139 Hz seems to still be increasing. Therefore, it is projected that the response at 139 Hz will continue to increase.

If QC1 continues to operate at 2910 or 2957 MWth, the ERV solenoid and pilot valve vibration levels are expected to continue. Based on the analysis that has been performed to-date [2], it is expected that ERV 3B (with an intact drain line and a tie-back support) would wear no more than ERV 3E (with a tie back support) at 2910 or 2957 MWth. The wear/degradation of ERVs 3C and 3D would not be worse than the wear found during Q1F51 due to the lower measured vibration levels at these valves.

ERV Recommendations

The following recommendations are made:

- Inspect the ERVs at the next outage to quantify the wear.
- Replace the current tie back support with supports that restrain the pilot valve pendulum motion (See Figure 1). The final configuration should be determined by testing and analysis.
- During the next available plant shutdown, install accelerometers on the ERV 3C pilot valve.
- Perform vibration testing of the ERV. This should include the tie back support and any proposed modification to confirm the analytical results.
- Perform a modal analysis of ERV 3E.

Evaluation of Piping

Piping models were generated for MSLs A, B, and C [4]. These models include the main steam piping from the reactor vessel nozzle to the drywell penetration, and include the relief valve discharge piping to the drywell penetration. In addition, piping models were generated for the ERV assembly and associated leakoff piping. The piping models were generated using the computer program, PIPESTRESS [5]. The leakoff piping is very similar for the four valves. Therefore, the valve assemblies and the connecting drain piping were evaluated by applying the highest measured acceleration measured on each ERV for the appropriate MSL.

The maximum peak accelerations listed in Table 1 were applied to each piping model and stresses were calculated. The analysis was done using the response spectrum modal superposition method. The vibration loading was input as response spectra in the three orthogonal directions. The shape of the input spectra is obtained from the frequency domain acceleration curves generated in Reference [1]. Peak acceleration values are used instead of rms, because even though the peak acceleration may occur in only a small percentage of the cycles, at 139 hz they can occur often enough to apply thousands of load cycles in a relatively short time. The analysis conservatively assumes that all vibration cycles are occurring at the peak levels. Use of peak rather than rms values conservatively assumes that all vibration cycles are at the

peak value, whereas the majority of the cycles are at a much lower level. The input spectra was conservatively applied to the entire piping system.

Stresses calculated for the piping were compared to the endurance limit as specified by OM Standard Part 3 [6]. The allowable stress is 3846 psi, based on the OM-3 criteria. The maximum calculated stress for the ERV drain piping was 3214 psi, which is within the OM-3 allowable. It should be noted that the OM-3 criteria contains a number of safety factors which, when taken together, produce a very conservative result.

The evaluation for the effect of the ERV vibration on the main steam piping was done by analyzing the ERVs on MSLs A, B, and C using the PIPESTRESS program [5]. MSLB is the limiting line because it contains two ERVs, plus the HPCI tie-in is located between the ERVs. The piping model includes the large bore MSLB piping, the safety valves, ERVs 3B and 3E, the ERV discharge lines, and the HPCI line. Response spectra analyses were performed applying the maximum measured peak response at the ERV 3B pilot valve to the entire piping. Although limited acceleration data is available on the MSLs, the accelerations along the pipe are expected to be significantly less than at the pilot valve. The maximum stress is 2396 psi, at the connection of the ¾" drain line to the 3E pilot valve. All of the stresses are within the OM-3 allowable. The high stress points are all associated with the ERVs, the safety valves, or the tees to those valves. The stress at the connection to the HPCI line is 607 psi.

A piping model was created for MSLA that included a tie-in to the RCIC system. MSLA does not contain an ERV (it has the Target Rock SRV). Thus, the Target Rock 3A measured accelerations were conservatively applied to all of the piping. The maximum stress is 1317 psi, at the branch end of the tee to safety valve 4E. The same point on the branch to safety valve 4A has a similar stress. All of the stresses are within the OM-3 allowable. The high stress points are all associated with the safety valves, or the tees to those valves. The stress at the connection to the RCIC line is 1137 psi.

Another piping model was created for MSLC. Response spectra analyses were performed by applying the maximum measured peak response at the ERV 3C to the entire piping. The maximum stress is 1405 psi at the connection of the ¾" drain line to the 3C pilot valve. All of the stresses are within the O&M-3 allowable. The high stress points are all associated with the ERV, the safety valves, or the tees to those valves.

Thus, the vibrational stresses in the piping are below the OM-3 allowable, so operation at EPU power levels will not affect the integrity of the piping.

Evaluation of the Safety Valves

The safety valve evaluations are based on the measured accelerations at the ERV inlet flanges. From Table 1, the maximum horizontal (X and Z directions) measured acceleration is 1.52 g rms and the maximum vertical (Y direction) acceleration is 4.17 g rms. The safety valves are Dresser relief valves. Similar valves at another nuclear power plant have seismic qualification levels of 11 g horizontal and 9 g vertical. Based on the number of cycles that the QC1 safety valves have already experienced at these vibration levels, plus the results of the walkdown which identified no visual wear, the valves at QC1 are considered to have adequate margin.

Evaluation of the HPCI Valves

The HPCI 4 and 5 valve evaluation is based on the measured accelerations at the HPCI 4 valve. The maximum resultant horizontal measured acceleration is 0.395 g rms and the maximum vertical acceleration is 0.589 g rms. The HPCI 4 and 5 valve operators are Limitorque SMB-2-80. Equipment qualification (EQ) tests for a Limitorque SMB-2-80 were performed for another power plant that included testing at 2.61g horizontal [7, 8]. The test duration was 300 minutes versus a typical 30 second seismic test. While these test results for the Limitorque operator, that includes the limit switch, bound the test measurements at the HPCI-4 valve, walkdown results showed excessive wear at only the limit switch of the HPCI-4 valve. Operation at EPU power levels will continue to cause wear at the HPCI-4 limit switch. Thus, it is recommended to create a detailed model of the HPCI-4 valve actuator and limit switch to determine the natural frequencies of the actuator and limit switch and establish the cause of the observed wear.

Based on the number of cycles that the QC1 HPCI-4 and 5 valves have already experienced at these vibration levels, and due to the fact that the valve design has been tested demonstrating operability, it can be concluded that the HPCI-4 and 5 valves can perform their safety function.

Conclusions and Recommendations

Based on the results of the various analyses, it is acceptable to operate at 2910 MWth for a duration less than the previous plant operation at 2910 MWth. Wear will occur at the ERV actuator, but based on previous plant experience, the ERV will still be able to perform its safety function. It is expected that continued wear of the limit switch would occur at the HPCI-4 valve at EPU power levels of 2910 MWth and higher. Failure of the HPCI-4 limit switch would not prevent the HPCI-4 valve from performing its safety function, which is to close on a containment isolation signal. The limit switch is required to open the valve, but not to close it.

From the results of the main steam piping analyses, the stresses in the main steam piping are acceptable for the vibration levels measured at 2910 MWth. The accelerations measured at power levels below 2910 MWth are enveloped by the accelerations applied in the analyses. The stresses meet the criteria of O&M Part 3 [5], which contain a number of conservatisms and safety factors. The analyses performed conservatively applied peak rather than rms accelerations, and assume that the entire piping is vibrating at the levels measured at the ERVs.

At some point in the future, power levels are anticipated to reach full licensed power of 2957 MWth. It is difficult to accurately predict the accelerations at 2957 MWth, as the change is a function of both flow rate and proximity of the vortex shedding frequency to the acoustic frequency. The turbulent flow forcing function is proportional to flow velocity squared, thus at 2957 MWth, this vibration component will increase by 3.3%. The vortex shedding frequency is proportional to flow velocity, and is expected to increase from 152 Hz to 154.5 Hz. The acoustic frequencies do not change, as they are a function of geometry. The vibration data shows prominent peaks at 139 Hz and 157 Hz. As the vortex shedding frequency increases, it is expected that the 139 Hz peak and the 157 Hz peak may also increase. In the piping stress results, there is at least 20% margin to the O&M-3 allowable, and it is not expected that the

increase in vibration at 2957 MWth will exceed 20%. It is recommended that vibration data be taken at 2957 MWth to confirm this expectation.

The following recommendations are provided for the ERVs.

- Inspect the ERV actuators at the next outage to quantify the wear.
- Replace the current tie back support with two supports that restrain the pilot valve (See Figure 1). The final configuration should be determined through testing and analysis.
- During the next available plant shutdown, install accelerometers on ERV 3C pilot valve.
- Perform vibration testing of a complete ERV assembly. This should include the tie back support and any proposed modification to confirm the analytical results.
- Perform a modal analysis of ERV 3E.

References

1. Structural Integrity Associates Calculation No. QC-11Q-302, Revision 0, "Quad Cities Unit 1 Main Steam Line Vibration Data Reduction."
2. Structural Integrity Associates Calculation No. QC-11Q-301, Revision 0, "ERV Finite Element Analysis and Vibration Evaluation."
3. ANSYS/Mechanical, Revision 5.7, ANSYS, Inc., December 2000.
4. Structural Integrity Associates Calculation No. QC-11Q-303, Revision 0, "Analysis of Main Steam Piping."
5. PIPESTRESS 2000 Solver, Version 3.5.0 +67, June 2001, DST Computer Services, S.A.
6. ASME OM-S/G Standard, Part 3, 1994 Edition.
7. Structural Dynamics Research Corporation (SDRC) Report, "Dynamic Qualification Report on Two Valves," (CQD File No. 018906), dated December 24, 1981.
8. Sargent & Lundy Calculation No. CQD-027159, Revision 0, dated 12/27/85, "Fatigue Evaluation of Equipment and Instruments."

Reason For Evaluation / Scope:

Purpose
This EC EVAL, Revision 1, provides additional technical evaluation for allowing operation of Quad Cities Unit 1 up to a continuous power level of greater than 2511 megawatts thermal (MWth) up to 2957 MWth. The additional information resolves the conditional (time limit) approval for the ERVs provided in Revision 0. This revision includes the results of additional investigation on the High Pressure Coolant Injection System (HPCI) -4 valve operator limit switch. Also added is a restriction on ramp up to prolonged full power until disposition of the Unit 2 failed Target Rock Safety/Relief Valve (S/RV).

Approach
The Electromatic Relief Valve (ERV) condition found was further evaluated by extensive testing at the Wyle Laboratories facility. This testing confirmed the failure mode, provided input to recommended corrective actions and validated the acceptability of the corrective actions.

The HPCI -4 valve degradation issue was further evaluated by performing calculations to determine if any resonance response of the four rotor internals was likely, given the Quad Cities Unit 1 measured vibrations. The calculations are referenced under this EC.

The as-found testing of the Target Rock S/RV from Unit 2 resulted in a setpoint lift at +6.8% of allowable value versus a code acceptance criterion of +3% and the Technical Specifications criteria of 1%. The failure is documented in CR 215874. Until the cause of this failure is determined and the Extent of Condition (EOC) as it relates to Unit 1 can be dispositioned, ramp up of Unit 1 to power levels above 2511 MWth, except for brief periods of data collection is restricted.

Results
All necessary actions to ensure full cycle, EPU power level operation of the evaluated components are completed or are scheduled for completion prior to prolonged power operation above 2511 MWth (previous, pre-EPU, 100% power). The open required actions are re-listed from Revision 0, for clarity, at the end of this Revision. Added to the prior actions is the disposition of the EOC for the Unit 2 Target Rock S/RV as-found setpoint test failure.

Detailed Evaluation:

Acceptability of ERV Component Operation at EPU Levels

Extensive testing performed at Wyle Laboratories provided clarification to the vibration response of the ERV actuators. This testing confirmed that the internals

of the ERV actuator were responding to some of the input frequencies in a manner that resulted in premature degradation of those components. Specifically, the plunger support spring, guide rod, and bracket bushing exhibited early wear. These components have been redesigned by utilizing harder materials and special edge treatments to ensure acceptable long-term performance. The modified configuration was then subjected to endurance testing to simulate in plant service and was determined to resolve the degradation concern. The revised configuration will be installed prior to any prolonged full EPU power operation under EC 347763. Detailed documentation of the testing and analysis of results can be found under EC 343933 (Unit 2 modification for Power-Operated Relief Valve (PORV) to ERV change – [installed]).

Evaluation of HPCI 4 Valve Operator

Based on walkdown data, the limit switch (four rotor design) internal to the HPCI -4 valve operator was found degraded. Additional analyses of the operator assembly have been performed to determine any susceptible internal components. No individual susceptibility was identified through these analyses. These evaluations (see References 4 and 5) and the as-found condition of the Unit 2 HPCI -4 rotor assembly (discussed in EC 348316) provide the necessary assurance that full cycle operation at EPU power is acceptable for this valve operator assembly. Confirmatory testing is planned as detailed under Action 9 below.

Conclusions / Findings:

This EC EVAL provides an engineering evaluation supporting operation above 2511 MWth up to 2957 MWth for long-term operation providing all required actions under AR 194877 are completed in the required timeframe. It has been determined that this operation will not result in unacceptable degradation levels of any components. The conclusion is provided by evaluation of the additional information on the ERVs and HPCI -4 valve as discussed above.

References:

1. EC 347763, Modification to remove tie-back and upgrade components on ERVs, Unit 1 [In Progress]
2. EC 343933, Rev. 0, Replacement of PORVS with ERVs, Unit 2
3. EC 348316, Rev. 0, Evaluation of Unit 2 Components.
4. QDC-2300-M-1372, Rev. 0, Parametric Analysis of Limitorque Limit Switch Vibration
5. QDC-2300-M-1373, Rev. 0, Evaluation of MOV 1-2301-4 Limitorque limit switch for Vibration effects
6. Independent review by MPR dated April 22, 2004 (copy attached here)

MPR
ASSOCIATES INC.
ENGINEERS

April 22, 2004

Ms. Sharon Eldridge
Lead Project Engineer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

Subject: Review of EC-EVAL #346515, "Evaluation of Quad Cities Main Steam Line Vibrations at EPU Power Levels," Revisions 0 and 1

Dear Ms. Eldridge:

Exelon recently provided MPR with EC-EVAL #346515, "Evaluation of Quad Cities Main Steam Line Vibrations at EPU Power Levels," Revisions 0 and 1, for review. The EC-EVAL summarizes technical analyses that assess the impact of operating Quad Cities Unit 1 (QC1) at power levels up to 2957 MWth. MPR has completed a review of the EC-EVAL and concluded that Exelon has taken effective compensatory actions and completed analyses to demonstrate that components susceptible to accelerated vibration damage will maintain their integrity to the end of the current operating cycle. In addition, Exelon has imposed a restriction on operating QC1 at power levels above 2511 MWth until emergency relief valves are modified with wear resistant components, inspections are completed of critical components (i.e., snubber brackets, HPCI valve limit switches and ERV actuators), and instrumentation is installed to acquire vibration data on the 1C Main Steam Isolation Valve. This restriction is warranted and completing the actions will provide adequate margin to operate QC1 at power levels above 2511 MWth.

Our review of the evaluation for Dresden Unit 3 is ongoing. We will contact you early next week with our comments or conclusions. Please call me or Bill McCurdy if you have any questions regarding this letter.

Sincerely,

Phillip J. Rush, P.E.

320 KING STREET ALEXANDRIA, VA 22314-3230 703-519-0200 FAX: 703-519-0224 http:\\www.mpr.com

McCurdy if you have any questions related to our review efforts or this letter.

Sincerely,

Phillip J. Rush, P.E.

For Information Only:

AR 194877 Required Actions:

Prior/During Planned Outage Prior to Above 2511 MWth Operation

1. Install accelerometers on the 1C MSIV during the planned shutdown following 3500 hours of operation. Evaluation of data obtained will be used to define the need for the inspection required in Action 14 below. (AR 194877-05)
2. Install modification EC 347763 to upgrade the ERV actuator to resist vibration effects. (AR 194877-07)
3. Inspect the components identified during Q1F51 as degraded. These include the ERV actuators, HPCI 4 valve operator and snubber mounting brackets. (AR 194877-10)
4. Disposition the extent of condition of the Unit 2 Target rock valve as found test failure under CR 215874 associated Action Tracking.

After Return to Full EPU Power Operation:

5. Obtain a vibration data set monthly, after return to full power operation after any down power of greater than 10%, and assess for variation/deviation from the analyzed data and any negative impacts. (AR 194877-03)
6. Monitor weekly individual MSL flows to provide any anomalies that indicate MSIV degradation. (AR 194877-11)
7. Obtain a full set of vibration data when the unit is moved to full thermal power of 2957 MWth for the first time. This data will be used to confirm the assumptions made on expected maximum vibration levels. (AR 194877-08)
8. Evaluate the current PM scope and frequency for all components included in this evaluation. The expectation is that some components will need increased PM frequencies for rebuilds, refurbishments and/or replacements. (AR 194877-12)
    a. Consideration should be given to perform a post-EPU baseline inspection during the Q1R18 outage.
9. Perform testing on the NAMCO limit switches using the Quad Cities specific vibration predominate frequencies and amplitudes. This testing to be performed prior to approval of full cycle operation. (AR 194877-13)
10. Perform testing on the Limitorque actuator type and size SMB-2-80 or equivalent using the Quad Cities specific vibration predominate frequencies and amplitudes. This testing to be performed prior to approval of full cycle operation (AR 194877-14)
11. Evaluate the snubber inspection plan to ensure that sufficient MSL snubbers are functionally tested during Q1R18 to ensure that degradation is not occurring. Inspections should include the previously

degraded snubbers, 1-66 and 1-71. Adjust inspection plan as necessary. (AR 194877-10)

12. Evaluate the HPCI4 valve limit switch to determine if a mod is justified to reduce wear due to vibration after completion of the testing from action 9. (AR 194877-15)
13. Complete comparison of Quad Cities Unit 1 to Dresden Unit 3. This comparison may provide further insights into changes that can be made for Quad Cities to minimize the measured vibration level responses such that accelerated component degradation does not occur. This evaluation includes the following elements:
    a. Expansion of the ongoing MSL circuit analyses and/or scale model testing to include the frequency range up to 180 Hz. This may provide insight into the source of the measured 139 and 157 Hz predominate frequencies.
    b. Detailed configuration differences in all MS line branch line connections between Dresden and Quad Cities.
    c. Detailed evaluation of the ERV configuration installed at Dresden.
    d. A comparison of steam flows between Dresden and Quad Cities on MSL and all branch connections during EPU power level operation.

<u>During Q1R18</u>

14. If results from vibration data on the 1C MSIV warrant, inspect the 1C MSIV; during the next refuel outage (Q1R18) to confirm that no degradation has occurred. If degradation is found, evaluate the need for expanding inspection to other MSIVs.
15. Perform any additional inspections as determined by evaluations performed in actions above.

EC 346515, Revision 2
Evaluation of Quad Cities Unit 1 Main Steam Line Vibrations at Extended Power Uprate (EPU) Power Levels

Reason For Evaluation / Scope:

Purpose

Revision 2 provides information on the turbine control valve (TCV) pressure switch acceptability (Reference 7) and the acceptability of Electromatic Relief Valves (ERVs) with pilot valve tiebacks for operation up to 2511 megawatts thermal (MWth) with short term excursions to 2957 MWth for data collection (≤ 72 hours above 2511 MWth).

Revision 1 provided additional technical evaluation for allowing operation of Quad Cities Unit 1 up to a continuous power level of greater than 2511 MWth up to 2957 MWth. The additional information resolves the conditional (time limit) approval for the ERVs provided in Revision 0. This revision includes the results of additional investigation on the High Pressure Coolant Injection System (HPCI) -4 valve operator limit switch. Also added is a restriction on ramp up to prolonged full power until disposition of the Unit 2 failed Target Rock Safety/Relief Valve (S/RV).

Approach

The ERV condition found was further evaluated by extensive testing at the Wyle Laboratories facility. This testing confirmed the failure mode, provided input to recommended corrective actions and validated the acceptability of the corrective actions.

The HPCI -4 valve degradation issue was further evaluated by performing calculations to determine if any resonance response of the four rotor internals was likely, given the Quad Cities Unit 1 measured vibrations. The calculations are referenced under this EC.

The as-found testing of the Target Rock S/RV from Unit 2 resulted in a setpoint lift at +6.8% of allowable value versus a code acceptance criterion of +3% and the Technical Specifications criteria of 1%. The failure is documented in CR 215874. Until the cause of this failure is determined and the extent of condition (EOC) as it relates to Unit 1 can be dispositioned, ramp up of Unit 1 to power levels above 2511 MWth, except for brief periods of data collection is restricted.

Steam dryer issues have impacted station performance. Ongoing analyses of the Unit 1 and Unit 2 steam dryers will dictate future actions and are being addressed separate from this EC.

Results

All necessary actions to ensure full cycle, EPU power level operation of the evaluated components are completed or are scheduled for completion prior to prolonged power operation above 2511 MWth (previous, pre-EPU, 100% power). The open required actions are re-listed from Revision 0, for clarity, at the end of

this Revision. Added to the prior actions is the disposition of the EOC for the Unit 2 Target Rock S/RV as-found setpoint test failure.

Revision 2 documents that the turbine control valve (TCV) pressure switches are acceptable for EPU operation (Reference 7) and that ERVs with pilot valve tiebacks are acceptable for pre-EPU levels with short duration power ascension (≤ 72 total hours above 2511 MWth) for the purpose of data collection based on Reference 8 and on vibration degradation being time dependent.

Detailed Evaluation:

Acceptability of ERV Component Operation at EPU Levels
Extensive testing performed at Wyle Laboratories provided clarification to the vibration response of the ERV actuators. This testing confirmed that the internals of the ERV actuator were responding to some of the input frequencies in a manner that resulted in premature degradation of those components. Specifically, the plunger support spring, guide rod and bracket bushing exhibited early wear. These components have been redesigned by utilizing harder materials and special edge treatments to ensure acceptable long-term performance. The modified configuration was then subjected to endurance testing to simulate in plant service and was determined to resolve the degradation concern. The revised configuration will be installed prior to any prolonged full EPU power operation under EC 347763. Detailed documentation of the testing and analysis of results can be found under EC 343933 (Unit 2 modification for Power-Operated Relief Valve (PORV) to ERV change – [installed]).

Evaluation of HPCI -4 Valve Operator
Based on walkdown data, the limit switch (four rotor design) internal to the motor-operated valve (MOV) 1-2301-4 valve operator was found degraded. Additional analyses of the operator assembly have been performed to determine any susceptible internal components. No individual susceptibility was identified through these analyses. These evaluations (See References 4 and 5) and the as-found condition of the Unit 2 HPCI -4 rotor assembly (discussed in EC 348316) provide the necessary assurance that full cycle operation at EPU power is acceptable for this valve operator assembly. Confirmatory testing is planned as detailed under Action 10 below.

Evaluation of Turbine Control Valve Pressure Switch
Based on seismic testing acceleration levels being significantly higher than operational vibrations, the turbine control valve pressure switch is judged to be acceptable for EPU levels (Reference 7).

Evaluation of ERVs with Pilot Valve Tiebacks

The QC1 ERV measured vibration responses (with tieback supports) are much lower than the Wyle Laboratories test input vibration (Reference 8). Therefore, QC1 ERV actuator wear will be much lower than that observed during the Wyle Laboratories actuator aging tests. Based on this determination and on vibration degradation being time dependent, ERVs with the pilot valve tiebacks are acceptable for pre-EPU (≤ 2511 MWth) levels with short duration power ascension (≤ 72 total hours above 2511 MWth) for the purpose of data collection.

Conclusions / Findings:

This EC EVAL provides an engineering evaluation supporting operation above 2511 MWth up to 2957 MWth for long-term operation provided all required actions under AR 194877 are completed in the required timeframe. It has been determined that this operation will not result in unacceptable degradation levels of any components.

Revision 2 also documents that the main steam line equipment, including ERVs with tiebacks, are acceptable for pre-EPU levels with short duration power ascension (≤ 72 total hours above 2511 MWth) for the purpose of data collection. The conclusion is provided by evaluation of the additional information on the ERVs, HPCI -4 valve and TCV pressure switch as discussed above.

References:

1. EC 347763, Modification to Remove Tieback and Upgrade Components on ERVs, Unit 1 [In Progress]
2. EC 343933, Rev. 0, Replacement of PORVS with ERVs, Unit 2
3. EC 348316, Rev. 0, Evaluation of Unit 2 Components.
4. QDC-2300-M-1372, Rev. 0, Parametric Analysis of Limitorque Limit Switch Vibration
5. QDC-2300-M-1373, Rev. 0, Evaluation of MOV 1-2301-4 Limitorque limit switch for Vibration effects
6. Independent review by MPR dated April 22, 2004 (copy attached below)
7. QDC-0200-M-1360, Rev. 1, Evaluation of Components for Vibration Effects Due to Power Uprate Quad Cities Unit 1
8. SIR-04-054, Revision 1, May 4, 2004, Quad Cities Unit 1 ERV Vibration Assessment (with tie-backs) (copy attached below)

MPR
ASSOCIATES INC.
ENGINEERS

April 22, 2004

Ms. Sharon Eldridge
Lead Project Engineer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

Subject: Review of EC-EVAL #346515, "Evaluation of Quad Cities Main Steam Line Vibrations at EPU Power Levels," Revisions 0 and 1

Dear Ms. Eldridge:

Exelon recently provided MPR with EC-EVAL #346515, "Evaluation of Quad Cities Main Steam Line Vibrations at EPU Power Levels," Revisions 0 and 1, for review. The EC-EVAL summarizes technical analyses that assess the impact of operating Quad Cities Unit 1 (QC1) at power levels up to 2957 MWth. MPR has completed a review of the EC-EVAL and concluded that Exelon has taken effective compensatory actions and completed analyses to demonstrate that components susceptible to accelerated vibration damage will maintain their integrity to the end of the current operating cycle. In addition, Exelon has imposed a restriction on operating QC1 at power levels above 2511 MWth until emergency relief valves are modified with wear resistant components, inspections are completed of critical components (i.e., snubber brackets, HPCI valve limit switches and ERV actuators), and instrumentation is installed to acquire vibration data on the 1C Main Steam Isolation Valve. This restriction is warranted and completing the actions will provide adequate margin to operate QC1 at power levels above 2511 MWth.

Our review of the evaluation for Dresden Unit 3 is ongoing. We will contact you early next week with our comments or conclusions. Please call me or Bill McCurdy if you have any questions regarding this letter.

Sincerely,

Phillip J. Rush, P.E.

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May 4, 2004
SIR-04-054 Revision 1
KKF-04-025

Ms. Sharon Eldridge
Lead Project Engineer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

Subject: Quad Cities Unit 1 ERV Vibration Assessment

Dear Sharon:

This letter report contains the Quad Cities Unit 1 main steam vibration assessment for the Electromatic Relief Valves (ERVs), with standard brass bushings, due to operation at 2511 MWe (pre-EPU) operation for the next 15 months. The assessment is contained in Attachment 1.

If you have any questions, please call me.

Prepared by

Kevin J. O'Hara
Associate

Reviewed by

Karen K. Fujikawa, P.E.
Associate

Approved by

Karen K. Fujikawa
Associate

kkf
Attachment: 1. Evaluation of Quad Cities Unit 1 ERV Vibration Assessment

cc: QC-16Q-102, -403
P. Hirschberg (SI)
K. J. O'Hara (SI)
G. L. Stevens (SI)

Evaluation of Quad Cities Unit 1 ERV Vibration Assessment

1.0 Introduction

The Quad Cities Unit 1 (QC1) 3B Electromagnetic Relief Valve (ERV) actuator was discovered with significant wear/damage during outage Q1F51. This ERV was exposed to 9 months of EPU (2910 MWth) operation and during EPU operation the ERV 3B pilot-valve vent pipe failed. The ERV 3B failure analysis indicated that this vent pipe failure caused the excessive bushing wear to ERV 3B. Prior to EPU operation (January to September, 2003), all QC1 ERVs operated at pre-EPU (2511 MWth) power levels without excessive actuator bushing wear.

The purpose of this letter report is to evaluate QC1 ERV actuators, with standard brass bushings and a tie-back support, for operation at pre-EPU power levels up to 2511 MWth for 15 months and operation at EPU power levels for ~2 days.

2.0 Analytical and Data Assessments

Four assessments will be performed to assess QC1 ERV actuator bushing survival capability. These assessments consist of:

- A review of the previous wear assessment from Reference [1], section 5
- A review of the dynamic vibration magnitudes at both pre-EPU and EPU power levels from Reference [2]
- A comparison of vibration and wear rates observed during the Wyle Labs ERV and actuator tests from Reference [3]. This comparison will be both with and without tie-back supports, and
- A correlation of all data with proposed QC1 operation.

2.1 QC1 Actuator Wear Assessments

The following evaluations were based on ERV 3D photographs after 5 months of EPU (2910 MWth) operation and visual inspection of QC1 ERVs after 24 months of pre-EPU (2511 MWth) operation. This information consists of:

1) Color photographs of the 3D valve actuator bushing wear
2) Visual inspections of ERVs by QC1 valve and maintenance engineers during standard maintenance programs that were conducted during previous outages.

The observations below determine magnitude of the wear and associated wear patterns in order to assess typical wear that would seen after 24 months of pre-EPU operation.

<u>ERV 3D – Typical EPU Wear</u>

Two color photographs show typical bushing wear after 5 months of EPU operation. This wear can be seen in each photograph (Figure 2). The wear shows ~2 mm elongation of the bushing holes. Although some wear was present, this valve was operable at the end of EPU operating cycle and was successfully actuated in this condition.

<u>ERV 3E and 3B – Pre-EPU Wear</u>

QC1 valve and maintenance engineers perform standard inspections on all ERVs after 24 months of pre-EPU operation. Post-inspection results of ERV 3E and 3B actuator bushings (both valves are located on main steam line (MSL) B) are as follows:

- Typically ERV 3E sustains more actuator wear than ERV 3B
- Wear typically consists of some minor hole elongation (less than 2 mm) and the bushing-to-spring contact surface has several wear marks (some shallow gouges from the retention spring)
- During the last several pre-EPU operating cycles, QC1 ERV 3E and 3B had less than 2 mm of wear. Based on the Wyle Labs testing these valves were operable.

ERV 3C and 3D – Pre-EPU Wear

Inspection of ERV bushings, after 24 months of pre-EPU operation, show wear similar to ERV 3E and 3B.

- Wear patterns are consistent with ERV 3E and 3B, but the magnitude of the wear is substantially less (based on visual inspections)
- During the last several pre-EPU operating cycles, QC1 ERV 3C and 3D had less than 2 mm of wear. Based on the Wyle Labs testing these valves were operable.

2.2 Quad Cities 1 Dynamic Data Assessment

Vibration data was acquired at QC1 from December 22 through December 30, 2003. All data was assessed for vibration on all four main steam line (MSL) ERVs, their associated pilot valves and other MSL components. These components were assessed for six power levels, from 2212 MWth up to 2910 MWth (EPU). It is important to note that the ERVs had tie-back supports when this data was acquired. A data summary of the peak and RMS vibration levels at 2488 and 2910 MWth is documented in Reference [2], and ERV accelerations are tabulated in Tables 1 and 2.

Based on testing at Wyle Labs (section 2.3 below), the bushing wear is the result of a local actuator mode [3]. Only during x-axis vibration did this mode become excited sufficient enough to cause the observed wear. Therefore, all the ERV frequency spectra for the x-axis (parallel to pilot valve turnbuckle) vibration were reviewed for vibration magnitude, frequency content and actuator excitation that could potentially result in excessive actuator wear. The power level of interest is pre-EPU (2511 MWth) power level which is 8.5% higher (based on flow velocity squared) than 2488 MWth (power level of measured data). A scale factor of 1.10 was conservatively applied to the 2488 MWth vibration levels to approximate the vibration levels at 2511 MWth.

Inspection of the spectra plots for ERV 3B, 3D and 3E inlet flanges indicate most of the energy is amassed at the discrete frequency of 139 or 157 Hz. It should be noted that ERV 3C x-axis (ch 10) is bad. The QC1 ERV valve inlet measured random and sine amplitudes are:

ERV 3B Spectral response (QCTD Ch 1)
- x-axis (vertical to MS flow): 0.086 grms with a superimposed sine of 0.064 g
- x-axis (scaled by 1.10): 0.095 grms with a superimposed sine of 0.070 g

ERV 3C Spectral response (DTD Ch 10)
- x-axis (vertical to MS flow): bad data for this channel
- x-axis (scaled by 1.10): bad data for this channel

ERV 3D Spectral response (DTD Ch 6)

- z-axis (parallel to MS flow): 0.106 grms with a superimposed sine of 0.042 g
- z-axis (scaled by 1.10): 0.117 grms with a superimposed sine of 0.047 g

ERV 3E Spectral response (QCTD Ch 7)

- x-axis (vertical to MS flow): 0.078 grms with a superimposed sine of 0.071g
- x-axis (scaled by 1.10): 0.086 grms with a superimposed sine of 0.078g

The maximum composite vibration (0-200 Hz) is on ERV 3D inlet flange (0.117 grms). Additionally, spectral energy in the 30-70 Hz range for ERV 3D is very low (less than 0.0065 grms). The only observed frequency content with any magnitude is at 23 Hz, which is outside the local modes of vibration observed in the Wyle Labs testing. Therefore, the QC1 vibration levels are low at 2511 MWth and would not produce much, wear after 15 months of operation.

2.3 Wyle Labs ERV and Actuator Test Results

Several tests were performed at Wyle Labs. Initial testing was performed with the tie-back (test configuration 1) and without the tie-back (test configuration 2). The random vibration testing for both configurations showed the maximum response to be higher with the tie-back:

- ERV with tie-back: test run 7 - 3X=0.92 grms, 3Y=5.57 grms, and 3Z=0.61 grms
- ERV w/o tie-back: test run 23 - 3X=0.38 grms, 3Y=0.67 grms, and 3Z=0.19 grms
- Plunger w/tie-back: test run 8 - 6X=3.37 grms, 6Y=1.53 grms, and 6Z=1.80 grms
- Plunger w/o tie-back: test run 24 - 6X=0.95 grms, 6Y=0.87 grms, and 6Z=0.58 grms

The ERV solenoid and plunger responses were higher with the tie-back. All ERV test results indicated that the most severe actuator wear came from x-axis vibration. Therefore, all actuator-only aging tests were performed in the x-axis only.

Actuator-only aging tests were complete at Wyle Labs towards the end of the test series (test cases 45-55). Test series 6 (test cases 45, 46 and 49) are relevant, since they represent actuator-only testing with the original brass bushings. These tests consisted of resonance search (6.A.X) and actuator aging tests (6.B.X). The resonance search identified a 62 Hz plunger mode that was active within the actuator, but the plunger also seemed to respond between 30-70 Hz.

The Wyle aging test consisted of two parts:

1) Test Case 46: low frequency 30-70 Hz at 0.8 grms for 49 minutes and
2) Test Case 49: high frequency 20-200 Hz at 0.8 grms for 3 hours

Post-test results showed that the Wyle Test Actuator bushing wear was very similar to ERV 3D bushings that experienced five months of EPU operation at QC1. Both sets of bushings had the same amount of wear (~2 mm, measured) and similar wear patterns.

3.0 QC1 Data Correlation and ERV Assessment

QC1 is planning to operate at pre-EPU (2511 MWth) for the next 15 months with ~2 days of operation at EPU (2910 MWth). Three assessments have been performed to understand the impact of operation at each power level and the duration.

EPU Operational Assessment

- QC1 has operational history at EPU power level for 9 months
- Wyle Labs actuator aging tests (with brass bushings) showed ~2 mm of wear at 0.8 grms
- Detailed inspections of ERV 3D, after 5 months of EPU operation with no tie back, revealed ~2 mm of wear, and the valve passed outage functional tests.
- It is clear that the Wyle Labs testing was equivalent to 5 months of EPU operation, based on the magnitude of wear and similar wear patterns. The valve/actuator passed all post-test functional checks.

Based on this operational history and the observed wear, operation for 2 days at EPU power should not show any significant wear, due to the limited time spent at this power level.

Pre-EPU Operational Assessment

- QC1 has several periods of operational history at pre-EPU power level, where all ERVs were exposed to this power level for 24 months
- Outage inspections of ERVs that were exposed to 24 months of pre-EPU operation, revealed less than 2 mm of wear, and all valves were operable
- Results from the Wyle Labs testing showed that ERV solenoid and plunger responses were higher with the tie-back than without the tie-back.
- Comparison of the QC1 per-EPU vibration data (ERVs had tie-backs) with Wyle Labs input spectra showed that the QC1 vibration data was much lower.

Based on this operational history, the lower vibration environments and the observed wear, QC1 operation for 15 months at pre-EPU power should exhibit less than 2 mm of wear (estimated to be ~1 mm). This includes the effect of the ERV with pilot-valve actuator tie-back (which is inherent in the QC1 ERV vibration data).

4.0 Conclusions

The QC1 ERV responses (with tie-back supports) are much lower than the Wyle Labs test input vibration, therefore, QC1 ERV actuator wear will be much lower than that observed the Wyle Labs actuator aging tests. Based on the above assessments, 15 months of operation at 2511 MWth are expected to produce very small bushing wear (~1mm), whereas, 2 days of operation at 2910 MWth should produce no visual wear, therefore, continued operation would present no safety risk.

5.0 References

1. Structural Integrity Associates Calculation No. QC-11Q-301, Revision 0, "ERV Finite Element Analysis and Vibration Evaluation."

2. Structural Integrity Associates Calculation No. QC-11Q-302, Revision 0, "Quad Cities Unit 1 Main Steam Line Vibration Data Reduction."

3. Structural Integrity Associates Report No. SIR-04-023, Revision 3, "Quad Cities ERV Vibration Assessment Wyle Labs Testing," SI File No. QC-16Q-401.

Table 1. QC1 ERV Measured Accelerations at 2488 MWth

Location	Acceleration (g)					
	X peak	Y peak	Z peak	X rms	Y rms	Z rms
ERV 3B Inlet (QCTD Ch 1, 2, 3, 15)	0.361	0.925	0.329	0.086	0.295	0.082
ERV 3B Pilot (QCTD Ch 4, 5, 6, 13, 14)	2.670	2.432	0.756	0.686	0.873	0.220
ERV 3E Inlet (QCTD Ch 7, 8, 9)	0.317	1.268	0.277	0.078	0.395	0.064
ERV 3D Inlet (DTD Ch 4, 5, 6)	0.288	0.989	0.474	0.069	0.269	0.106
ERV 3D Pilot (DTD Ch 7, 8, 9)	na	1.537	0.927	na	0.340	0.223
ERV 3C Inlet (DTD Ch 10, 11, 12)	na	8.338	1.218	na	4.145	0.480
TRV 3A Inlet (DTD Ch 1, 2, 3)	0.841	1.343	0.452	0.234	0.335	0.077

Table 2. QC1 ERV Measured Accelerations at 2910 MWth

Location	Acceleration (g)					
	X peak	Y peak	Z peak	X rms	Y rms	Z rms
ERV 3B Inlet (QCTD Ch 1, 2, 3, 15)	0.318	2.207	0.682	0.081	0.703	0.181
ERV 3B Pilot (QCTD Ch 4, 5, 6, 13, 14)	5.460	5.395	1.876	1.516	2.042	0.561
ERV 3E Inlet (QCD Ch 7, 8, 9)	0.729	3.323	0.534	0.181	1.018	0.126
ERV 3D Inlet (DTD Ch 4, 5, 6)	0.474	1.948	0.886	0.125	0.667	0.180
ERV 3D Pilot (DTD Ch 7, 8, 9)	na	2.284	1.544	na	0.749	0.564
ERV 3C Inlet (DTD Ch 10, 11, 12)	na	8.764	1.120	na	4.168	0.466
TRV 3A Inlet (DTD Ch 1, 2, 3)	1.015	3.241	0.711	0.218	0.363	0.151

Notes:
1. Inlet Valve & 3D Pilot Valve: X is perpendicular to the main steam line, Y is vertical and Z is parallel to the steam flow.
2. 3B Pilot Valve: X is perpendicular to the main steam line, Y is vertical and Z is parallel to the steam flow.
3. na - instruments were bad
4. QCTD – Quad Cities Tape Deck: DTD – Dresden Tape Deck

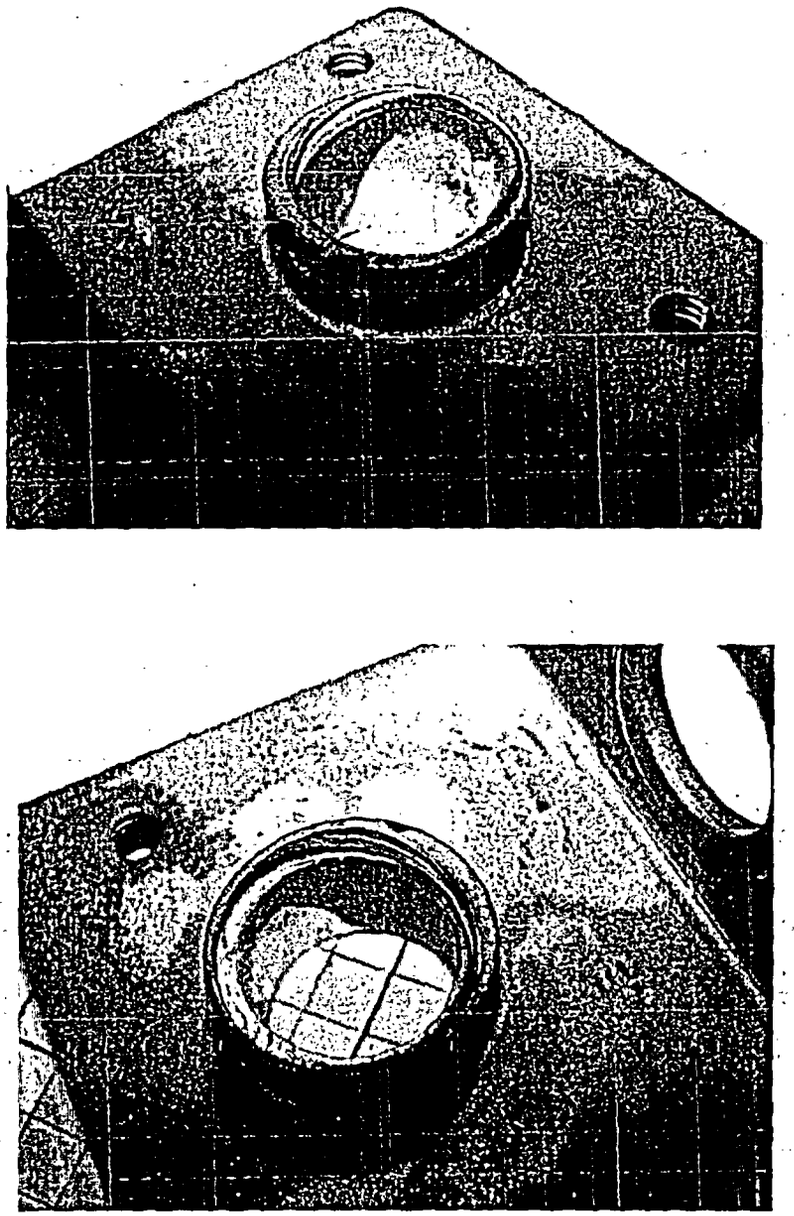

Figure 1: Wyle Labs Test Actuator Aging Test – Bushing Wear

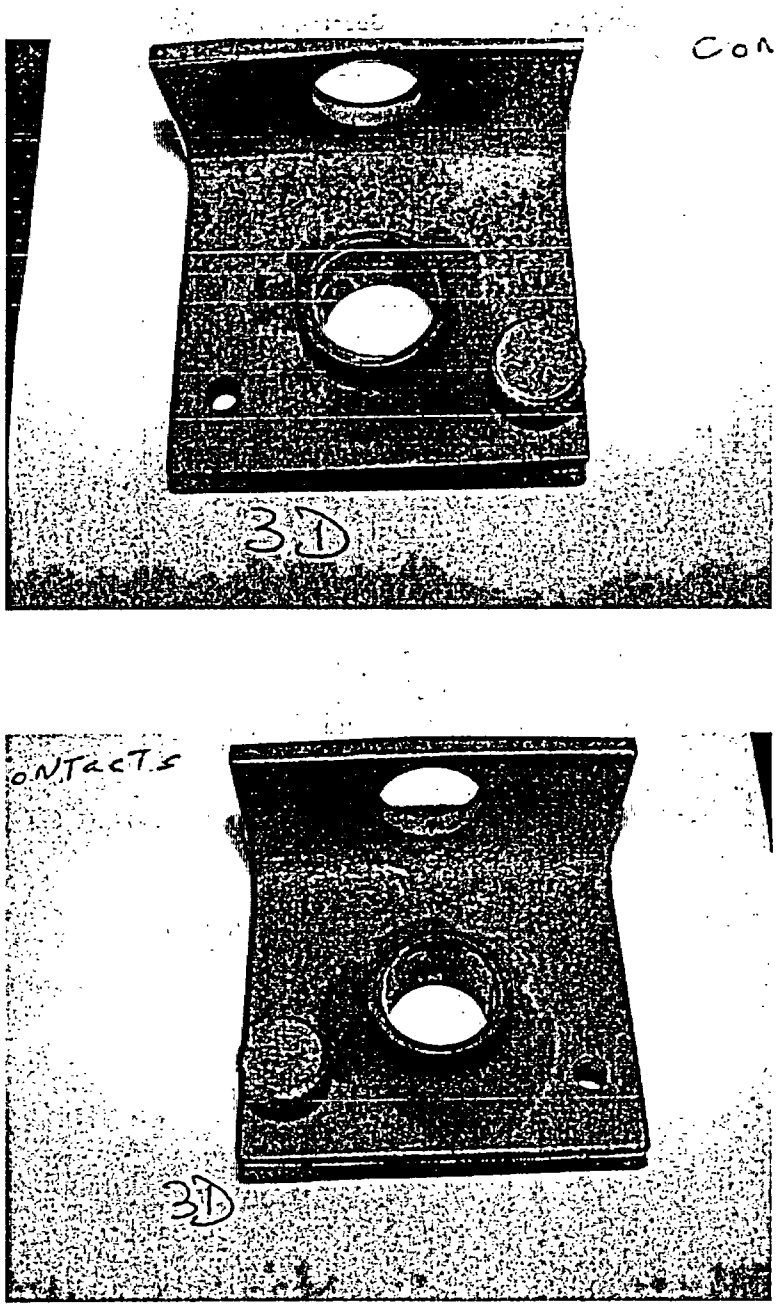

Figure 2: Wyle ERV 3D Actuator after 5 month EPU operation – Bushing Wear

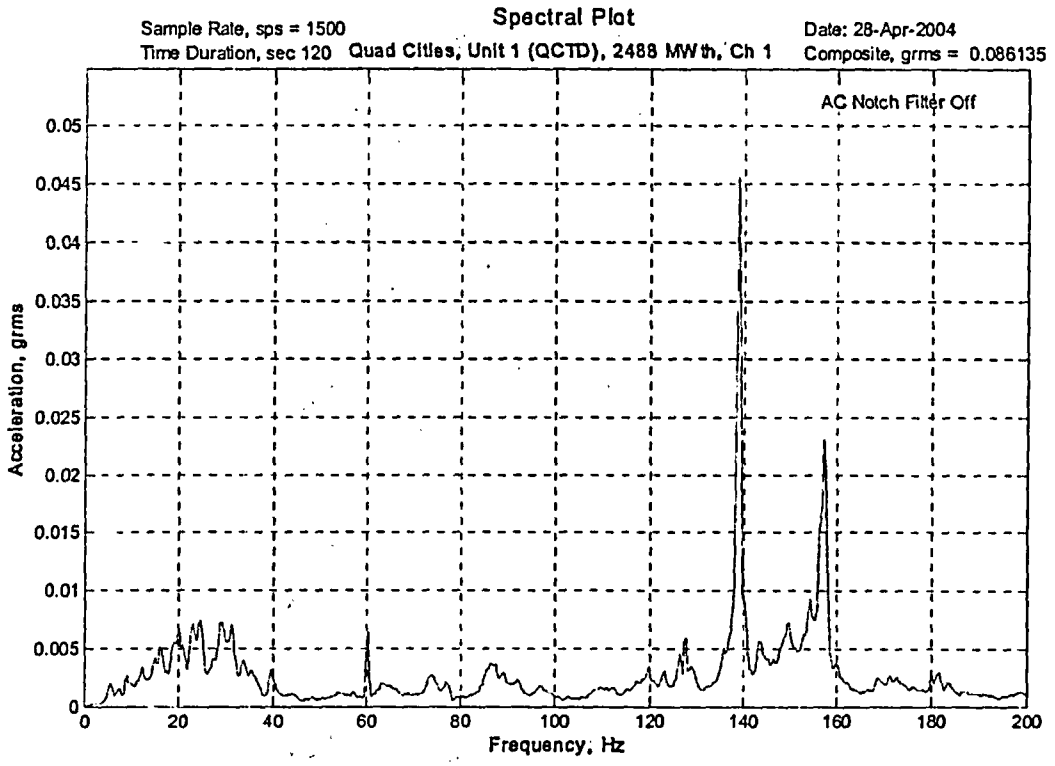

Figure 1: QC1 2488 MWth, ERV 3B Inlet Flange, x-axis, (QCTD Ch 1)

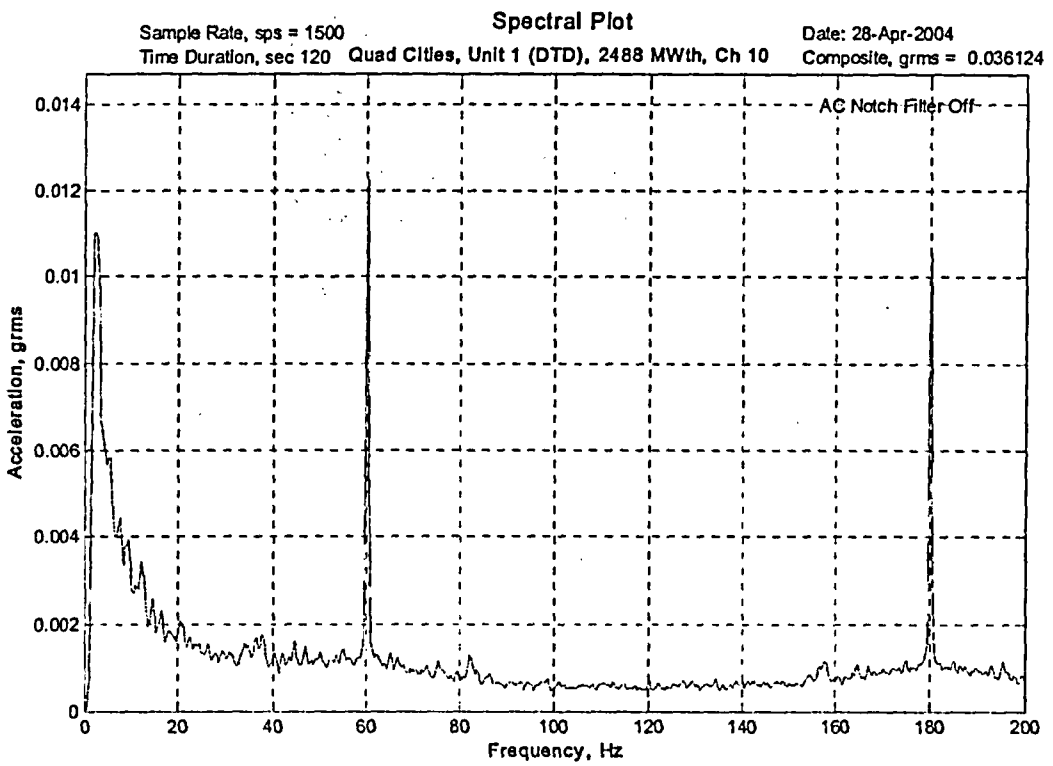

Figure 2: QC1 2488 MWth, ERV 3C Inlet Flange, x-axis, (DTD Ch 10 – bad data)

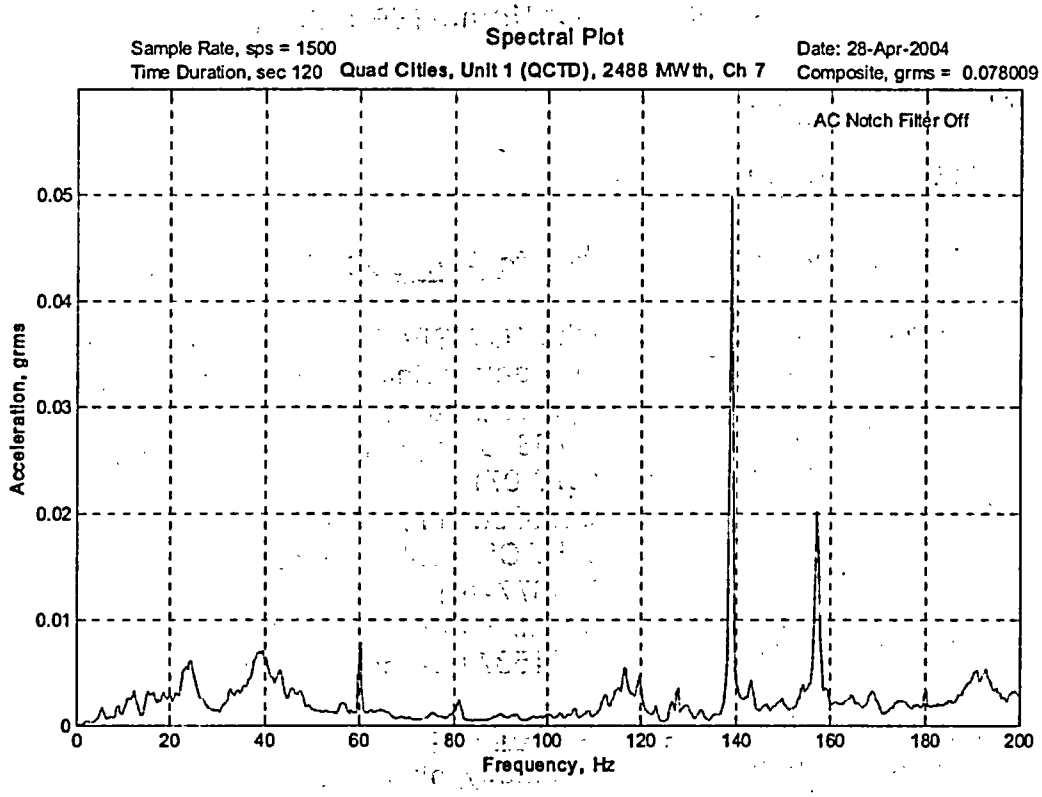

Figure 3: QC1 2488 MWth, ERV 3E Inlet Flange, x-axis, (QCTD Ch 7)

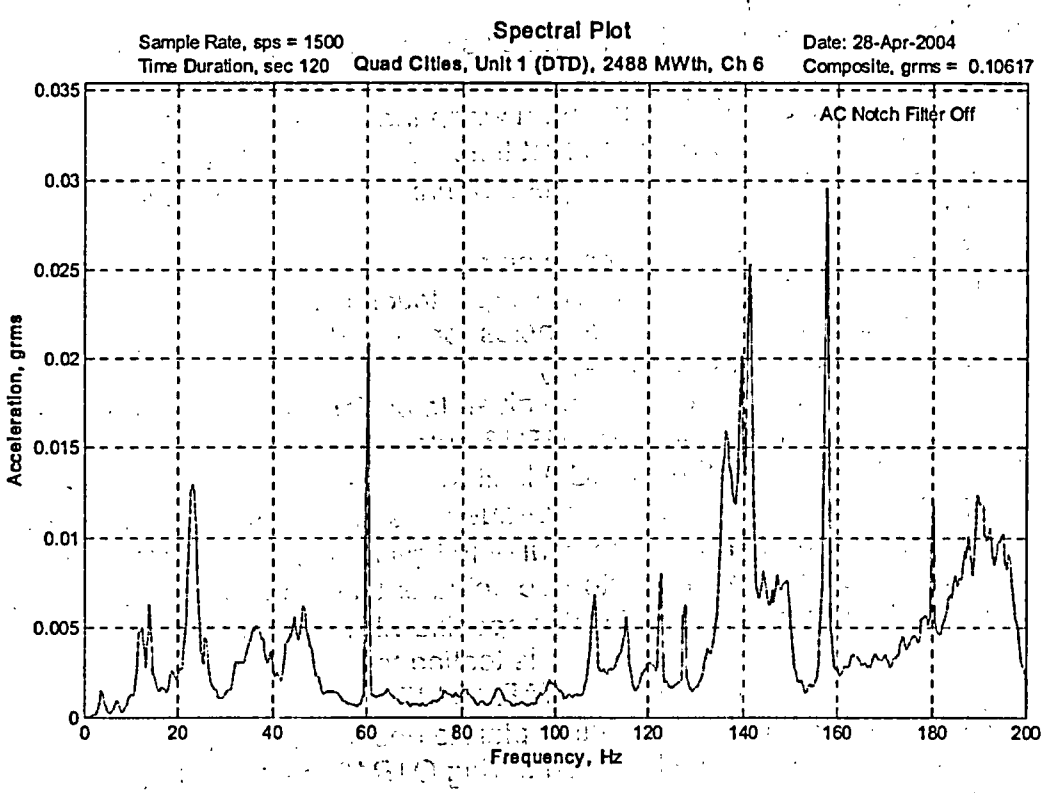

Figure 4: QC1 2488 MWth, ERV 3D Inlet Flange, z-axis, (DTD Ch 6)

EC 346515, Revision 2
Evaluation of Quad Cities Unit 1 Main Steam Line Vibrations at Extended Power Uprate (EPU) Power Levels

For Information Only:

AR 194877 Required Actions:

Prior/During Planned Outage Prior to Above 2511 MWth Operation

1. Install accelerometers on the 1C MSIV during a planned shutdown or forced outage. Evaluation of data obtained will be used to define the need for the inspection required in Action 14 below. (AR 194877-05)
2. Install modification EC 347763 to upgrade the ERV actuator to resist vibration effects. (AR 194877-07)
3. Inspect the components identified during Q1F51 as degraded. These include the ERV actuators, HPCI -4 valve operator and snubber mounting brackets. (AR 194877-10)
4. Disposition the extent of condition of the Unit 2 Target rock valve as found test failure under CR 215874 associated Action Tracking.

After Return to Full EPU Power Operation:

5. Obtain a vibration data set monthly, after return to full power operation after any down power of greater than 10%, and assess for variation/deviation from the analyzed data and any negative impacts. (AR 194877-03)
6. Monitor weekly individual MSL flows to provide any anomalies that indicate MSIV degradation. (AR 194877-11)
7. Obtain a full set of vibration data when the unit is moved to full thermal power of 2957 MWth for the first time. This data will be used to confirm the assumptions made on expected maximum vibration levels. (AR 194877-08)
8. Evaluate the current PM scope and frequency for all components included in this evaluation. The expectation is that some components will need increased PM frequencies for rebuilds, refurbishments and/or replacements. (AR 194877-12)
    a. Consideration should be given to perform a post-EPU baseline inspection during the Q1R18 outage.
9. Perform testing on the NAMCO limit switches using the Quad Cities specific vibration predominate frequencies and amplitudes. This testing to be performed prior to approval of full cycle operation. (AR 194877-13)
10. Perform testing on the Limitorque actuator type and size SMB-2-80 or equivalent using the Quad Cities specific vibration predominate frequencies and amplitudes. This testing to be performed prior to approval of full cycle operation (AR 194877-14)
11. Evaluate the snubber inspection plan to ensure that sufficient MSL snubbers are functionally tested during Q1R18 to ensure that degradation is not occurring. Inspections should include the previously

degraded snubbers, 1-66 and 1-71. Adjust inspection plan as necessary. (AR 194877-10)

12. Evaluate the HPCI -4 valve limit switch to determine if a mod is justified to reduce wear due to vibration after completion of the testing from action 9. (AR 194877-15)
13. Complete comparison of Quad Cities Unit 1 to Dresden Unit 3. This comparison may provide further insights into changes that can be made for Quad Cities to minimize the measured vibration level responses such that accelerated component degradation does not occur. This evaluation includes the following elements:
    a. Expansion of the ongoing MSL circuit analyses and/or scale model testing to include the frequency range up to 180 Hz. This may provide insight into the source of the measured 139 and 157 Hz predominate frequencies.
    b. Detailed configuration differences in all MS line branch line connections between Dresden and Quad Cities.
    c. Detailed evaluation of the ERV configuration installed at Dresden.
    d. A comparison of steam flows between Dresden and Quad Cities on MSL and all branch connections during EPU power level operation.

<u>During Q1R18</u>

14. If results from vibration data on the 1C MSIV warrant, inspect the 1C MSIV; during the next refuel outage (Q1R18) to confirm that no degradation has occurred. If degradation is found, evaluate the need for expanding inspection to other MSIVs.
15. Perform any additional inspections as determined by evaluations performed in actions above.

Engineering Change (EC) 346515, Revision 3
Evaluation (EVAL) of Quad Cities Unit 1 Main Steam Line (MSL) Vibrations at Extended Power Uprate (EPU) Power Levels

Reason For Evaluation / Scope:

Purpose

This EC EVAL, Revision 3, provides clarification to the previous revisions and summarizes the technical evaluations, contents of previous revisions and resultant recommendations, to support continuous operation of Quad Cities Unit 1 at EPU power levels. The clarifying information provides the open actions required to be completed to support continuous full EPU power operation. Included in those actions is an action that created a predefined Preventative Maintenance (PM) action, that details the planned walkdowns for each refueling cycle to validate continued acceptable performance.

Revision 2 provided information on the turbine control valve (TCV) pressure switch acceptability (Reference 7) and the acceptability of Electromatic Relief Valves (ERVs) with pilot valve tiebacks for operation up to 2511 megawatts thermal (MWth) with short term operation up to 2957 MWth for data collection (≤ 72 hours above 2511 MWth). Also included is information further validating the acceptability of the High Pressure Coolant Injection (HPCI) System inboard steam supply valve, HPCI-1-2301-4, operator limit switch. The information included as Revision 2 remains valid.

Revision 1 provided an additional technical evaluation for allowing operation of Quad Cities Unit 1 up to a continuous power level of greater than 2511 MWth up to 2957 MWth. The additional information resolved the conditional (time limit) approval for the ERVs provided in Revision 0. This revision included the results of additional investigation on the HPCI -4 valve operator limit switch. Also added was a restriction on ramp up to prolonged full power until disposition of the Unit 2 failed Target Rock Safety/Relief Valve (S/RV).

Revision 0 provided a technical evaluation for allowing operation of Quad Cities Unit 1 up to a continuous power level of greater than 2511 MWth up to 2957 MWth for a maximum of 3500 hours. The restriction was based on the potential loss of function of the ERVs after prolonged operation. Installation of the upgraded actuator will remove any restriction for ERVs. Other components evaluated in Revision 0 remain qualified for continuous operation, as documented in Revision 0. These components include small bore piping, MSIVs, and other components listed in the summary table on pages 14 – 16 of Revision 0.

Approach

Vibration data throughout the range of power operation, component failure history, and analytical modeling were used to assess component and system responses for full EPU power operation. Vibration data were obtained during the ramp up to and at the pre-EPU power level of 2511 MWth (nominal) and then

during the ramp up to and at 912 MWe (2910 MWth) on December 30, 2003. This data was extracted and then utilized in evaluations. The purpose of these evaluations was to provide assurance that full EPU power operation will pose no threat to continued equipment operation throughout the remainder of the current fuel cycle or to make specific actions required prior to allowing full cycle operation. A listing of the components evaluated and the evaluation method used is included as Attachment 3.

The as-found ERV condition was evaluated by analytical methods, performance history, and extensive testing at the Wyle Laboratories facility. These evaluations confirmed the failure mode, provided input to recommended corrective actions, and validated the acceptability of the corrective actions. Detailed results of the testing and changes made to the actuator are provided in EC 343933, "Replace [Power-Operated Relief Valves] PORVs with ERVs"

The HPCI 1-2301-4-valve operator limit switch degradation was further evaluated by performing calculations to determine if any resonance response of the four rotor internals was likely, given the Quad Cities Unit 1 measured vibrations. The calculations are referenced under this EC. Subsequent shaker table testing confirmed the acceptability of this component and is documented under EC EVAL 350691.

Conclusions / Findings:

This EC EVAL, including Revisions 0,1, and 2, provides an engineering evaluation supporting continuous operation at EPU power, provided all required actions under Action Request (AR) 194877 as listed below, are completed in the required timeframe. It has been determined that this operation will not result in unacceptable degradation of any components.

1. Install accelerometers on the 1C Main Steam Isolation Valve (MSIV) during a planned shutdown or forced outage. Evaluation of data obtained will be used to define the need for the inspection required in Action 14 of AR 194877. (AR 194877-05)
2. Install modification EC 347763 to upgrade the ERV actuators to resist vibration effects. (AR 194877-07)
3. Inspect the components identified as degraded during Q1F51 (see Attachment 1). These include the ERV actuators, HPCI -4 valve operator and snubber mounting brackets. (AR 194877-10) Included as a part of this action is the creation of a repetitive walkdown action for each refuel cycle.
4. Disposition the extent of condition (EOC) of the Unit 2 Target Rock S/RV as-found test failure under CR 215874 (see detailed evaluation section below).

Note:
Revision 2 also documents that the main steam line (MSL) equipment, including ERVs with tieback modifications, are acceptable for continuous operation at pre-EPU power levels, with short duration power ascension (≤ 72 total hours above 2511 MWth) for the purpose of data collection. The determination is supported by evaluation of the additional information on the ERVs, HPCI -4 valve, and Turbine Control Valve pressure switch, as discussed above.

Detailed Evaluation:

Acceptability of ERV Component Operation at EPU Levels
Extensive testing performed at Wyle Laboratories provided clarification to the vibration response of the ERV actuators. This testing confirmed that the internals of the ERV actuator were responding to some input frequencies that resulted in premature degradation of those components. Specifically, the plunger support spring, guide rod, and bracket bushing exhibited accelerated wear. These components have been redesigned by utilizing harder materials and special edge treatments to ensure acceptable long-term performance. The modified configuration was subjected to endurance testing to simulate in-plant service and was determined to resolve the degradation concern. The revised configuration will be installed prior to any prolonged full EPU power operation under EC 347763, "Remove Actuator Tieback Supports on Unit 1 ERVs and Upgrade Actuators and Drain Line Flanges." Documentation of the testing and analysis of results can be found under EC 343933, "Replace the Current PORVs with ERVs." (Unit 2 modification for PORV to ERV change – [installed])

Evaluation of the Target Rock S/RV – First Stage Pilot
The as-found testing of the Target Rock S/RV from Unit 2 resulted in a setpoint lift at +6.8% of allowable value in contrast to a code acceptance criteria of +3% and the Technical Specifications criteria of 1%. The failure is documented in CR 215874. Until the cause of this failure is determined and the EOC as it relates to Unit 1 can be dispositioned, ramp up of Unit 1 to power levels above 2511 MWth, except for brief periods of data collection, is restricted. Final corrective actions and their implementation will be tracked under AR 215874 and EC EVAL 350693. EC EVAL 346515 will not be revised to reflect that resolution.

Evaluation of HPCI -4 Valve Operator
Based on walkdown data, the limit switch (four rotor design) internal to the HPCI 1-2301-4 valve operator was found degraded. Additional analyses of the operator assembly were performed to determine susceptible internal components. No individual susceptibility was identified through these analyses. These evaluations (i.e., References 4 and 5) and the as-found condition of the Unit 2 HPCI -4 rotor assembly (discussed in EC 348316, "Evaluation of Quad Cities Unit 2 Main Steam Line Vibrations at Extended Power Uprate (EPU) Power Levels," provide the necessary assurance that full-cycle operation at EPU

power is acceptable for this valve operator assembly. Confirmatory testing has also been completed and is documented under EC EVAL 350691, "Evaluation of Test Results on NAMCO Limit Switches and Limitorque Operator," as detailed under Action 10 below.

Evaluation of TCV Pressure Switch
Based on seismic testing acceleration levels being significantly higher than operational vibrations, the TCV pressure switch is determined to be acceptable for EPU levels (Reference 7).

Evaluation of ERVs with Pilot Valve Tieback Modifications
The Quad Cities Unit 1 ERV measured vibration responses (with tie-back supports) are much lower than the Wyle Laboratories test input vibration (Reference 8). Therefore, Quad Cities Unit 1 ERV actuator wear will be much lower than that observed during the Wyle Laboratories actuator aging tests. Based on this determination, and on vibration degradation being time dependent, ERVs with the pilot valve tiebacks are acceptable for pre-EPU power levels (i.e., ≤ 2511 MWth) with short duration power ascension (≤ 72 total hours above 2511 MWth) for the purpose of data collection.

Summary Conclusion:

This EC EVAL provides an engineering evaluation supporting continuous operation at EPU power, provided all required actions under AR 194877 are completed in the required timeframe. It has been determined that this operation will not result in unacceptable degradation of any components.

References:

1. EC 347763, Modification to remove tie-back and upgrade components on ERVs, Unit 1 [In Progress]
2. EC 343933, Rev. 0, Replacement of PORVS with ERVs, Unit 2
3. EC 348316, Rev. 2, Evaluation of Unit 2 Components.
4. QDC-2300-M-1372, Rev. 0, Parametric Analysis of Limitorque Limit Switch Vibration
5. QDC-2300-M-1373, Rev. 0, Evaluation of MOV 1-2301-4 Limitorque limit switch for Vibration effects
6. Independent review by MPR dated April 22, 2004 (copy attached)
7. QDC-0200-M-1360, Rev. 1, Evaluation of Components for Vibration Effects Due to Power Uprate Quad Cities Unit 1
8. SIR-04-054, Revision 1, May 4, 2004, Quad Cities Unit 1 ERV Vibration Assessment (copy attached)
9. EC EVAL 350691, Rev. 0, "Evaluation of Test Results on NAMCO Limit Switches and Limitorque Operator"

MPR
ASSOCIATES INC.
ENGINEERS

April 22, 2004

Ms. Sharon Eldridge
Lead Project Engineer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

Subject: Review of EC-EVAL #346515, "Evaluation of Quad Cities Main Steam Line Vibrations at EPU Power Levels," Revisions 0 and 1

Dear Ms. Eldridge:

Exelon recently provided MPR with EC-EVAL #346515, "Evaluation of Quad Cities Main Steam Line Vibrations at EPU Power Levels," Revisions 0 and 1, for review. The EC-EVAL summarizes technical analyses that assess the impact of operating Quad Cities Unit 1 (QC1) at power levels up to 2957 MWth. MPR has completed a review of the EC-EVAL and concluded that Exelon has taken effective compensatory actions and completed analyses to demonstrate that components susceptible to accelerated vibration damage will maintain their integrity to the end of the current operating cycle. In addition, Exelon has imposed a restriction on operating QC1 at power levels above 2511 MWth until emergency relief valves are modified with wear resistant components, inspections are completed of critical components (i.e., snubber brackets, HPCI valve limit switches and ERV actuators), and instrumentation is installed to acquire vibration data on the 1C Main Steam Isolation Valve. This restriction is warranted and completing the actions will provide adequate margin to operate QC1 at power levels above 2511 MWth.

Our review of the evaluation for Dresden Unit 3 is ongoing. We will contact you early next week with our comments or conclusions. Please call me or Bill McCurdy if you have any questions regarding this letter.

Sincerely,

Phillip J. Rush

Phillip J. Rush, P.E.

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May 4, 2004
SIR-04-054 Revision 1
KKF-04-025

Ms. Sharon Eldridge
Lead Project Engineer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

Subject: Quad Cities Unit 1 ERV Vibration Assessment

Dear Sharon:

This letter report contains the Quad Cities Unit 1 main steam vibration assessment for the Electromatic Relief Valves (ERVs), with standard brass bushings, due to operation at 2511 MWe (pre-EPU) operation for the next 15 months. The assessment is contained in Attachment 1.

If you have any questions, please call me.

Prepared by

Kevin J. O'Hara
Associate

Reviewed by

Karen K. Fujikawa, P.E.
Associate

Approved by

Karen K. Fujikawa
Associate

kkf
Attachment: 1. Evaluation of Quad Cities Unit 1 ERV Vibration Assessment

cc: QC-16Q-102, -403
P. Hirschberg (SI)
K. J. O'Hara (SI)
G. L. Stevens (SI)

Evaluation of Quad Cities Unit 1 ERV Vibration Assessment

1.0 Introduction

The Quad Cities Unit 1 (QC1) 3B Electromagnetic Relief Valve (ERV) actuator was discovered with significant wear/damage during outage Q1F51. This ERV was exposed to 9 months of EPU (2910 MWth) operation and during EPU operation the ERV 3B pilot-valve vent pipe failed. The ERV 3B failure analysis indicated that this vent pipe failure caused the excessive bushing wear to ERV 3B. Prior to EPU operation (January to September, 2003), all QC1 ERVs operated at pre-EPU (2511 MWth) power levels without excessive actuator bushing wear.

The purpose of this letter report is to evaluate QC1 ERV actuators, with standard brass bushings and a tie-back support, for operation at pre-EPU power levels up to 2511 MWth for 15 months and operation at EPU power levels for ~2 days.

2.0 Analytical and Data Assessments

Four assessments will be performed to assess QC1 ERV actuator bushing survival capability. These assessments consist of:

- A review of the previous wear assessment from Reference [1], section 5
- A review of the dynamic vibration magnitudes at both pre-EPU and EPU power levels from Reference [2]
- A comparison of vibration and wear rates observed during the Wyle Labs ERV and actuator tests from Reference [3]. This comparison will be both with and without tie-back supports, and
- A correlation of all data with proposed QC1 operation.

2.1 QC1 Actuator Wear Assessments

The following evaluations were based on ERV 3D photographs after 5 months of EPU (2910 MWth) operation and visual inspection of QC1 ERVs after 24 months of pre-EPU (2511 MWth) operation. This information consists of:

1) Color photographs of the 3D valve actuator bushing wear
2) Visual inspections of ERVs by QC1 valve and maintenance engineers during standard maintenance programs that were conducted during previous outages.

The observations below determine magnitude of the wear and associated wear patterns in order to assess typical wear that would seen after 24 months of pre-EPU operation.

<u>ERV 3D – Typical EPU Wear</u>

Two color photographs show typical bushing wear after 5 months of EPU operation. This wear can be seen in each photograph (Figure 2). The wear shows ~2 mm elongation of the bushing holes. Although some wear was present, this valve was operable at the end of EPU operating cycle and was successfully actuated in this condition.

<u>ERV 3E and 3B – Pre-EPU Wear</u>

QC1 valve and maintenance engineers perform standard inspections on all ERVs after 24 months of pre-EPU operation. Post-inspection results of ERV 3E and 3B actuator bushings (both valves are located on main steam line (MSL) B) are as follows:

- Typically ERV 3E sustains more actuator wear than ERV 3B
- Wear typically consists of some minor hole elongation (less than 2 mm) and the bushing-to-spring contact surface has several wear marks (some shallow gouges from the retention spring)
- During the last several pre-EPU operating cycles, QC1 ERV 3E and 3B had less than 2 mm of wear. Based on the Wyle Labs testing these valves were operable.

ERV 3C and 3D – Pre-EPU Wear

Inspection of ERV bushings, after 24 months of pre-EPU operation, show wear similar to ERV 3E and 3B.

- Wear patterns are consistent with ERV 3E and 3B, but the magnitude of the wear is substantially less (based on visual inspections)
- During the last several pre-EPU operating cycles, QC1 ERV 3C and 3D had less than 2 mm of wear. Based on the Wyle Labs testing these valves were operable.

2.2 Quad Cities 1 Dynamic Data Assessment

Vibration data was acquired at QC1 from December 22 through December 30, 2003. All data was assessed for vibration on all four main steam line (MSL) ERVs, their associated pilot valves and other MSL components. These components were assessed for six power levels, from 2212 MWth up to 2910 MWth (EPU). It is important to note that the ERVs had tie-back supports when this data was acquired. A data summary of the peak and RMS vibration levels at 2488 and 2910 MWth is documented in Reference [2], and ERV accelerations are tabulated in Tables 1 and 2.

Based on testing at Wyle Labs (section 2.3 below), the bushing wear is the result of a local actuator mode [3]. Only during x-axis vibration did this mode become excited sufficient enough to cause the observed wear. Therefore, all the ERV frequency spectra for the x-axis (parallel to pilot valve turnbuckle) vibration were reviewed for vibration magnitude, frequency content and actuator excitation that could potentially result in excessive actuator wear. The power level of interest is pre-EPU (2511 MWth) power level which is 8.5% higher (based on flow velocity squared) than 2488 MWth (power level of measured data). A scale factor of 1.10 was conservatively applied to the 2488 MWth vibration levels to approximate the vibration levels at 2511 MWth.

Inspection of the spectra plots for ERV 3B, 3D and 3E inlet flanges indicate most of the energy is amassed at the discrete frequency of 139 or 157 Hz. It should be noted that ERV 3C x-axis (ch 10) is bad. The QC1 ERV valve inlet measured random and sine amplitudes are:

ERV 3B Spectral response (QCTD Ch 1)
- x-axis (vertical to MS flow): 0.086 grms with a superimposed sine of 0.064 g
- x-axis (scaled by 1.10): 0.095 grms with a superimposed sine of 0.070 g

ERV 3C Spectral response (DTD Ch 10)
- x-axis (vertical to MS flow): bad data for this channel
- x-axis (scaled by 1.10): bad data for this channel

ERV 3D Spectral response (DTD Ch 6)

- z-axis (parallel to MS flow): 0.106 grms with a superimposed sine of 0.042 g
- z-axis (scaled by 1.10): 0.117 grms with a superimposed sine of 0.047 g

ERV 3E Spectral response (QCTD Ch 7)

- x-axis (vertical to MS flow): 0.078 grms with a superimposed sine of 0.071g
- x-axis (scaled by 1.10): 0.086 grms with a superimposed sine of 0.078g

The maximum composite vibration (0-200 Hz) is on ERV 3D inlet flange (0.117 grms). Additionally, spectral energy in the 30-70 Hz range for ERV 3D is very low (less than 0.0065 grms). The only observed frequency content with any magnitude is at 23 Hz, which is outside the local modes of vibration observed in the Wyle Labs testing. Therefore, the QC1 vibration levels are low at 2511 MWth and would not produce much, wear after 15 months of operation.

2.3 Wyle Labs ERV and Actuator Test Results

Several tests were performed at Wyle Labs. Initial testing was performed with the tie-back (test configuration 1) and without the tie-back (test configuration 2). The random vibration testing for both configurations showed the maximum response to be higher with the tie-back:

- ERV with tie-back: test run 7 - 3X=0.92 grms, 3Y=5.57 grms, and 3Z=0.61 grms
- ERV w/o tie-back: test run 23 - 3X=0.38 grms, 3Y=0.67 grms, and 3Z=0.19 grms
- Plunger w/tie-back: test run 8 - 6X=3.37 grms, 6Y=1.53 grms, and 6Z=1.80 grms
- Plunger w/o tie-back: test run 24 - 6X=0.95 grms, 6Y=0.87 grms, and 6Z=0.58 grms

The ERV solenoid and plunger responses were higher with the tie-back. All ERV test results indicated that the most severe actuator wear came from x-axis vibration. Therefore, all actuator-only aging tests were performed in the x-axis only.

Actuator-only aging tests were complete at Wyle Labs towards the end of the test series (test cases 45-55). Test series 6 (test cases 45, 46 and 49) are relevant, since they represent actuator-only testing with the original brass bushings. These tests consisted of resonance search (6.A.X) and actuator aging tests (6.B.X). The resonance search identified a 62 Hz plunger mode that was active within the actuator, but the plunger also seemed to respond between 30-70 Hz.

The Wyle aging test consisted of two parts:

1) Test Case 46: low frequency 30-70 Hz at 0.8 grms for 49 minutes and
2) Test Case 49: high frequency 20-200 Hz at 0.8 grms for 3 hours

Post-test results showed that the Wyle Test Actuator bushing wear was very similar to ERV 3D bushings that experienced five months of EPU operation at QC1. Both sets of bushings had the same amount of wear (~2 mm, measured) and similar wear patterns.

3.0 QC1 Data Correlation and ERV Assessment

QC1 is planning to operate at pre-EPU (2511 MWth) for the next 15 months with ~2 days of operation at EPU (2910 MWth). Three assessments have been performed to understand the impact of operation at each power level and the duration.

EPU Operational Assessment

- QC1 has operational history at EPU power level for 9 months
- Wyle Labs actuator aging tests (with brass bushings) showed ~2 mm of wear at 0.8 grms
- Detailed inspections of ERV 3D, after 5 months of EPU operation with no tie back, revealed ~2 mm of wear, and the valve passed outage functional tests.
- It is clear that the Wyle Labs testing was equivalent to 5 months of EPU operation, based on the magnitude of wear and similar wear patterns. The valve/actuator passed all post-test functional checks.

Based on this operational history and the observed wear, operation for 2 days at EPU power should not show any significant wear, due to the limited time spent at this power level.

Pre-EPU Operational Assessment

- QC1 has several periods of operational history at pre-EPU power level, where all ERVs were exposed to this power level for 24 months
- Outage inspections of ERVs that were exposed to 24 months of pre-EPU operation, revealed less than 2 mm of wear, and all valves were operable
- Results from the Wyle Labs testing showed that ERV solenoid and plunger responses were higher with the tie-back than without the tie-back.
- Comparison of the QC1 per-EPU vibration data (ERVs had tie-backs) with Wyle Labs input spectra showed that the QC1 vibration data was much lower.

Based on this operational history, the lower vibration environments and the observed wear, QC1 operation for 15 months at pre-EPU power should exhibit less than 2 mm of wear (estimated to be ~1 mm). This includes the effect of the ERV with pilot-valve actuator tie-back (which is inherent in the QC1 ERV vibration data).

4.0 Conclusions

The QC1 ERV responses (with tie-back supports) are much lower than the Wyle Labs test input vibration, therefore, QC1 ERV actuator wear will be much lower than that observed the Wyle Labs actuator aging tests. Based on the above assessments, 15 months of operation at 2511 MWth are expected to produce very small bushing wear (~1mm), whereas, 2 days of operation at 2910 MWth should produce no visual wear, therefore, continued operation would present no safety risk.

5.0 References

1. Structural Integrity Associates Calculation No. QC-11Q-301, Revision 0, "ERV Finite Element Analysis and Vibration Evaluation."

2. Structural Integrity Associates Calculation No. QC-11Q-302, Revision 0, "Quad Cities Unit 1 Main Steam Line Vibration Data Reduction."

3. Structural Integrity Associates Report No. SIR-04-023, Revision 3, "Quad Cities ERV Vibration Assessment Wyle Labs Testing," SI File No. QC-16Q-401.

Table 1. QC1 ERV Measured Accelerations at 2488 MWth

Location	Acceleration (g)					
	X peak	Y peak	Z peak	X rms	Y rms	Z rms
ERV 3B Inlet (QCTD Ch 1, 2, 3, 15)	0.361	0.925	0.329	0.086	0.295	0.082
ERV 3B Pilot (QCTD Ch 4, 5, 6, 13, 14)	2.670	2.432	0.756	0.686	0.873	0.220
ERV 3E Inlet (QCTD Ch 7, 8, 9)	0.317	1.268	0.277	0.078	0.395	0.064
ERV 3D Inlet (DTD Ch 4, 5, 6)	0.288	0.989	0.474	0.069	0.269	0.106
ERV 3D Pilot (DTD Ch 7, 8, 9)	na	1.537	0.927	na	0.340	0.223
ERV 3C Inlet (DTD Ch 10, 11, 12)	na	8.338	1.218	na	4.145	0.480
TRV 3A Inlet (DTD Ch 1, 2, 3)	0.841	1.343	0.452	0.234	0.335	0.077

Table 2. QC1 ERV Measured Accelerations at 2910 MWth

Location	Acceleration (g)					
	X peak	Y peak	Z peak	X rms	Y rms	Z rms
ERV 3B Inlet (QCTD Ch 1, 2, 3, 15)	0.318	2.207	0.682	0.081	0.703	0.181
ERV 3B Pilot (QCTD Ch 4, 5, 6, 13, 14)	5.460	5.395	1.876	1.516	2.042	0.561
ERV 3E Inlet (QCD Ch 7, 8, 9)	0.729	3.323	0.534	0.181	1.018	0.126
ERV 3D Inlet (DTD Ch 4, 5, 6)	0.474	1.948	0.886	0.125	0.667	0.180
ERV 3D Pilot (DTD Ch 7, 8, 9)	na	2.284	1.544	na	0.749	0.564
ERV 3C Inlet (DTD Ch 10, 11, 12)	na	8.764	1.120	na	4.168	0.466
TRV 3A Inlet (DTD Ch 1, 2, 3)	1.015	3.241	0.711	0.218	0.363	0.151

Notes:
1. Inlet Valve & 3D Pilot Valve: X is perpendicular to the main steam line, Y is vertical and Z is parallel to the steam flow.
2. 3B Pilot Valve: X is perpendicular to the main steam line, Y is vertical and Z is parallel to the steam flow.
3. na - instruments were bad
4. QCTD – Quad Cities Tape Deck: DTD – Dresden Tape Deck

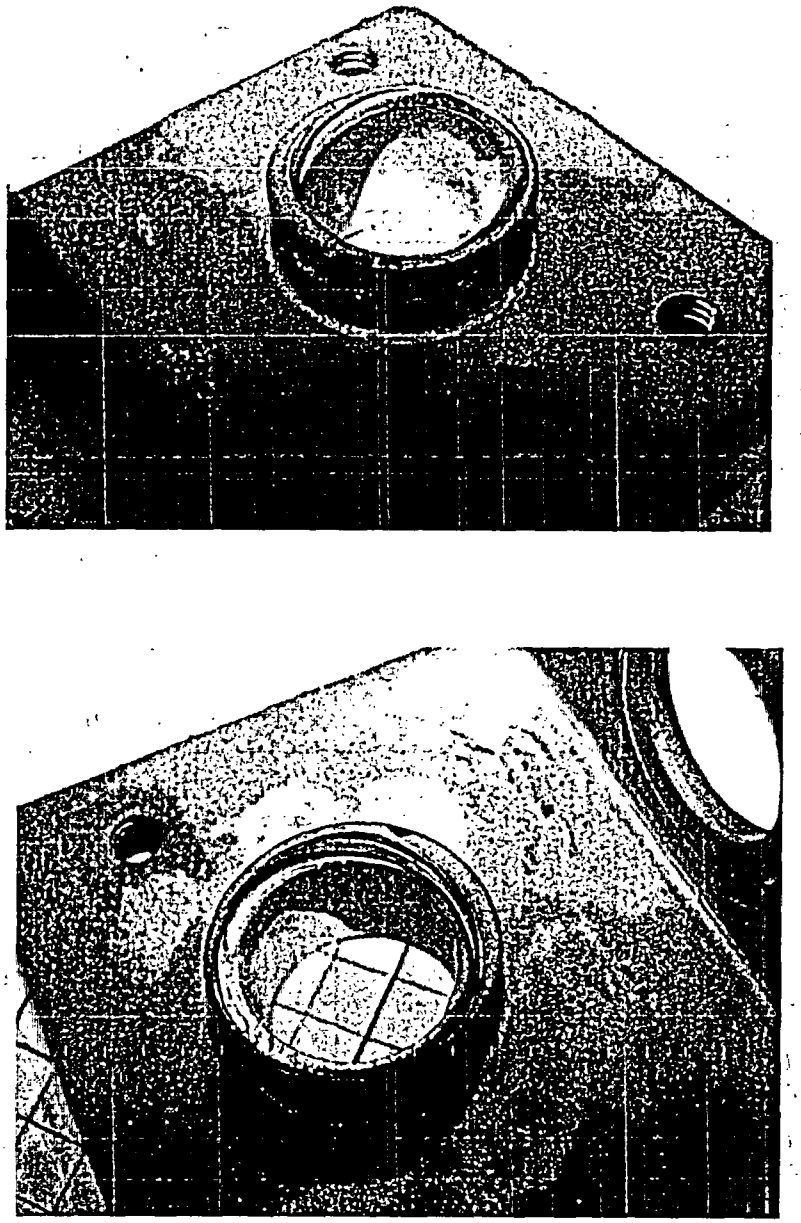
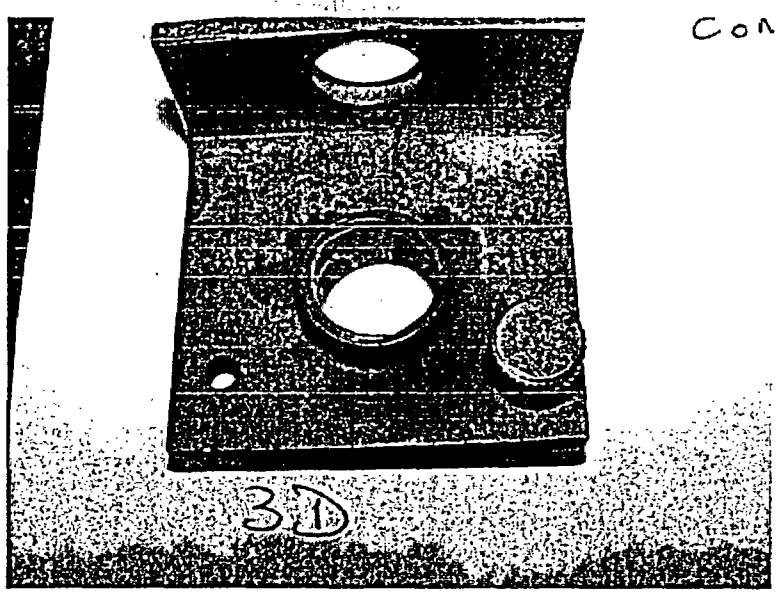


Figure 1: Wyle Labs Test Actuator Aging Test – Bushing Wear



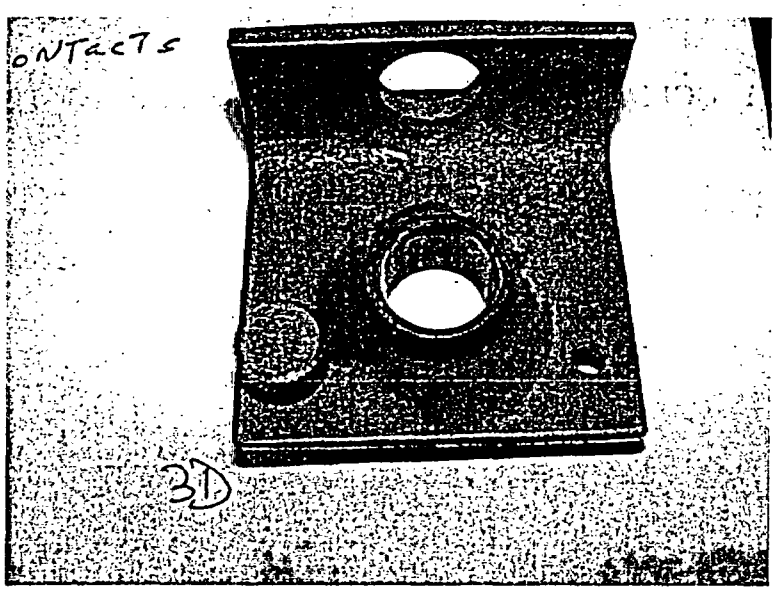


Figure 2: Wyle ERV 3D Actuator after 5 month EPU operation – Bushing Wear

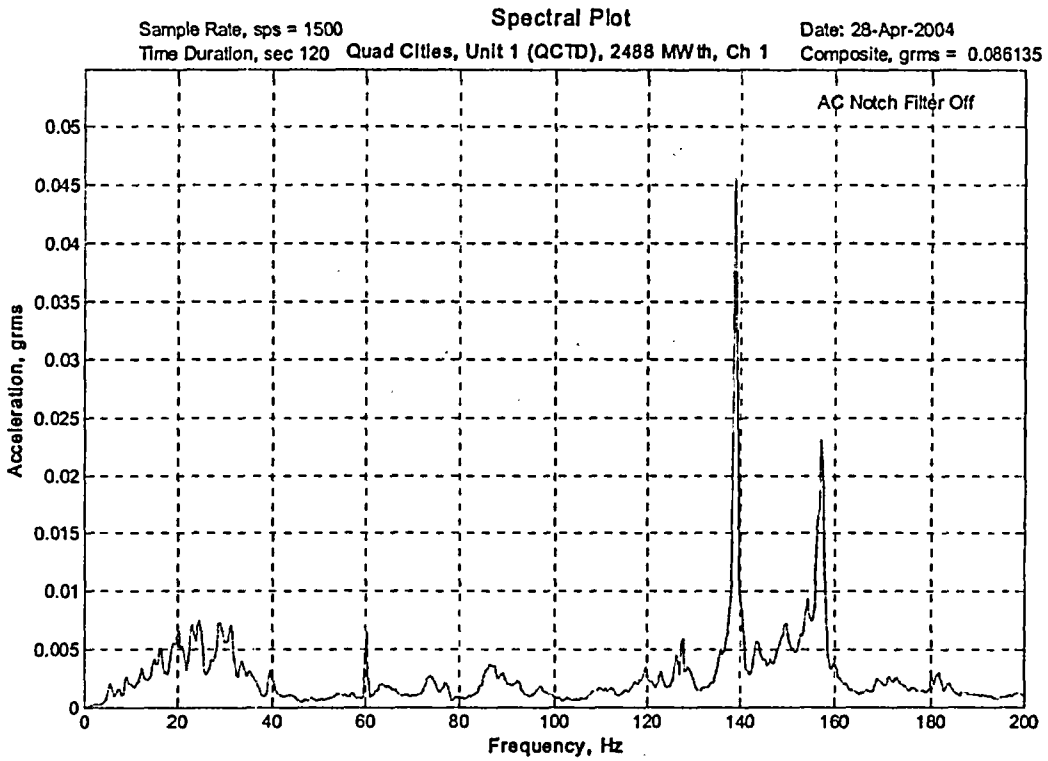


Figure 1: QC1 2488 MWth, ERV 3B Inlet Flange, x-axis, (QCTD Ch 1)

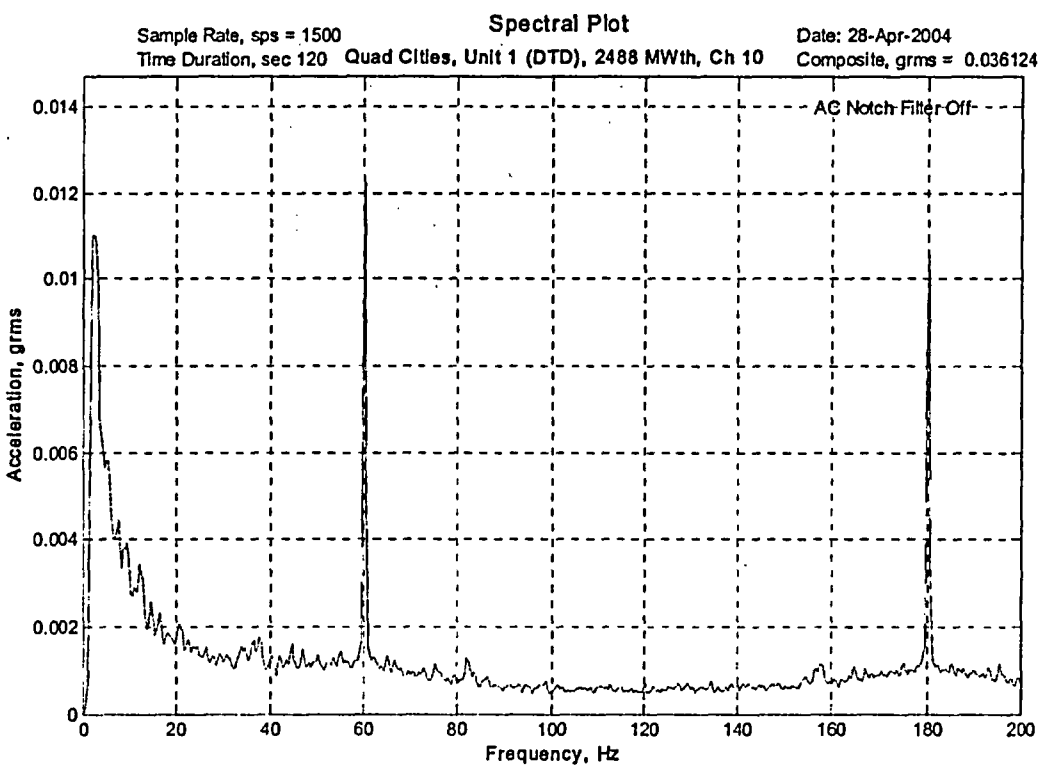


Figure 2: QC1 2488 MWth, ERV 3C Inlet Flange, x-axis, (DTD Ch 10 – bad data)

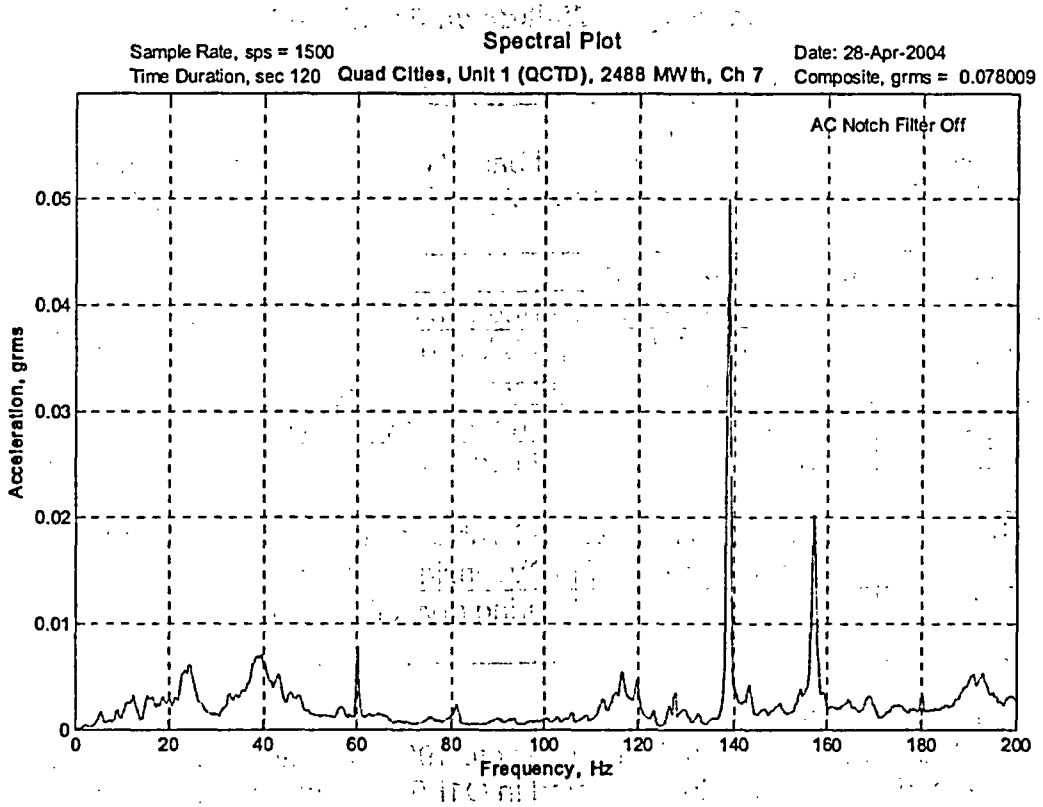


Figure 3: QC1 2488 MWth, ERV 3E Inlet Flange, x-axis, (QCTD Ch 7)

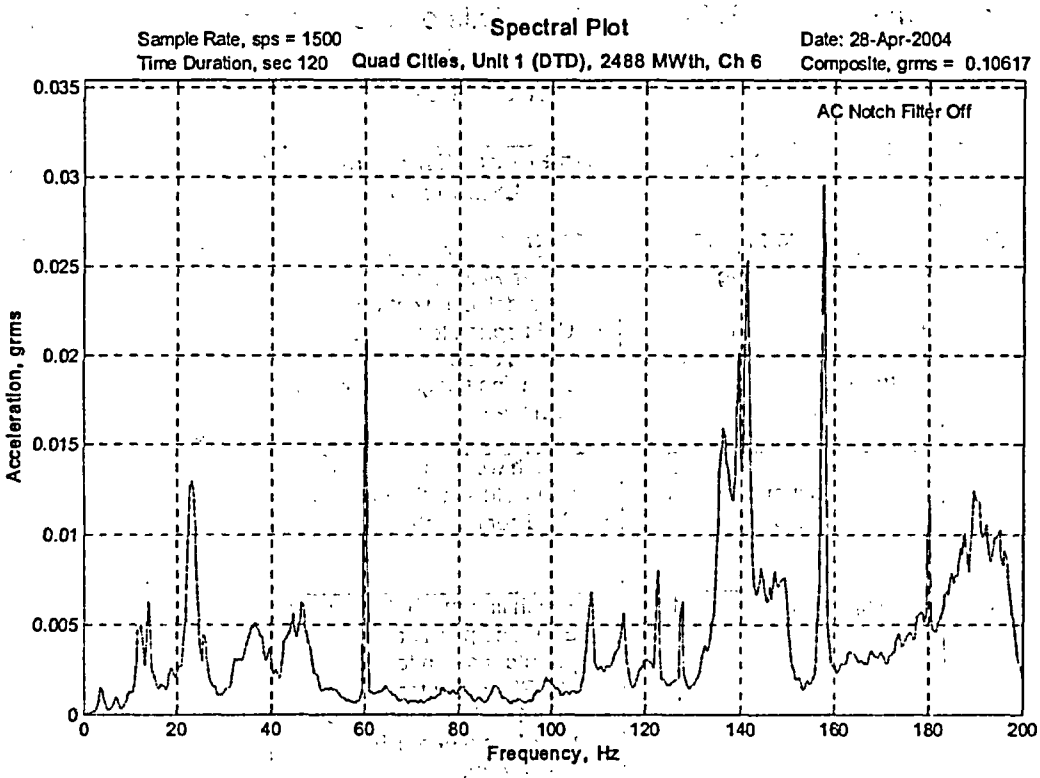


Figure 4: QC1 2488 MWth, ERV 3D Inlet Flange, z-axis, (DTD Ch 6)

Engineering Change (EC) 346515, Revision 3
Attachment 1
Summary of Walkdown Results – Unit 1

Component Damage Summary (Quad Cities Unit 1)

The below information is summarized based on information contained in Condition Reports written during the subject forced outage (Q1F51).

CR No.	Brief Description
186979	Documents as-found condition of 3B ERV actuator and drain pipe
188050	Walkdown of drywell, MSIV room and top of torus for structural and support issues, summary of three issues found
188052	Walkdown results from heater bay and Unit 1 turbine deck
188128	Equipment issues identified during walkdown of drywell first level and MSIV room
188185	Issues with HPCI 2301-4 valve motor operated valve (MOV) found during walkdown of MOVs inside containment
188202	Discrepancies noted during drywell walkdown of second, third and fourth levels.

ERV/Solenoid Historical Failure Summary:
ERV actuator refurbishment history back to 1990 indicates that ERVs have experienced wear similar to that noted in Q1F51, although not as severe, prior to EPU operation. ERV issues identified in Q1F51 are included in the table below, with repairs made as stated. A formal root cause is documented under Action Tracking Item (ATI) 186979-13. More details on the historical problems with ERV actuator are provided in the root cause report referenced here. This information was used in part as input into the current evaluations.

EPN	As-Found Actuator Conditions	As-Found Visual and PT Exam results	As-Left Actuator Conditions	As-Left Visual and PT Exam results
1-0203-3B	▪ Left actuator spring protrudes through bushing ▪ Cover welds broken ▪ Guide Rods grooved from spring wear ▪ Limit switch arms broken and/or missing	▪ Pilot drain line broken ▪ Other welds had no recordable indications ▪ Cold spring load noted during attachment of pilot drain line flanges after valve replaced	Actuator refurbished	▪ Drain line replaced to first flange ▪ Cold spring removed
1-0203-3C	▪ General condition sat, springs just beginning to wear into bushings	▪ 1 inch weld on pilot drain line weld slightly under-filled ▪ Cold spring noted	Actuator refurbished	▪ No action taken - under-filled weld acceptable ▪ Cold spring removed
1-0203-3D	▪ Trip plate pivot point sloppy	▪ 1 inch line at ERV (between ERV and pilot) had 1/8 inch puddle mark in toe of weld at ERV (acceptable) ▪ Slight undercut indication at toe of weld near ERV all the way around	Actuator refurbished	▪ Valve replaced ▪ Results on replacement valve were sat (visual and PT) ▪ No cold spring noted

Engineering Change (EC) 346515, Revision 3
Attachment 1
Summary of Walkdown Results – Unit 1

EPN	As-Found Actuator Conditions	As-Found Visual and PT Exam results	As-Left Actuator Conditions	As-Left Visual and PT Exam results
1-0203-3E	▪ Cracking of spring plate noted at shorting bar ▪ Grooves worn into guide rods ▪ Springs just beginning to wear into bushings	▪ 1-inch pilot drain line has small pitting filled with paint. Pits indicated to be less than 1/8 inch. ▪ No cold spring noted	Actuator refurbished	▪ Flapped the area of the drain line near the pilot valve and re-inspected

ERV Pilot Drain Line Historical Failure Summary:
Drain line breakage has occurred prior to EPU conditions, attributable to both cyclic fatigue, as well as personnel interaction issues.

Snubber Issues discovered during walkdowns in Q1F51:
- TS 1-66 (for 3E ERV discharge piping) - clamp found loose and rotated. A nearby clamp on same pipe was tight. Snubber was tested satisfactorily. The loose clamp was most likely the result of improper installation.
- TS 1-71 (support for C MSL) - load pin was nearly pulled out. This was likely the result of snubber proximity to workers and worker catching protective clothing on the pin. The pin will be replaced by a stud in the upcoming refuel outage, preventing recurrence.

Steam Dryer Issues
Steam dryer issues have impacted station performance. Repairs to the dryer have been completed. A formal root cause is underway under CR 188129 and was completed in February 2004.

Other vibration related issues identified on Unit 1:
- Vibrations in HPCI piping resulted in local leak rate test (LLRT) tap removal
- Small bore piping failures in MS, Feedwater, and Condensate Demineralizer systems (but issue primarily on Unit 2).
- Miscellaneous vibration-related clamp looseness/damage issues identified in Q1F51 (moisture separator supports, MSL low point drain line tiebacks) - these items were corrected.
- Damage noted to 1-2301-4 limit switch rotors in Q1F51.

Conclusions
- Damage, particularly that on the ERVs, generally can be characterized as fretting, wear, and piece part fatigue. This damage can be considered relatively long-term (months), and would not lead to rapid, catastrophic failure.
- With the exception of 3B ERV (failed drain line), Quad Cities Unit1 ERVs were operational.

Engineering Change (EC) 346515, Revision 3
Attachment 2
Summary of Recommended Actions – Unit 1

AR 194877 Required Actions:

Prior /During Planned Outage Prior to Above 2511 MWth Operation

1. Install accelerometers on the 1C MSIV during a planned shutdown or forced outage. Evaluation of data obtained will be used to define the need for the inspection required in Action 14 of AR 194877. (AR 194877-05)
2. Install modification EC 347763 to upgrade the ERV actuator to resist vibration effects. (AR 194877-07)
3. Inspect the components identified as degraded during Q1F51 (see Attachment 1). These include the ERV actuators, HPCI -4 valve operator and snubber mounting brackets. (AR 194877-10)
4. Disposition the EOC of the Unit 2 Target Rock S/RV as-found test failure under CR 215874.
5. A predefine, PMID 172599, has been created with the scope as defined under ATI 197877-16 for a walkdown to be conducted each refuel outage. Changes to the frequency of this PM will be controlled under the PM program and may be adjusted depending on as-found conditions.

After Return to Full EPU Power Operation:

6. Obtain a vibration data set monthly, after return to full power operation after any down power of greater than 10%, and assess for variation/deviation from the analyzed data and any negative impacts. (AR 194877-03)
7. Monitor weekly individual MSL flows to provide any anomalies that indicate MSIV degradation. (AR 194877-11)
8. Obtain a full set of vibration data when the unit is moved to full thermal power of 2957 MWth for the first time. This data will be used to confirm the assumptions made on expected maximum vibration levels. (AR 194877-08)
9. Evaluate the current PM scope and frequency for all components included in this evaluation. The expectation is that some components will need increased PM frequencies for rebuilds, refurbishments and/or replacements. (AR 194877-12)
    a. Consideration should be given to perform a post-EPU baseline inspection during the Quad Cities Unit 1 refueling outage scheduled to begin in March 2005 (i.e., Q1R18).
10. Perform testing on the NAMCO limit switches using the Quad Cities specific vibration predominate frequencies and amplitudes. This testing to be performed prior to approval of full cycle operation. (Complete) (AR 194877-13)
11. Perform testing on the Limitorque actuator type and size SMB-2-80 or equivalent using the Quad Cities specific vibration predominate

frequencies and amplitudes. This testing is to be performed prior to approval of full-cycle operation (Complete) (AR 194877-14)

12. Evaluate the snubber inspection plan to ensure that sufficient MSL snubbers are functionally tested during Q1R18 to ensure that degradation is not occurring. Inspections should include the previously degraded snubbers, 1-66 and 1-71. Adjust inspection plan as necessary. (AR 194877-10)
13. Evaluate the HPCI -4 valve limit switch to determine if a modification is justified to reduce wear due to vibration after completion of the testing from action 9. (AR 194877-15)
14. Complete comparison of Quad Cities Unit 1 to Dresden Unit 3. This comparison may provide further insights into changes that can be made for Quad Cities to minimize the measured vibration level responses such that accelerated component degradation does not occur. This evaluation includes the following elements:
    a. Expansion of the ongoing MSL circuit analyses and/or scale model testing to include the frequency range up to 180 Hz. This may provide insight into the source of the measured 139 and 157 Hz predominate frequencies.
    b. Detailed configuration differences in all MS line branch line connections between Dresden and Quad Cities.
    c. Detailed evaluation of the ERV configuration installed at Dresden.
    d. A comparison of steam flows between Dresden and Quad Cities on MSL and all branch connections during EPU power level operation.

<u>During Q1R18</u>

15. If results from vibration data on the 1C MSIV warrant, inspect the 1C MSIV; during the next refuel outage (Q1R18) to confirm that no degradation has occurred. If degradation is found, evaluate the need for expanding inspection to other MSIVs.
16. Perform any additional inspections as determined by evaluations performed in actions above.

EC 346515, Revision 3
Attachment 3
Evaluation Scope

Components	Commodities	EPNs	Make/ Model	Evaluation Method
All Four ERVs	Relief Valve	ERVs 1-0203-3B, C, D, E	Dresser/ 1525-VX-6-SR-A-N (Number in Passport)/1525-VX-3-XFB11-NC120 for C and E / 1525VX-2-XOS108 for B and D (based on S/N)	Analysis
All Four ERVs	Small Bore Lines	ERV EPNs are RV 1-0203-3B, C, D, E	None	Analysis
All Four ERVs	Vacuum Breakers	1-0220-105B, C, D, E	G. P. E. Controls/ LD-233-63	Analysis
All Four ERVs	Actuators (V27)	No unique EPN in Passport, use same EPN as ERV.	Dresser/1525VX-3-XNC217 (Solenoid)	Analysis
All Four ERVs	Acoustic Monitors	FE 1-0261-60B, C, D, E	NDT International, Inc./NDT-838-CNX-S	Analysis
All Four ERVs	TE	TE 1-0261-14B, C, D, E	14B/C/E-Comm Ed BOMs exist and have Catalog Identification Numbers attached to them; 14D - Pall Trinity / 14-T-2H	Analysis
All Four ERVs	Junction Box	1-0203-3B (-J05)	No Model no. in PASSPORT	Walkdown
Target Rock	Flex hose	Target Rock EPN is RV 1-0203-3A	Flex hose not in PASSPORT	Walkdown
Target Rock	PS 1-0262-37A, B and C	1-0262-37A, B, C	EPN not in PASSPORT	Evaluation
Target Rock	PS 1-0203-34A	1-0203-34A	EPN not in PASSPORT	Evaluation
Target Rock	Accumulator Tubing		None	Walkdown

EC 346515, Revision 3
Attachment 3
Evaluation Scope

Components	Commodities	EPNs	Make/ Model	Evaluation Method
Target Rock	SO 1-0203-3A	SO 1-0203-3A	Target Rock/7467F	Evaluation
Target Rock	Acoustic Monitor	FE 1-0261-60A	NDT International, Inc./NDT-838-CNX-S	Analysis
Target Rock	Vacuum breaker	1-0220-105A	GPE Controls / LD-233-63	Evaluation
All Four MSLs	Venturi Small Bore		None	Walkdown
All Four MSLs	Drain Lines		None	Walkdown
All Four MSLs	Penetrations		None	Walkdown
All Four MSLs	Supports		None	Walkdown
All Four MSLs	Snubbers			Walkdown
All Four Inboard MSIV	Limit Switches	1-0203-1A1A (-ZS), 1B1A (-ZS), 1C1A (-ZS), 1D1A (-ZS); ALSO 1-0203-1A2A (-ZS), 1B2A (-ZS), 1C2A (-ZS), 1D2A (-ZS); ALSO 1-0203-1A2B (-ZS), 1B2B (-ZS), 1C2B (-ZS), 1D2B (-ZS); ALSO 1-0203-1A3A (-ZS), 1B3A (-ZS), 1C3A (-ZS), 1D3A (-ZS)	NAMCO Controls / EA180-21302 are in **BLACK** or EA180-22302 are in **RED**	Analysis
All Four Inboard MSIV	Actuators	1-0203-1A (-A12), 1-0203-1B (-A12), 1-0203-1C (-A12), -0203-1D (-A12)	No Model no. in PASSPORT	Evaluation

EC 346515, Revision 3
Attachment 3
Evaluation Scope

Components	Commodities	EPNs	Make/ Model	Evaluation Method
All Four Inboard MSIV	Air Lines/Small Valves		None	Walkdown
All Four Inboard MSIV	Solenoids	1-0203-1A1 (-S45), 1B1 (-S45), 1C1 (-S45), 1D1 (-S45); ALSO 1-0203-1A2 (-S45), 1B2 (-S45), 1C2 (-S45), 1D2 (-S45); ALSO 1-0203-1A2B (-S45), 1B2B (-S45), 1C2B (-S45), 1D2B (-S45); ALSO 1-0203-1A3 (-S45), 1B3 (-S45), 1C3 (-S45), 1D3 (-S45)	Automatic Valve Corp. (A613) / 6910-020 or 6910-010. EPNs in RED don't exist in PASSPORT. EPNs in BLUE are 6910-010.	Evaluation
All Four Inboard MSIV	IA Leaks Snoop		None	Walkdown
All Four Inboard MSIV	Flex Line		None	Walkdown
All Four Inboard MSIV	TE	TE 1-0203-1A, B, C, D	Catalog Identification No. 353038, Type T 0.188 inch diameter by 2.25 inches long	Analysis
All Four Inboard MSIV	Junction Box		None	Walkdown

EC 346515, Revision 3
Attachment 3
Evaluation Scope

Components	Commodities	EPNs	Make/ Model	Evaluation Method
All Four Outboard MSIV	Limit Switches	1-0203-2A1A (-ZS), 2B1A (-ZS), 2C1A (-ZS), 2D1A (-ZS); ALSO 1-0203-2A2A (-ZS), 2B2A (-ZS), 2C2A (-ZS), 2D2A (-ZS); ALSO 1-0203-2A2B (-ZS), 2B2B (-ZS), 2C2B (-ZS), 2D2B (-ZS); ALSO 1-0203-2A3A (-ZS), 2B3A (-ZS), 2C3A (-ZS), 2D3A (-ZS)	NAMCO Controls (N007) / EA180-21302 in **BLACK** or EA180-22302 are RED highlighted EPNs	Analysis
All Four Outboard MSIV	Actuators	1-0203-2A (-A12), 1-0203-2B (-A12), 1-0203-2C (-A12), 1-0203-2D (-A12)	No Model no. in PASSPORT. Make is GE. Exelon Catalog Identification No. 0037399	Evaluation
All Four Outboard MSIV	Air Lines, Small Valves		None	Walkdown
All Four Outboard MSIV	LLRT Taps		None	Walkdown

Evaluation Scope

Components	Commodities	EPNs	Make/ Model	Evaluation Method
All Four Outboard MSIV	Solenoids	1-0203-2A1 (-S45), 2B1 (-S45), 2C1 (-S45), 2D1 (-S45); ALSO 1-0203-2A2A (-S45), 2B2A (-S45), 2C2A (-S45), 2D2A (-S45); ALSO 1-0203-2A2B (-S45), 2B2B (-S45), 2C2B (-S45), 2D2B (-S45); ALSO 1-0203-2A3 (-S45), 2B3 (-S45), 2C3 (-S45), 2D3 (-S45)	Automatic Valve Corp. (A613) Model 6910-020 in RED or 6910-010 in BLUE highlight. There are only 12 (-S45) items in PASSPORT.	Evaluation
All Four Outboard MSIV	Flex Line		None	Walkdown
All Four Outboard MSIV	TE	TE 1-0203-2A, B, C, D	Catalog Identification No. 353038, Type T 0.188 inch diameter by 2.25 inches long	Analysis
All Four Outboard MSIV	Junction Box			Walkdown
All Eight Safety Valves	Small Bore Lines		None	Walkdown
All Eight Safety Valves	Acoustic Monitors	FE 1-0261-63A, B, C, D, E, F, G, H	NDT International, Inc./NDT-838-CNX-S	Analysis
All Eight Safety Valves	TE	TE 1-0261-13A, B, C, D, E, F, G, H	Grounded TC (Catalog Identification No. 400152) on 8-5/8 inch pipe clamp	Analysis
MO 1-0220-1	Operator	1-0220-1 (-L05); In PASSPORT	Limitorque SMB-000 OAR-33.5; SMB-000-5 per installation history	Analysis
MO 1-0220-1	Limit Switch	1-0220-1 (-ZS)	Not in PASSPORT	Analysis

Evaluation Scope

Components	Commodities	EPNs	Make/ Model	Evaluation Method
MO 1-0220-1	LLRT Taps		None	Analysis
MO 1-0220-2	Operator	1-0220-2 (-L05); In PASSPORT	Limitorque SMB-00	Analysis
MO 1-0220-2	Limit Switch	1-0220-2 (-ZS)	Not in PASSPORT	Analysis
MO 1-0220-2	LLRT Taps		None	Walkdown
HPCI -4 Valve	Operator	1-2301-4 (L05); In PASSPORT	Limitorque SMB-2-80 (from passport) – from System Engineer - Crane 783-U, SMB-1, four-rotor limits installed 10", Crane drawing H-29528,	Evaluation
HPCI -4 Valve	Limit Switch	1-2301-4 (ZS);	Not in PASSPORT	Evaluation
HPCI -4 Valve	Penetration		None	Walkdown
HPCI -4 Valve	LLRT Taps		None	See comments
HPCI -5 Valve	Operator	1-2301-5 (-L05); In PASSPORT	Limitorque SMB-2-80	Evaluation
HPCI -5 Valve	Limit Switch	1-2301-5 (-ZS);	Not in PASSPORT	Evaluation
HPCI -5 Valve	Penetration		None	Walkdown
HPCI -5 Valve	LLRT Taps		None	See comments
Reactor Core Isolation Cooling (RCIC) Inboard Primary Containment Isolation (PCI)	Operator	1-1301-16 (-L05); In PASSPORT	Limitorque SMB-00-OAR94; SMB-00-10 per installation history (from Passport) – from Systems - Crane 783-U with SMB-000 actuator and internal limit switch	Evaluation
RCIC Inboard PCI	Limit Switch	1-1301-16 (-ZS)	See above	Evaluation

Evaluation Scope

Components	Commodities	EPNs	Make/ Model	Evaluation Method
RCIC Inboard PCI	Penetration		None	Walkdown
RCIC Inboard PCI	LLRT Taps		None	Walkdown
RCIC Outboard PCI	Operator	1-1301-17 (-L05); In PASSPORT	Limitorque SMB-00 (from Passport) - from Systems - Crane 783-U with SMB-000 actuator and internal limit switch	Evaluation
RCIC Outboard PCI	Limit Switch	1-1301-17 (-ZS)	See above	Evaluation
RCIC Outboard PCI	Penetration		None	Walkdown
RCIC Outboard PCI	LLRT Taps		None	Walkdown
Electrohydraulic Control (EHC)	TCV Accumulators	1-5672-V1, V2, V3, V4	No Model no. in PASSPORT	See comments
EHC	Lines Near TCVs, Turbine Stop Valves	EPN for TSVs: HO-1-5699-MSV1, 2, 3, 4; EPN for TCVs: HO 1-5699-CV1, 2, 3, 4; EPN for CIVs: HO 1-5699-CIV1, 2, 3, 4, 5, 6	V15-GE Model G6; ZS-NAMCO EA 700-90964; for CIVs – GE 754E602 and GE G6.	Walkdown
EHC	Pressure Switches	261-30A,B,C,D	Barksdale TC9622-3	Analysis
MSL A	RCIC tie-in			Analysis
MSL B	HPCI tie-in			Analysis
MS Small Bore Lines				Previous evaluations / walkdowns

ATTACHMENT 1C

Engineering Change Evaluation 348316, Revision 2
"Evaluation (EVAL) of Quad Cities Unit 2 Main Steam Line Vibrations at Extended Power Uprate (EPU) Power Levels"

Engineering Change (EC) # 348316, Revision 2
Evaluation (EVAL) of Quad Cities Unit 2 Main Steam Line Vibrations at Extended Power Uprate (EPU) Power Levels

Reason For Evaluation / Scope:

Revision 2 of this EC EVAL is performed to:

1. Clarify the approval status and open actions that are required to be completed prior to Quad Cities Unit 2 resuming continuous full EPU power operation.
2. Include the completed action that added a Preventative Maintenance (PM) recurring activity to repeat the walkdowns conducted for Extent of Condition (EOC) each refuel cycle.
3. Provide the walkdown scope as Attachment 6.

Purpose

Because of issues identified on Quad Cities Unit 1 in November 2003, an evaluation of Unit 2 main steam line (MSL) components for operation under Extended Power Uprate (EPU) conditions has been performed. This EC EVAL provides a technical evaluation for allowing operation of Quad Cities Unit 2 at power levels greater than 2511 megawatts thermal (MWth) up to 2957 MWth for the full 24-month fuel cycle and beyond.

The components evaluated are listed in the table in the section titled, "Summary: Long Term Component Responses to EPU Vibration Levels," on page 5. All components were found acceptable for full cycle operation with the exception of the Target Rock Safety/Relief Valve (S/RV). Due to failure of its as-found leak test, further evaluation of this component is required prior to allowing extended operation above 2511 MWth. Short duration power ascension (≤ 72 total hours above 2511 MWth) for the purpose of data collection is acceptable based on vibration degradation being time dependent. The final evaluation and resolution will be documented under Condition Report (CR)/Action Request (AR) 215874 and EC EVAL 350693. Completion of those documents is sufficient to support a complete cycle of full EPU power operation and revision to this EC is not required. Recommendations are also made for future inspections to ensure continuing acceptable component performance.

Background

During the Quad Cities Unit 2 refueling outage in March 2004 (Q2R17), the Power Operated Relief Valves (PORVs) were replaced with Electromatic Relief Valves (ERVs). This modification (i.e., EC 343933) resulted in Unit 2 being similar to Unit 1 in installed equipment. Testing was performed at Wyle Laboratories in support of that modification and to resolve potential degradation issues for the ERVs. The testing was structured to validate that the modification would be sufficiently robust to perform for a minimum of one fuel cycle (i.e., 24 months). The results of that testing are documented under modification EC 343933 and include recommended changes to actuator internal parts and PM instructions, which were subsequently installed/performed during Q2R17.

Approach

Vibration data was collected throughout the range of power operation (see Reference 3), including that obtained during the March 2004 ramp up to maximum electrical power. Component failure history, analytical modeling, and testing were used to assess MSL

and feedwater (FW) component and system responses for full EPU power operation. These evaluations utilize the previous Unit 1 assessments (found in EC 346515), lab testing results, and the input vibration data to provide assurance that operation above 2511 MWth will pose no threat to continued equipment operation throughout the remainder of the current fuel cycle and throughout a complete 24-month fuel cycle.

Results

The data obtained from the power ascension to 2796 MWth, with extrapolation to 2957 MWth, has been assessed against the previously completed evaluations. Acceptability of the Target Rock S/RV performance under the vibration environment was identified as the only concern that potentially jeopardizes operation above 2511 MWth. Details of the evaluation and recommendations for further actions are provided below. Actions are tracked under AR 194877, general actions, and AR 215874, Target Rock S/RV. Completion of the delineated actions constitutes the only requirements for resuming continuous EPU power operation. No further revision to this EC is required.

Required Actions:

Prior to Continuous Power Operation Above 2511 MWth

1. Resolve the cause of the problem with the Target Rock S/RV and the resultant EOC under AR 215874.

Prior to Q2R18

2. A predefine, PMID 172600, has been created with the scope as defined under Action Tracking Item (ATI) 197877-16 for a walkdown to be conducted each refuel outage. Changes to the frequency of this PM will be controlled under the PM program and may be adjusted depending on as-found conditions.
3. Obtain a full set of vibration data when/if the unit is operating at full power (i.e., 2957 MWth) for the first time. This data will be used to confirm the assumptions made on expected maximum vibration levels. (Assignee: A8426CMO; Analysis action assignee: A8064MW-DR)
4. Evaluate the current PM scope and frequency for all components included in this evaluation. The expectation is that some MSL components will need increased PM frequencies for inspections, rebuilds, refurbishments and/or replacements until sufficient EPU operational experience proves that accelerated aging is not occurring. (Assignee: A8430NESSC).
5. Establish component inspections required during the next outage to validate conclusions and to ensure ongoing component acceptability for EPU power operation. (Complete see ATI 197877-16 and action 2 above)

During Q2R18 Outage

6. Inspect the components internal to the ERV actuators. (Assignee: A8430TP)
7. Inspect one MSIV during the next refuel outage (Q2R18) to confirm that no degradation has occurred, especially in the area of disk to stem. If degradation is found, evaluate the need for expanding inspection to other MSIVs. (Assignee: A8451NESPR)

Detailed Evaluation:

Component Damage Summary (Quad Unit 2)

Walkdowns of MSL affected components have been performed to identify any components that exhibited vibration induced degradation. The results are summarized in ATI 194877-16, which documents the details of the walkdowns conducted in Q2R17.

The review of walkdown data identified 16 instances of potential vibration related equipment degradation (documented in 14 Condition Reports (CRs)) where operation above 2511 MWth may be a contributing factor. Most of these issues can be characterized as loosening of mechanical joints. Individual corrective actions were taken to repair each item and follow-up actions are being tracked to ensure acceptable performance. There were no issues identified where additional monitoring during unit operation was deemed necessary or where restricted operation was warranted.

Steam Dryer Issues
Steam dryer issues have impacted station performance. Repairs to the dryer have been completed and additional analyses of and related to the dryer are ongoing. Evaluation of future dryer performance is contained under separate actions and will not be addressed here.

Effects of Increase to Full EPU Power 2957 MWth

All of the component evaluations utilized data obtained between March 29, 2004, and the April 7, 2004, ramp up to 912 MWe (2796 MWth). These values represent the plant response from low power to 2796 MWth. In the future, power levels may reach full licensed power of 2957 MWth. Each component evaluation includes a discussion of any potential effects of increased vibration levels as the unit approaches full thermal power. Actual vibration levels will be measured if/when the unit achieves full thermal power to confirm the assumptions made.

Acceptability of ERV Component Operation at EPU Levels

The four new ERVs have virtually identical assemblies and are identical to the Unit 1 assemblies, which consist of the main ERV valve body, pilot valve, and solenoid actuator. The pilot valve is connected to the ERV by means of a turnbuckle and a pilot valve tube. Each valve has small diameter leak off piping that is routed back to the ERV discharge line.

The finite element analyses and testing performed for the Unit 1 valve configurations determined natural frequencies and their corresponding mode shapes. For Unit 2, the natural frequencies and mode shapes were compared to the frequency content of the measured and tested vibration data. Testing determined that an independent structural mode of the actuator plunger assembly was responding to input vibrations causing premature wear degradation of the bushing, spring, and guide rod assembly. The valve assemblies installed in Unit 2 were upgraded with hardened components of X750

material for bushings and guide rods, and a modified spring with chamfered edges. These components underwent testing prior to their use to ensure that they would perform at measured vibration levels without experiencing degradation. Details of the testing and results can be found in the documentation package supporting modification EC 343933 and Reference 1. Comparison of the vibration values used for the modification evaluations are bounding for the Unit 2 measured values and, therefore, these valves are acceptable for full EPU power operation, per Attachment 2.

Evaluation of High Pressure Coolant Injection (HPCI) -4 Valve Operator

Based on walkdown data, the limit switch (four-rotor design) internal to the operator for the Quad Cities Unit 2 High Pressure Coolant Injection (HPCI) motor-operated steam supply valve, 2-2301-4, was found in like-new condition. This limit switch was replaced during Q2R17 and the previously installed component was retained to facilitate analysis and comparison to Unit 1. The inspection results and the measured vibration data comparison confirm the acceptability of this component for full cycle operation at full EPU power (see QDC-0200-M-1380, Reference 2).

Evaluation of Piping

Piping models were generated and evaluated for Unit 2 and are documented in EC 343933, the modification to replace the PORVs with ERVs. These models include the main steam (MS) piping from the reactor vessel nozzle to the drywell penetration, and include the relief valve discharge piping to the drywell penetration. Leak off piping for the ERVs was also assessed. The calculations were then validated or re-performed using the measured data from Unit 2. All maximum calculated stresses for the piping were below the allowable per the OM-3 criteria. Therefore, operation at EPU power levels will not impact piping structural integrity (see References 4 and 5).

Evaluation of MSIVs

The evaluation of the main steam isolation valves (MSIVs) was performed by comparing measured values to seismic aging qualification test data for the Quad Cities Unit 1 actuators (see Attachment 1). Seismic aging testing consisted of sinusoidal variation of frequency between 25 and 200 Hz at 0.75 g applied to the valve bonnet in accordance with IEEE-344, "IEEE Recommended Practice for Seismic Qualification Of Class 1E Equipment for Nuclear Power Generating Stations," for vibration endurance testing. Measured values for the B MSIV (inboard) and extrapolated values for the A, D, and C MSIVs were well bounded by this aging evaluation and were, therefore, determined to be acceptable for continuous full EPU power operation (Attachment 1).

Additional internal inspection information indicated that after a full cycle of EPU power operation, no deleterious effects on MSIV condition were found. This information is included in Transmittal of Design Information (TODI) QDC-04-16 (i.e., Attachment 3). Wear aging criteria does not imply the components are qualified for the full five-year recommended maintenance life of the components at the observed steady state vibration levels. In addition, degradation of various internal components may not be identified through external inspection. Therefore, preventive actions to perform the following two monitoring / inspection activities are recommended.

1. Prior to the next Quad Cities Unit 2 refueling outage (Q2R18) scheduled for Spring 2006, monitor MSL flows for any change that could be indicative of degradation of an MSIV.
2. During Q2R18, inspect one inboard MSIV for any signs of wear-related degradation in the area of the stem including pneumatic control components, limit switches, electrical leads, and fasteners.

Evaluation of Target Rock S/RVs

The Target Rock S/RV is evaluated in Attachment 1. Results indicate that there is some risk that future acceptable performance will be impacted due to the measured vibration levels exceeding the tested values. In addition, the as-found testing of the valve removed from Unit 2 during Q2R17 failed its acceptance criteria. CR 215874 has been written to document this failure and to ensure resolution. Full EPU power operation is not acceptable until the root cause of the failure and the EOC reviews are understood and any necessary corrective actions are completed. Documentation of Target Rock S/RV acceptability will be completed under AR 215874 and EC EVAL 350693.

Evaluation of FW Regulating Valves (FRVs)

To address potential impacts to the FRV and small bore FW piping, additional hand held vibration data was obtained at selected locations. The small bore piping evaluations are contained in EC EVAL 348717. The FRV was evaluated in Attachment 2. The vibration levels at EPU power ($\geq$ 2811 MWth) were the same as or lower than those taken at

2488 MWth. Therefore, because these valves have performed acceptably at previous power levels, they are acceptable for full EPU power operation.

Summary: Long Term Component Responses to EPU Vibration Levels

An assessment of the component response to EPU power levels has been performed and is summarized in the following table. The scope for the evaluations was determined based on the Unit 1 EC EVAL 346515 and the walkdown results as documented in ATI 194877-16. For a summary of the walkdown scope see Attachment 6.

Component	2910 MWt Assessment Results	Expected long term EPU Performance	Comments
MSIV Actuators	Sinusoidal aging test at 0.75 g from 25 to 200 Hz -	Acceptable at EPU levels based on existing margin. Recommendation includes continuing internal inspections to inspect for stem to disk degradation.	Reference 2
Inboard and Outboard MSIV Limit Switches	Seismically rugged 21.06 g, 29-minute test at 7.5 g – Acceptable margin	Acceptable at EPU levels based on existing margin,	Reference 2
Inboard and Outboard MSIV Temperature Elements	Passive item, very rigid subjected to small g – Acceptable	Acceptable -	Reference 2
MSIV LLRT Taps	Very low frequency based on analysis – no significant excitation measured at low frequency – Acceptable for EPU levels	Acceptable –	Reference 2
Safety Valve Acoustic Monitors	Sine beat test accelerations 6.0 g compared to 4.226 g	Acceptable at EPU levels based on existing margin	Reference 2
Safety Valve Temperature Elements	Pipe clamp style thermocouple, passive element without any extended masses	Acceptable	Reference 2
Pressure Switches	Seismic qualification tested at 20 g subjected to less than 0.2 g – Acceptable based on margin	Acceptable	Reference 2
ERVs Valves, Actuators, Drain Lines, etc.	Analyzed using 2851 MWth vibration data and Finite Element Model (FEM)	All components acceptable for EPU levels based on margins and operating experience.	Attachment 2
Safety Valves	Seismic qualification 11 g horizontal and 9 g vertical compared to ERV measured values of 1.52 g horizontal and 4.17 g vertical – Acceptable based on margin	Acceptable – Based on existing margin and no indication of degradation after 1 year at EPU levels	Reference 2

Component	2910 MWt Assessment Results	Expected long term EPU Performance	Comments
Target Rock Safety/Relief Valve (3A)	Compared to qualification information on new type two-stage valves	There is some potential for pilot leakage and thread wear on the main piston/stem joint. pilot valve leakage would be detectable by increased tailpipe temperatures. Due to failure of the removed valve from Unit 2, qualification for full power is in question. Further evaluation will be performed under CR 215874 to resolve this issue.	Attachment 1 CR 215874
MSL Drain Valves 1-0220-1 and -2, Operators and Limit Switches	Estimated life duration based on the seismic testing information and extrapolated measured vibration levels is 38 years	Acceptable for full power operation. Valves are closed after turbine warm-up and left closed during normal operation; hence, position change is only required during a unit shutdown.	Reference 2

Conclusions / Findings:

Except for the Target Rock S/RV, this EC EVAL provides an engineering evaluation of MSL components supporting operation up to 2957 MWth. It has been determined that full EPU power operation will not result in imminent failure or unacceptable degradation levels of any components with the exception of the Target Rock S/RV. Disposition of the acceptability of that valve for full EPU power operation will be documented in the actions from CR 215874. Due to this valve's failure during its as-found leak test, further evaluation of this component is required prior to allowing extended operation above 2511 MWth. Short duration power ascension (≤ 72 total hours above 2511 MWth) for the purpose of data collection is acceptable based on the time dependency of vibration degradation. The conclusions for acceptability of other components is provided by evaluation of the empirical data, including operational data, measured vibration data, endurance test reports, and ERV laboratory testing.

To confirm the conclusion, an action item to measure the actual vibration at full EPU thermal power will be performed. The acceptance level is less than the extrapolated values for each evaluated component. Additional actions are recommended to ensure that the wear aging of the components remains acceptable and the appropriate PM frequencies are in place.

Attachments:

Attachment 1: "Evaluation of Quad Cities Unit 2 MSIV and SRV Vibration Data," General Electric Report No. DRF-0000-0023-4260, dated April 16, 2004

Attachment 2: Structural Integrity Associates Vibration Assessment, SIR-04-048

Attachment 3: Transmittal of Design Information QDC-04-16 – Inspection Information from Q2R17 MSIV Internal Inspections

Attachment 4: MPR Independent Review Letter

Attachment 5: Stevenson and Associates Independent Review Letter

Attachment 6: Walkdown Scope, Table of Components Walked Down for EOC.

References:

1. SIR-04-023, Rev. 0, ERV Vibration Testing Assessment.
2. QDC-0200-M-1380, Rev. 0, "Evaluation of Components for Vibration Effects"
3. QC-16Q-303, Rev. 0, Quad Cities Unit 2 ERV Vibration Data Reduction
4. SIR-03-136, Rev. 3, Evaluation of Main Steam Line Vibration for Quad Cities Unit 2 PORV Replacement
5. SIR-04-034, Rev. 1, Evaluation of Effects of Main Steam Vibration on Attached Small Bore lines.
6. AR 194877, Follow-On Actions from EC 346515 and 348316.

ATTACHMENT 1

General Electric Report No. DRF-0000-0023-4260, "Evaluation of Quad Cities Unit 2 MSIV and SRV Vibration Data," Dated April 16, 2004

DRF 0000-0023-4260

April 16, 2004

To: Sharon Eldridge
From: PD Knecht

Subject: Evaluation of Quad Cities Unit 2 MSIV and SRV Vibration Data

References:

1. "Steam Dryer Action Items – Technical Issues", MS Project Schedule, 3/31/04
2. "Quad Cities Nuclear Power Station Units 1 and 2 Environmental Qualification Report - MSIV Actuator", NEDC-31886P, Class III, December 1990
3. "Quad Cities Unit 2 ERV Vibration Data Reduction", Structural Integrity Associates Calculation, QC-16Q-303, April 12, 2004
4. "Quad Cities Unit 1 Assessment of High Frequency Acoustic Vibration on MSL Components", GE-NE 0000-0008-1763-02, December 2002.
5. "Target Rock Relief Valve failure to Fully Open", SIL 646, December 20, 2002
6. "Hope Creek Generating Station Unit 1 Environmental Qualification Report, Book No. S20, NEDC 30942, November 1985
7. "MSIV inspection results during Q2R17 (1A, 2B, 2C and 1D MSIV's)", Exelon email Patrick K. Yost to Sharon Eldridge, March 30, 2004. (TODI QDC 04-16)

Background

The Quad Cities Unit 2 current licensed thermal power (CLTP) is 2957 MWt). With the exception of two forced outages due to dryer hood failures of the steam dryer assembly at Unit 2 has operated at full generator electrical output (912 MWe) up until the most recent outage (Q2R17).

Because main steam line vibration was identified as the potential cause of several component as-found degraded conditions, vibration data on various components external to the reactor pressure vessel at Quad Cities Unit 2 were taken following the restart from Q2R17 on March 31, 2004 and evaluated at ten power levels between approximately 20% and 95% CLTP. Following collection of these data, the unit was returned to the original licensed thermal power (OLTP) until the data could be more closely examined and recent questions from the NRC have been resolved.

GENE was asked in Reference 1 to evaluate the inboard MSIV on the "B" Main Steam line and the Target Rock Safety Relief Valve vibration data and provide recommendations regarding long term operation at CLTP. This letter provides the GENE response to this request.

Methodology

Vibration data for MSIVs were compared against seismic aging qualification test data for the Unit 1 MSIV actuators (Reference 2). The seismic aging test consisted of sinusoidal variation of frequency between 25 hz and 200 hz at 0.75g applied to the valve bonnet. No evidence of damage was observed during these qualification tests. Engineering judgment was used to conclude that the Unit 2 actuators will perform in a similar manner to the Unit 1 valves. This judgment assumes that no modifications have been made that would be impacted by high frequency vibrations.

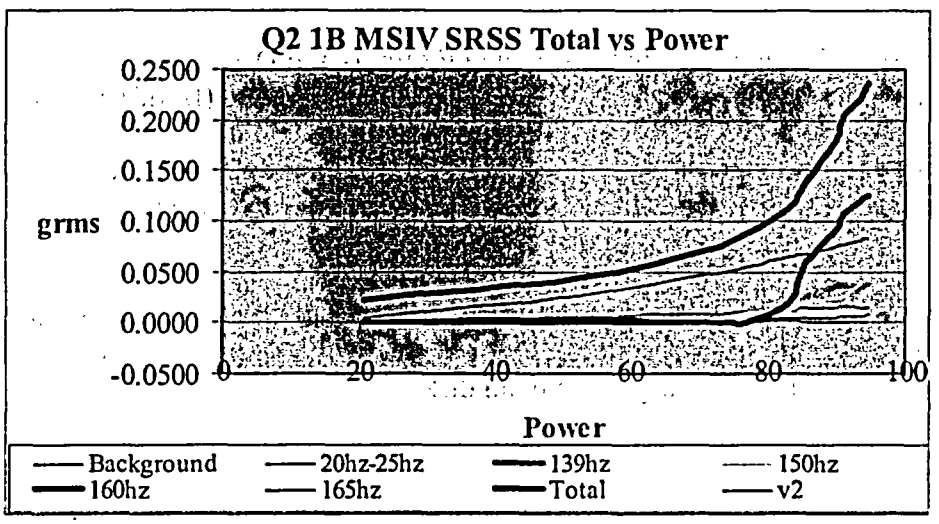
No similar data exists for the 3-stage Target Rock SRVs installed in Unit 2 However, it was assumed that seismic wear aging would have been successful for these valves at the standard IEEE 382-1980 test value of 0.75g. Furthermore, there are other BWRs that have operated at above original rated power that have not reported evidence of wear or stuck open valve events.

Evaluations

1. Main Steam Isolation Valves

<u>B MSIV</u>

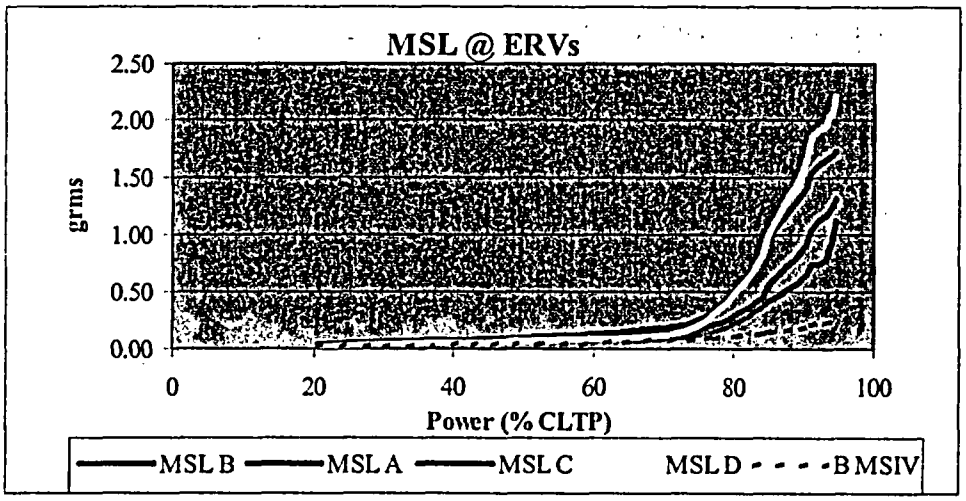
Square Root Sum of Squares (SRSS) values of the three monitored vibration axes corresponding to the peak acceleration values at the "B" MSIV stem are shown in the figure below for the dominant 139 hz and 157 hz frequencies during power ascension. Also shown is the total grms value from reference 3. For reference the velocity squared relationship also is shown.



As can be seen, the amplitude of the vibrations increases with power with a sharp increase in magnitude beginning at about 75% CLTP. Pronounced peaks are evident above the 80% power level for the 139 hz and 160 hz frequencies. In all cases, the amplitudes are relatively low in comparison with the qualification basis of 0.75g and the displacement velocities are below 0.05 in/sec. No damage is expected for these conditions.

A, C and D MSIVs

Because no vibration data was obtained for the A, C or D MSIVs, the total vibration amplitudes at the Electromatic Relief Valve (ERV) and SRV inlets was reviewed as shown in the figure below to determine if the other MSIVs would have a similar response to the B MSIV. For comparison, the dotted line shows the MSIV vibration in the B main



steam line.

As can be seen the amplitude in the B and D ERVs (B and D MSLs) is somewhat larger than the magnitude of the A and C MSLs. The MSIV vibration (B MSL) is substantially lower than the vibration at the ERV inlet on the same steam line. Due to the similarity of amplitudes and geometry of the piping, the other MSIVs likely will have a similar response and be well within the 0.75g wear aging criterion and displacement velocity will be low.

Based on the walkdown results at Unit 2 (Reference 7), some evidence of wear was identified during internal inspections of the valves. Some wear was evidenced on the upper liners and due to rocking of the main disk.

Discussion

It should be noted that the wear aging criteria does not imply that the components are qualified for the full five-year recommended maintenance life of the components with that steady state vibration level. If this amount of vibration occurs, there can be long term degradation of the valve internals. The higher steam flow rate can cause agitation of the internal parts causing accelerated fretting corrosion wear in comparison with pre-EPU conditions (refer to SIL 568). Accelerated wear of other subassemblies on the MSIVs such as pneumatic control components, limit switches, electrical leads and fasteners also may not be identified though external inspection.

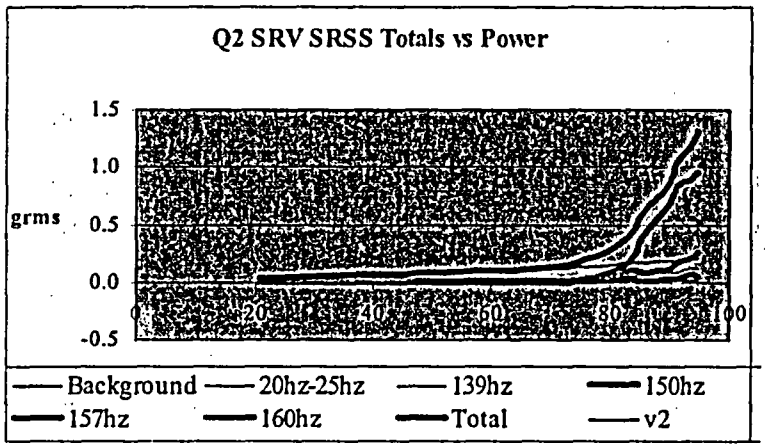
There is no evidence that the wear observed during the Q2R17 internal inspections were the result of EPU operation or long term pre-EPU operation. However, because the rate of wear is expected to increase due to the increase in steam flow with EPU and because this failure mode cannot be detected through external inspection, a followup internal inspection should again be made during Q2R18 to assess the rate of wear following extended EPU operation. A reassessment of the inspection frequency can be made at that time.

In Reference 4, the impact of MSL vibrations was qualitatively discussed. Because the main disk is free to move, high vibration could potentially result in degradation of the lower disk guide liner. The Q2R17 internal inspection revealed some evidence of main disk rocking. If significant, this could slow or possibly prevent valve motion. Although this was considered a low to medium risk in Reference 4, a slow full-stroke closure of at least one MSIV should be considered after 12 to 14 months of full EPU power operation to confirm that accelerated degradation is not occurring. This test will not provide an indication of a degradation trend, but it would confirm operability of the valves given the lack of information on the rate of degradation.

Because no degradation of the MSIVs was shown during the walkdown and to date operability has been demonstrated during the most recent outage, it is concluded that operation at CLTP is acceptable. However, it is recommended that individual MSL velocities be monitored during the operating cycle to detect any variations between steam lines that could be due to MSIV degradation. A slow full-stoke closure of the MSIVs should be considered at least once prior to Q2R18, after 12 to 14 months of full EPU power operation, to confirm operability before the followup internal wear inspection can be conducted.

2. Target Rock Safety Relief Valves

Square Root Sum of Squares (SRSS) values of the three monitored vibration axes corresponding to the peak acceleration values at the "3A" Target Rock valve are shown in the figure below for the power ascension. Also shown is the total grms value from reference 3. For reference the velocity squared relationship also is shown.



As can be observed, the vibration levels are dominated by the 157 hz peak and increase significantly beyond OLTP power. At 96% CLTP the vibration velocity is about 0.37 in/sec. An increased amplitude in the 160 hz frequency also is evident although not at a significant amplitude at the 96% CLTP operating condition.

Discussion

The Target Rock valve at Quad Cities is a 3-stage design that has been replaced with a 2-stage design in later BWRs. No qualification data exist for these valves because they were qualified through the Seismic Qualification Users Group (SQUG) methodologies that did not address wear aging. Qualification aging of 2-stage valve has been successful with wear aging at the same 0.75 g acceleration in a similar manner to that which has been applied to the MSIV actuators. In comparison with the 2-stage Target Tock, the actuators on the 3-stage valve are lighter and it should display higher natural frequencies. Reference 6 identified natural frequencies of 128, 152 and 160 hz in a 2-stage Target Rock valve.

The emergence of a 160 hz presence at the Target Tock inlet and the dominant 157 hz frequency is of some concern. As CLTP is approached during the summer months, there is a risk of vibration induced pilot valve leakage and eventual inadvertent opening at CLTP. Extended operation with leaking pilot valves has been shown to cause steam cutting and eventual inadvertent opening of the Target Rock 3-stage SRV. Because of these concerns, the downstream thermocouples on the SRVs should be closely monitored

for evidence of leakage. The location of the thermocouples has been confirmed to comply with SIL 196 supplement 11.

Because there is no known test data, it is recommended that Exelon conduct testing or develop a finite element model of the three-stage Target Rock valve to further assess the natural frequencies of the valve and its actuator. This analysis would enable Exelon to characterize the vulnerability of the valve to high frequency vibrations and of an inadvertent valve opening that would have similar consequence to the PORV opening which occurred at Unit 2 in April 2003.

Based on the existing Quad Cities experience at CLTP without observed degradation, it is concluded that continuous operation is acceptable subject to monitoring for evidence of pilot valve leakage.

ATTACHMENT 2

Structural Integrity Associates Vibration Assessment, SIR-04-048

Structural Integrity Associates, Inc.

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May 4, 2004
SIR-04-048 Revision 0
KKF-04-020

Ms. Sharon Eldridge
Lead Project Engineer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

Subject: Quad Cities Unit 2 Main Steam Vibration Assessment

Dear Sharon:

This letter report contains the Quad Cities Unit 2 main steam vibration assessment for the Electromatic Relief Valves (ERVs) and main steam piping due to operation at Extended Power Uprate. The assessment is contained in Attachment 1.

If you have any questions, please call me.

Prepared by

Kevin J. O'Hara
Associate

Reviewed by

Karen K. Fujikawa, P.E.
Associate

Approved by

Karen K. Fujikawa
Associate

kkf
Attachment: 1. Evaluation of Quad Cities Unit 2 Main Steam Line Vibration

cc: P. Hirschberg (SI)
K. J. O'Hara (SI)
G. L. Stevens (SI)

Evaluation of Quad Cities Unit 2 Main Steam Line Vibration

1.0 Introduction

The main steam system at Exelon's Quad Cities Unit 2 (QC2) plant has had a history of vibration issues. A number of failures have occurred in the condensate drain lines from the relief valves. The leak-off piping from valve 2-203-3E in particular has had at least five failures since 1985. Internal valve damage has been observed in the previously installed Electromatic relief valve (ERV) at this location. After replacement of these valves by a power operated relief valve (PORV) design, spurious lifting of the disc has occurred. Vibration data and evaluation has shown that the lines are subject to flow-induced vibration.

QC2 has replaced the PORV design relief valves with the ERV design used in Unit 1 and previously used in Unit 2. The predominant vibration frequency has been observed to be at the acoustic quarter-wave frequency of the relief valve branch line. Due to differences in valve internal dimensions, the acoustic frequency with the ERVs installed is somewhat lower than with the PORVs. Past experience prior to implementing Extended Power Uprate (EPU) has shown that the magnitude of the vibrations was higher with the ERVs installed than with the PORVs. Now that EPU has been implemented, there is a concern that the ERV design, coupled with the much higher EPU flow velocities, could result in an increase in the vibration levels.

The purpose of this letter report is to evaluate the vibration measurements recently obtained at QC2 and determine the adequacy of the ERVs and associated piping for long term EPU operation. Vibration measurements were taken at 33 accelerometer locations on the QC2 main steam piping at various operating levels up to 912 MWe (2796 MWth). Most of the accelerometers were located on the four ERVs and three adjoining pilot valves. The vibration time histories were converted to frequency domain spectra and grms (root-mean-square) values. The time histories were then filtered and maximum and minimum peak values were determined. Reference [1] contains details of the vibration data reduction.

2.0 Data Comparison/Assessment

2.1 Quad Cities 2 Dynamic Data Assessment

Vibration data was acquired at QC2 from March 29 through April 7, 2004. All data was assessed for peak vibration on the four main steam line (MSL) ERVs, their associated pilot valves and other MSL components. These components were assessed for ten power levels, from 200 (~25% power level) up to 912 MWe (2796 MWth). A data summary of the vibration levels is documented in Reference [1]. The maximum accelerations were observed at the ERV 3D and 3E inlet flanges and their associated pilot valves (see Figures 1 through 8). The peak amplitude for the ERV 3E inlet flange, and its associated pilot valve, was 1.18 grms at 160 hz, whereas, the peak amplitude for the ERV 3D inlet flange, and its associated pilot valve, was 1.52 grms at 152 hz. The vibration data trends for each channel are shown in Figures 5 through 8. From these figures, it can be seen that the vibration increases rapidly from 780 to 800 MWe and then increases proportionally to power level from 800 to 912 MWe.

The unfiltered root mean-square (RMS) values are shown in Table 1 and the acceleration time history trends for ERV 3E and 3D are shown in Figures 9 through 11.

2.2 Comparison to Quad Cities Unit 1 EPU Data

Vibration measurements were taken at 30 accelerometer locations on the Quad Cities Unit 1 (QC1) main steam piping at various operating levels up to EPU (2910 MWth). Most of the accelerometers were located on the four ERVs and two adjoining pilot valves, which were the locations of high vibration. Acceleration time histories were converted to frequency domain spectra and grms values. The acceleration time histories were then filtered and maximum peak values were determined. Reference [2] contains details of the vibration data reduction and Table 2 summarizes the RMS and peak accelerations for the ERVs and the Target Rock safety relief valve at 2910 MWth. For the locations that had redundant instruments, the highest values are shown.

The QC2 vibration data was compared to the QC1 vibration data. Specific maximum and RMS values are shown in Tables 1 and 2, whereas, Table 3 is the ratio of the QC2 vibration levels divided by the QC1 vibration levels. Values in Table 3 that are greater than 1.0 correspond to locations where measured QC1 data does not envelope QC2. Observations from Table 3 are as follows:

- QC2 ERV 3B inlet flange accelerations are lower than QC1 for the y-axis, equivalent for the z-axis, but higher for the x-axis.
- QC2 ERV 3B pilot valve accelerations are lower than QC1 for all axes.
- QC2 ERV 3E inlet flange accelerations are lower than QC1 for the x-axis, equivalent for the y-axis, but higher for the z-axis.
- QC2 ERV 3D inlet flange accelerations are higher than QC1 for all axes. The QC2 x-axis has a vibration level of 1.34 grms or a vibration level 10.3 times higher than QC1.
- QC2 ERV 3D pilot valve accelerations are higher than QC1 for all observed axes.
- QC2 ERV 3C inlet flange accelerations are lower than QC1 or equivalent for the y-axis, except for the z-axis peak acceleration.
- QC2 Target Rock 3A is slightly higher than QC1 for axes, except for the y-axis.

The highest QC2 vibration levels are seen on the ERV 3D and 3E on the x and z axes. The maximum accelerations for the QC1 and QC2 ERV inlet flange and pilot valve are tabulated below:

Max Vibration, grms	X-axis	Y-axis	Z-axis
QC-1 ERV Inlet Flange	0.18	4.17	0.47
QC-2 ERV Inlet Flange	1.34	1.71	0.49
QC-1 ERV Pilot Valve	0.56	2.04	1.52
QC-2 ERV Pilot Valve	0.18	1.64	0.82

In general, the QC1 vibrations are higher than QC2 for the y and z axes, but are lower for the ERV x-axis inlet, which is the axis that resulted in the vibration damage.

2.3 Comparison to QC1 Wyle Labs Data

The testing at Wyle Labs concluded that significant wear observed in the QC1 pilot valve actuators could be eliminated by replacing the brass bushings and stainless steel guide rods with Alloy X750 bushings and Alloy X750 guide rods. The endurance testing consisted of two levels of acceleration and test durations: 1) low frequency 20-70 hz at 0.8 grms and 2) high frequency 20-200 hz at 0.8 grms with sine sweeps (138-142 hz at 1.8 g, 154-158 hz at 3.0 g).

The QC2 actuators were modified to incorporate the design changes identified by the Wyle actuator testing. To verify that these design modifications are adequate, the QC2 RMS accelerations from Sections 2.1 and 2.2 above were compared to the Wyle Lab test acceleration levels. A review of Table 3 shows that the ERV 3D inlet and pilot valve accelerometers have the highest vibration levels (QC2 exceeds QC1 RMS values). Inspection of the ERV 3D inlet spectra plots indicate most of the energy (80%) is amassed at the discrete frequency of 152 hz. The QC2 ERV 3D inlet valve measured random and sine amplitudes are:

- y-axis (vertical to MS flow): 0.375 grms with a superimposed sine of 1.28 g (0-peak).
- z-axis (parallel to MS flow): 0.232 grms with a superimposed sine of 0.98 g (0-peak).

The QC2 ERV 3D z-axis, which is parallel to the pilot valve turnbuckle, is the same vibration direction that caused the greatest actuator wear during the Wyle actuator testing (x-axis direction for actuator only testing – test case 55). Comparing QC2 ERV 3D inlet data to the Wyle actuator test (test case 55) demonstrates that the QC2 vibration magnitudes are lower than the Wyle test levels. Therefore, the Wyle Labs actuator wear-aging tests bound the QC2 ERV 3D valve configuration, and this actuator should show negligible wear during EPU operation.

3.0 Sustained EPU Operation

3.1. Operation at 2796 MWth (912 MWe)

QC2 was monitored during plant start-up from March 29 through April 7, 2004. Thirty-three accelerometers were recorded for ten power levels up to EPU (912 MWe). All accelerometers show that vibration levels increased as power levels increased, but there was a significant jump in vibration levels from 780 to 800 MWe. For power levels above 800 MWe, the vibration levels increased uniformly with power level (see Figures 5 through 8). A comparison of the vibration magnitudes between QC2 and QC1 indicate that the maximum values were higher for ERV 3B and 3C, but lower ERV 3D and 3E. Based on the comparison with Wyle Labs testing, the QC2 ERVs are expected to show negligible wear during EPU operation.

Three hoop strain gages were also attached to the MSL B 20" pipe upstream of the 3E ERV and on each branch pipe for ERV 3E and 3B. The strain data was converted to pressure using thick-wall pipe equations and general strain theory [1]. The calculated dynamic pressure oscillations within the pipe were 7.8-9.2 psi rms. This represents a low level pressure oscillation within the pipe.

3.2. Operation at 2957 MWth

Operation at 2957 MWth represents a 5.8% increase in power level, based on mega-watts thermal. Vibration levels are expected to increase proportionally to main steam flow velocity

squared, which represents an 11.8% in increased velocity flow. Vibration increases can also be predicted from the vibration trend plots (Figures 5 through 8) and the RMS plots shown in Figures 9 through 11.

If QC2 continues to operate at 2796 MWth up to 2957 MWth, the ERV solenoid and pilot valve vibration levels are expected to increase proportionally. The acceleration levels were scaled by the same scaling factor (15%) as the piping assessment (Section 4.0). Therefore, the maximum composite acceleration level (ERV 3D inlet) would increase from 1.71 grms to 1.96 grms and the measured random floor would increase to 0.43 grms (less than 0.8 grms) with a superimposed sine of 1.47 g (less than 3.0 g). Since the vibration increase is still below the Wyle Labs actuator endurance test levels, the conclusions drawn from Section 2.3 above are still valid – negligible actuator wear would be observed.

4.0 MSL Piping Assessment

Four piping systems were analyzed to evaluate the effect of the main steam system vibration on the large and small bore piping [3]. These analyses were re-assessed to reflect the results of the QC2 measured vibration. For MSL B large bore piping, and for the small bore drain piping from ERVs 3B and 3E, the spectral peaks were increased according to the measured data at channels 22, 23, and 24, which are the highest location for MSL B. The data was taken at 2796 MWth, however operating power will eventually reach 2957 MWth. The accelerations were therefore increased by recognizing that flow induced vibration is generally proportional to flow velocity squared, and the flow velocity is approximately proportional to MWth. This correction amounts to 11.8%; the accelerations were increased by 15% and input to the piping analyses. The peak accelerations were applied over a range of ±10% around the two prominent vibration frequencies. The accelerations applied at frequencies away from the peak region clearly envelope the measured accelerations at those frequencies. For MSL D, a similar approach was used, with the peak values of channels 16, 17, and 18 applied. The broadened spectral peak was somewhat narrower than MSL B because the two peak frequencies are closer together.

The results of the analysis of the four piping models with QC2 as-measured vibration levels show that the calculated stresses are below the allowable stress per OM part 3 [4] of 3846 psi. Therefore, the piping can withstand the vibration loading measured in this system without sustaining any damage.

An evaluation of 52 small bore lines connecting to the main steam, HPCI, and RCIC systems in Quad Cities Unit 2 was also performed to assess the susceptibility of these lines to damage induced by vibration in the main steam system [5]. The vibration input conservatively enveloped the shapes of the main steam B and D response curves and applied them to all lines. The vibration input was also increased by an additional 15% to account for the expected increase in vibration from 2796 MWth, at which the data was taken, to 2957 MWth, the expected maximum power. A screening process was used, in which thirty-one of the lines were analyzed in detail, and 16 of the lines were accepted by review of the piping isometrics and application of engineering judgment. For all of the analyzed lines, the vibration stress calculated for each pipe was below the stress allowable recommended by the OM Part 3 Standard.

5.0 Feedwater Regulator Valve

The measured vibration at the feedwater regulator valve shows a peak acceleration of only 0.02g. A similar valve (16" instead of 14") has been seismically qualified for 3.7g horizontal and 2g vertical. Thus, this valve is structurally adequate by inspection due to the low magnitude of measured vibration.

6.0 Conclusions

Based on an assessment of the measured QC2 vibration levels to the QC1 measured vibration levels and the Wyle Labs test results, the QC2 ERVs are expected to experience negligible wear due to EPU operation (2957MWth).

From the results of the main steam piping analyses, the stresses in the main steam piping are acceptable for the vibration levels measured at 2796 MWth. The accelerations measured at power levels below 2796 MWth are enveloped by the 2796 MWth accelerations. In addition, the acceleration values used in the piping analyses have been increased to account for operation at full licensed power of 2957 MWth. The resulting stresses meet the criteria of OM Part 3 [4], which contain a number of conservatisms and safety factors.

It is recommended that vibration data be taken at 2957 MWth to confirm that the extrapolated values assumed for 2957 MWth bound the measured QC2 vibration levels at 2957 MWth.

7.0 References

1. Structural Integrity Associates Calculation No. QC-16Q-303, Revision 0, "Quad Cities Unit 2 ERV Vibration Data Reduction."

2. Structural Integrity Associates Calculation No. QC-11Q-302, Revision 0, "Quad Cities Unit 1 Main Steam Line Vibration Data Reduction."

3. Structural Integrity Associates Report No. SIR-03-136, Revision 3, "Evaluation of Main Steam Line Vibration for Quad Cities Unit 2 PORV Replacement," SI File No. QC-07Q-402.

4. ASME OM-S/G Part 3, 1994 Edition.

5. Structural Integrity Associates Report No. SIR-04-034, Revision 1, "Evaluation of Effects of Main Steam Vibration on Attached Small Bore Lines, Quad Cities Unit 2", SI File No. QC-12Q-401.

Table 1. QC2 ERV Measured Accelerations at 912 MWe (2796 MWth)

Location	Acceleration (g)					
	X peak	Y peak	Z peak	X rms	Y rms	Z rms
ERV 3B Inlet (PVMS Ch 7, 8, 9)	0.92	1.27	0.88	0.26	0.38	0.21
ERV 3B Pilot (PVMS Ch 10, 11, 12)	1.33	2.72	3.25	0.37	0.76	0.82
ERV 3E Inlet (PVMS Ch 1, 2, 3)	0.71	4.19	1.34	0.13	1.54	0.49
ERV 3E Pilot (new chan) (PVMS Ch 4, 5, 6)	1.75	4.18	1.99	0.58	1.54	0.54
ERV 3D Inlet (PVMS Ch 16, 17, 18)	4.30	4.67	5.81	1.34	1.71	0.38
ERV 3D Pilot (PVMS Ch 19, 20, 21)	2.60	4.66	3.97	0.68	1.64	1.18
ERV 3C Inlet (PVMS Ch 13, 14, 15)	1.05	2.19	3.46	0.27	0.77	0.27
TRV 3A Inlet (PVMS Ch 28, 29, 30)	1.29	4.38	0.69	0.38	1.24	0.19

Notes: 1. Inlet Valve: X is perpendicular to the main steam line, Y is vertical and Z is parallel to the steam flow.
2. Pilot Valve: X is perpendicular to the main steam line, Y is vertical and Z is parallel to the steam flow.

Table 2. QC1 ERV Measured Accelerations at 2910 MWth

Location	Acceleration (g)					
	X peak	Y peak	Z peak	X rms	Y rms	Z rms
ERV 3B Inlet (QCTD Ch 1, 2, 3, 15)	0.32	2.20	0.68	0.08	0.70	0.18
ERV 3B Pilot (QCTD Ch 4, 5, 6, 13, 14)	1.88	5.40	5.46	0.56	2.04	1.52
ERV 3E Inlet (QCD Ch 7, 8, 9)	0.73	3.32	0.53	0.18	1.01	0.13
ERV 3D Inlet (DTD Ch 4, 5, 6)	0.47	1.95	0.89	0.13	0.67	0.18
ERV 3D Pilot (DTD Ch 7, 8, 9)	na	2.28	1.54	na	0.75	0.56
ERV 3C Inlet (DTD Ch 10, 11, 12)	na	8.76	1.12	na	4.17	0.47
TRV 3A Inlet (DTD Ch 1, 2, 3)	1.01	3.24	0.71	0.22	0.36	0.15

Notes: 1. X is perpendicular to the main steam line, Y is vertical and Z is parallel to the steam flow.
2. na - instruments were bad
3. QCTD – Quad Cities Tape Deck
4. DTD – Dresden Tape Deck

Table 3. QC2 versus QC1 ERV Acceleration Comparison at 912 MWe

Location	Acceleration Ratio - QC2/QC1					
	X peak	Y peak	Z peak	X rms	Y rms	Z rms
ERV 3B Inlet	2.88	0.58	1.29	3.25	0.54	1.17
ERV 3B Pilot	0.71	0.50	0.60	0.66	0.37	0.54
ERV 3E Inlet	0.97	1.26	2.53	0.72	1.52	3.77
ERV 3D Inlet	9.15	2.39	3.01	10.31	2.55	2.11
ERV 3D Pilot	-	2.39	3.01	-	2.55	2.11
ERV 3C Inlet	-	0.25	1.49	-	0.18	0.57
TRV 3A Inlet	1.28	1.35	0.97	1.73	3.44	1.27

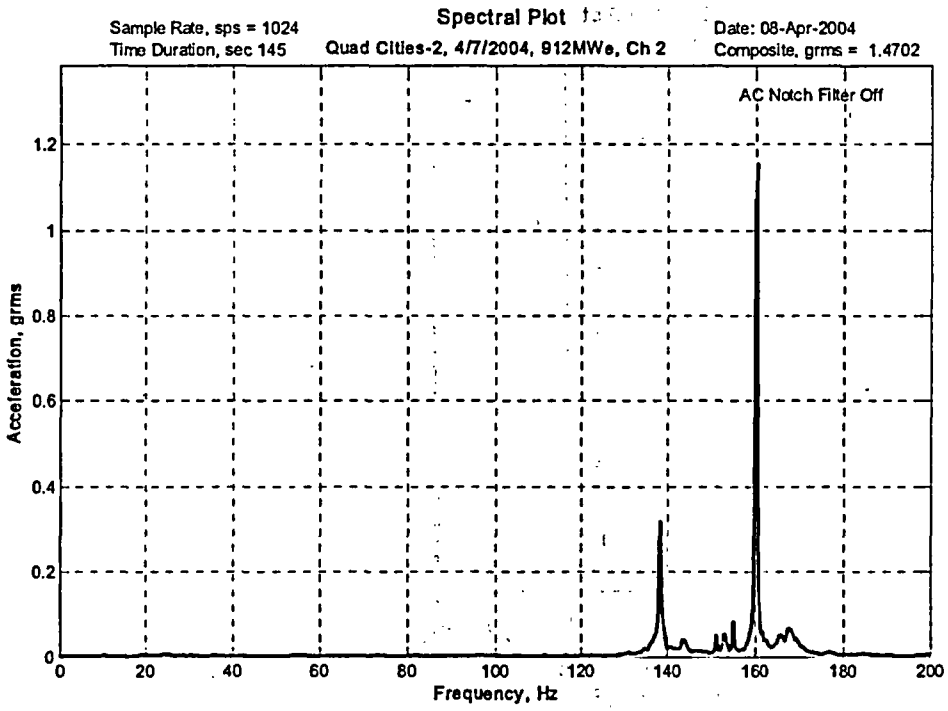

Figure 1: ERV 3E Inlet Flange, y-axis Vibration at 912 MWe

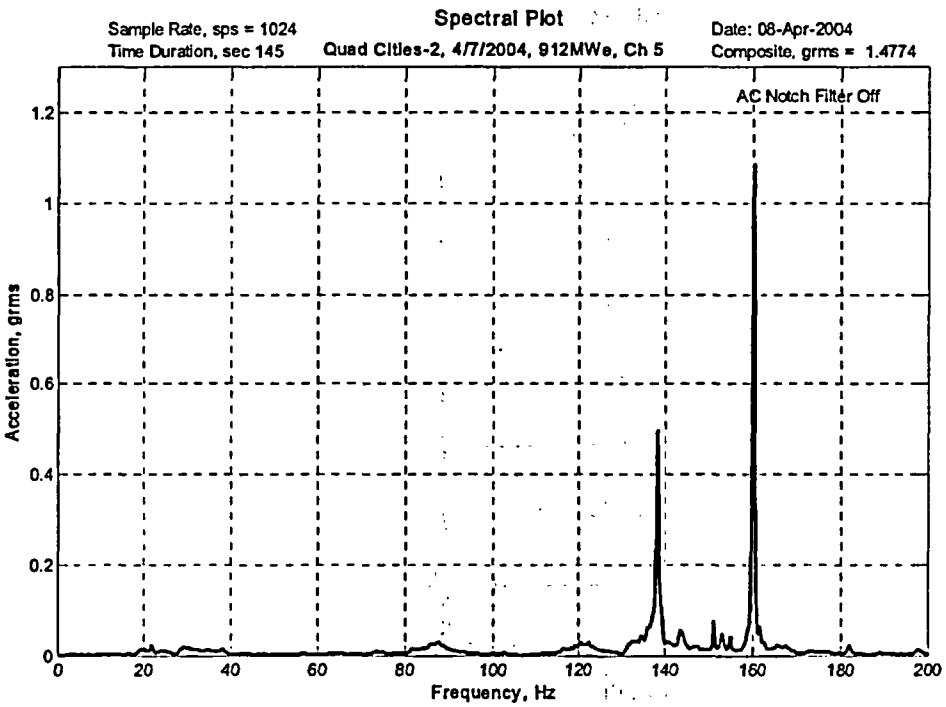

Figure 2: ERV 3E Pilot Valve, y-axis Vibration at 912 MWe

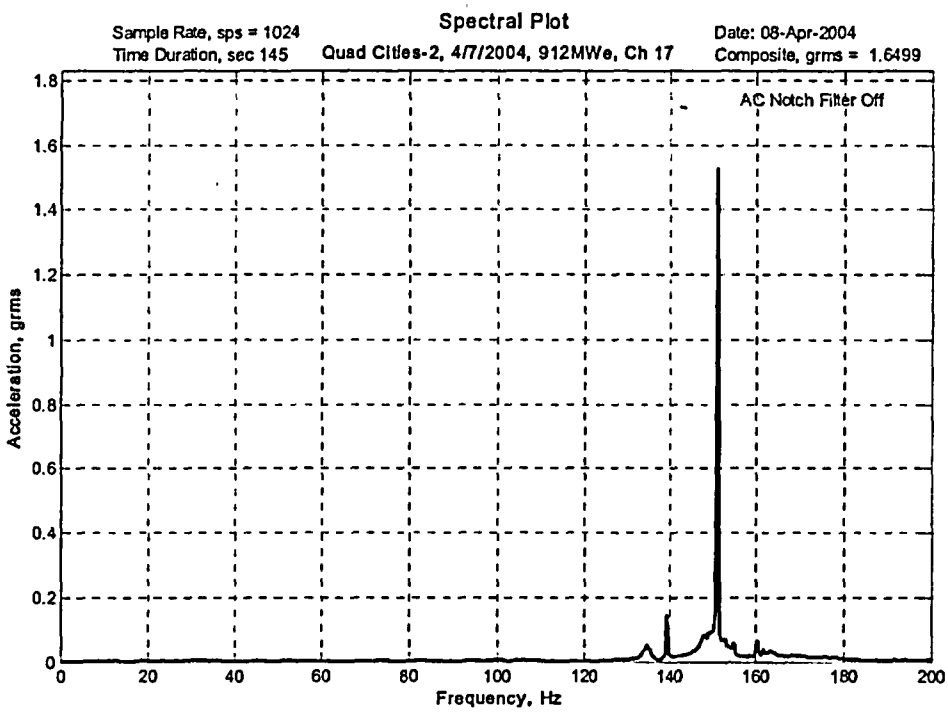

Figure 3: ERV 3D Inlet Flange, y-axis Vibration at 912 MWe

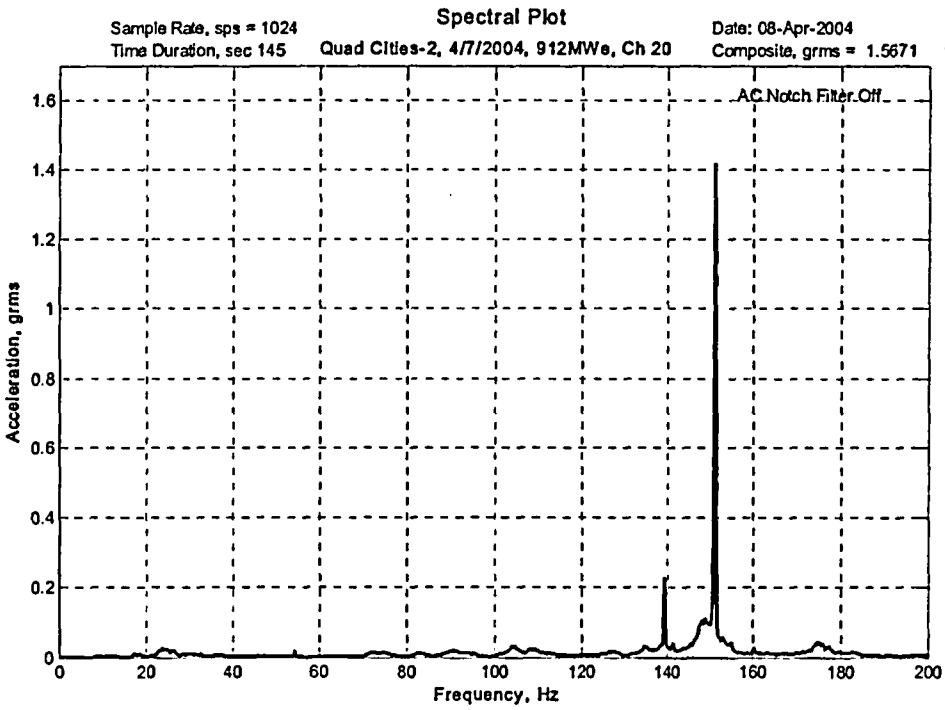

Figure 4: ERV 3D Pilot Valve, y-axis Vibration at 912 MWe

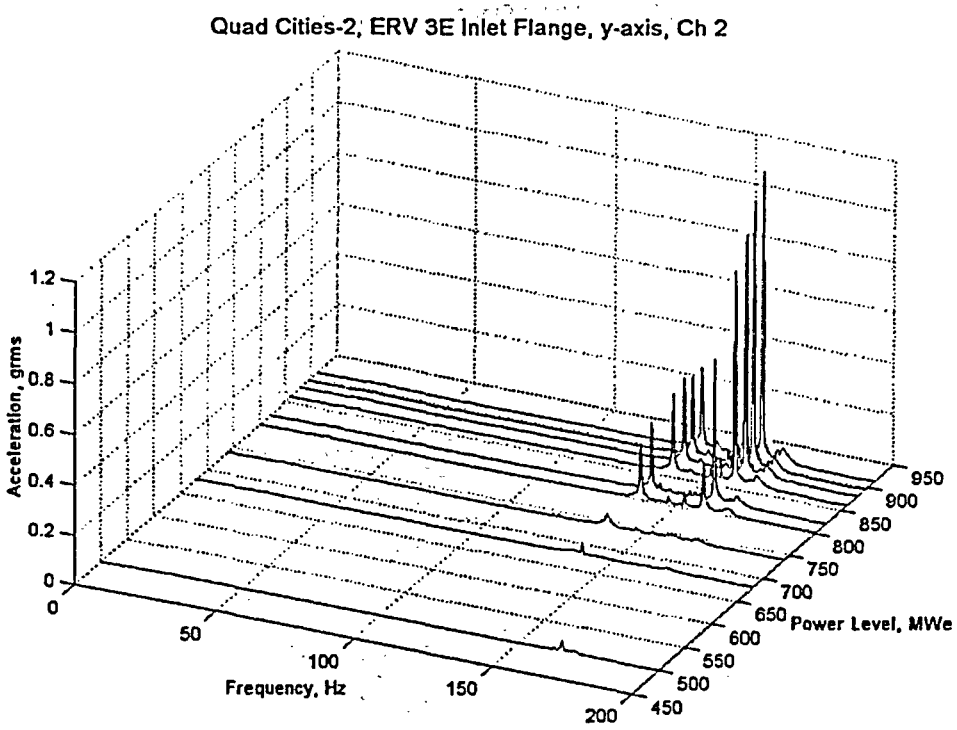

Figure 5: ERV 3E Inlet Flange, y-axis Vibration Trends to 912 MWe

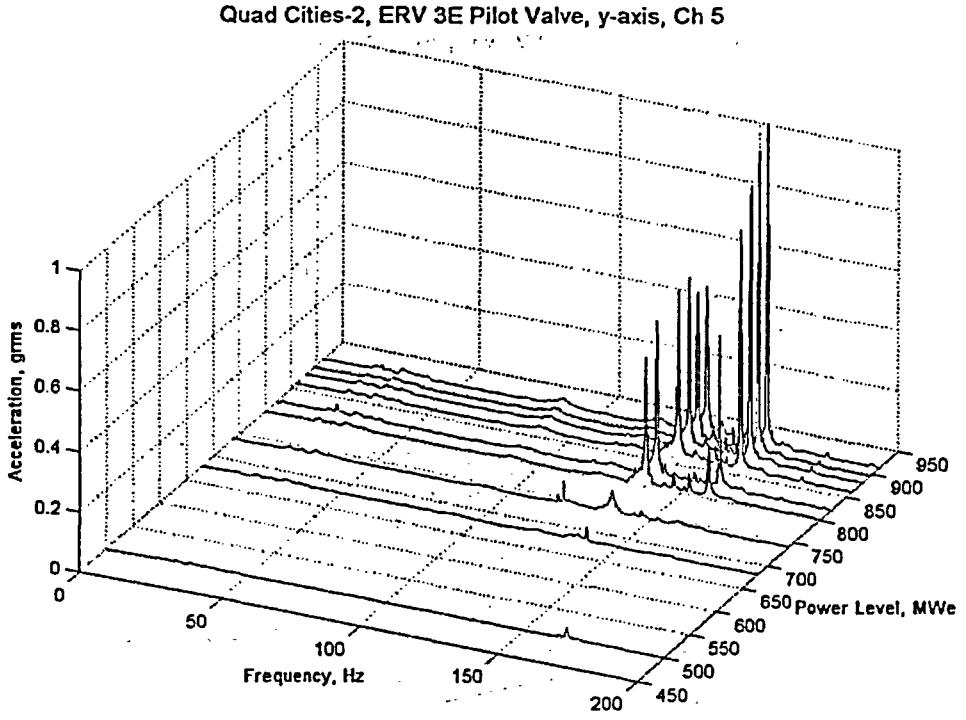

Figure 6: ERV 3E Pilot Valve, y-axis Vibration Trends to 912 MWe

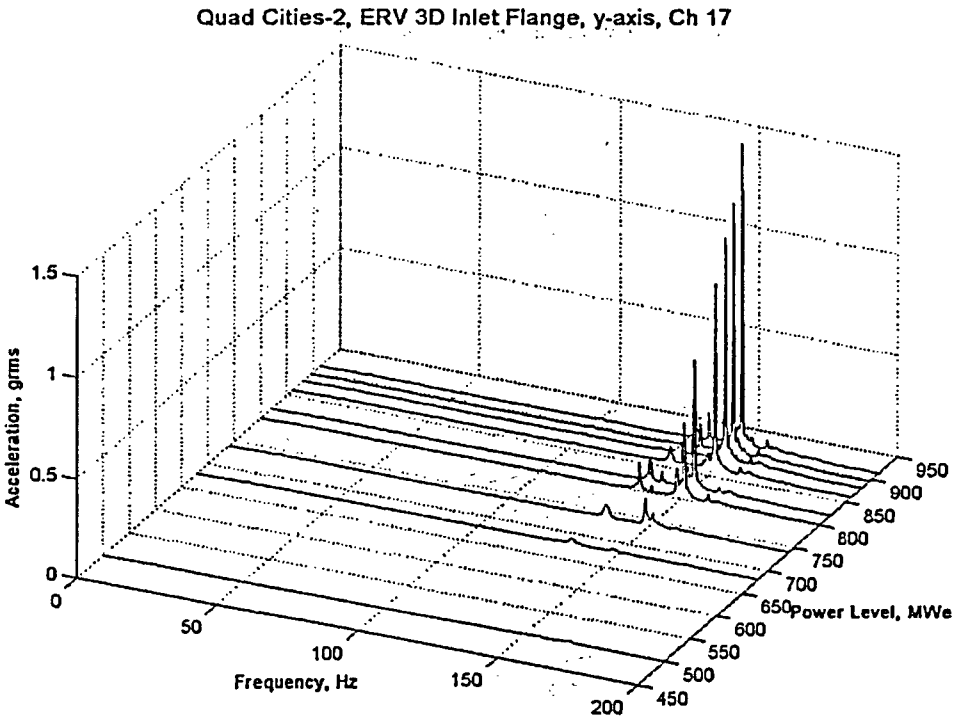

Figure 7: ERV 3D Inlet Flange, y-axis Vibration Trends to 912 MWe

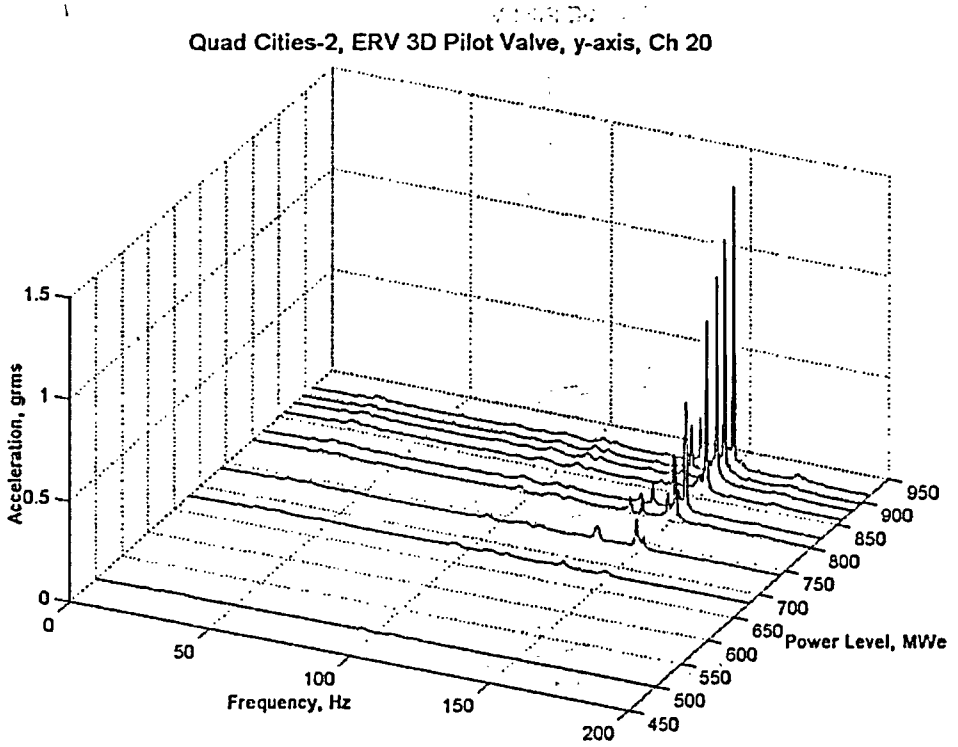

Figure 8: ERV 3D Pilot Valve, y-axis Vibration Trends to 912 MWe

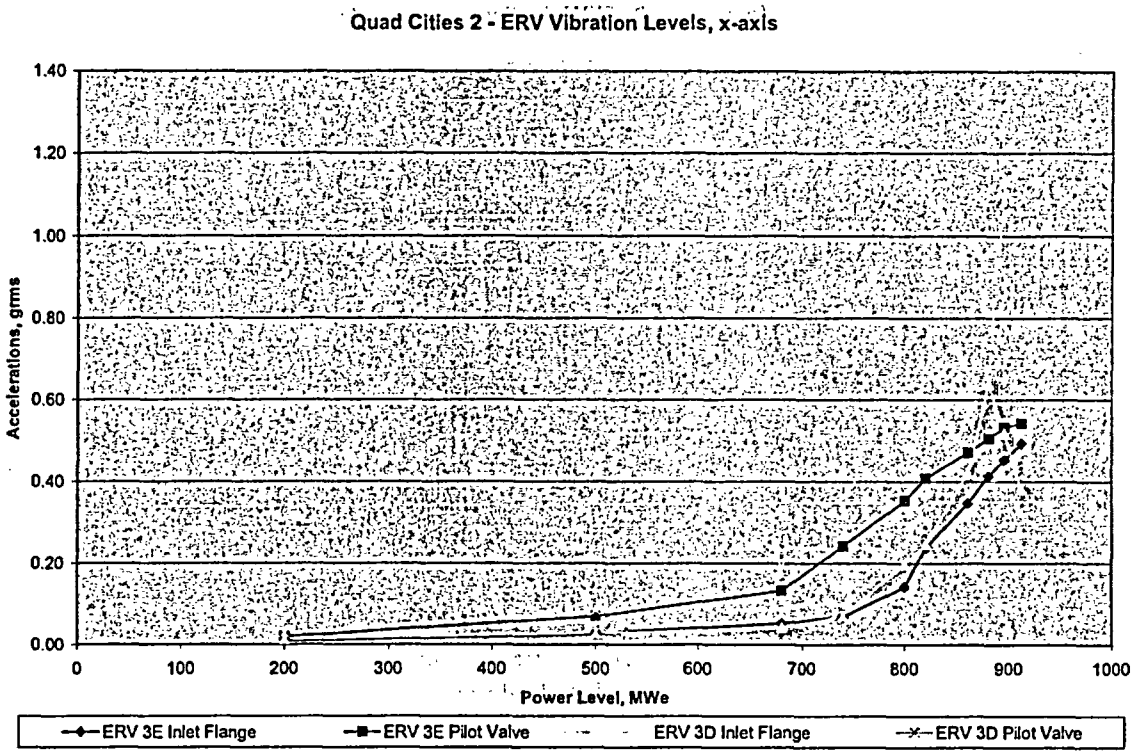

Figure 9: ERV x-axis Vibration Trends to 912 MWe

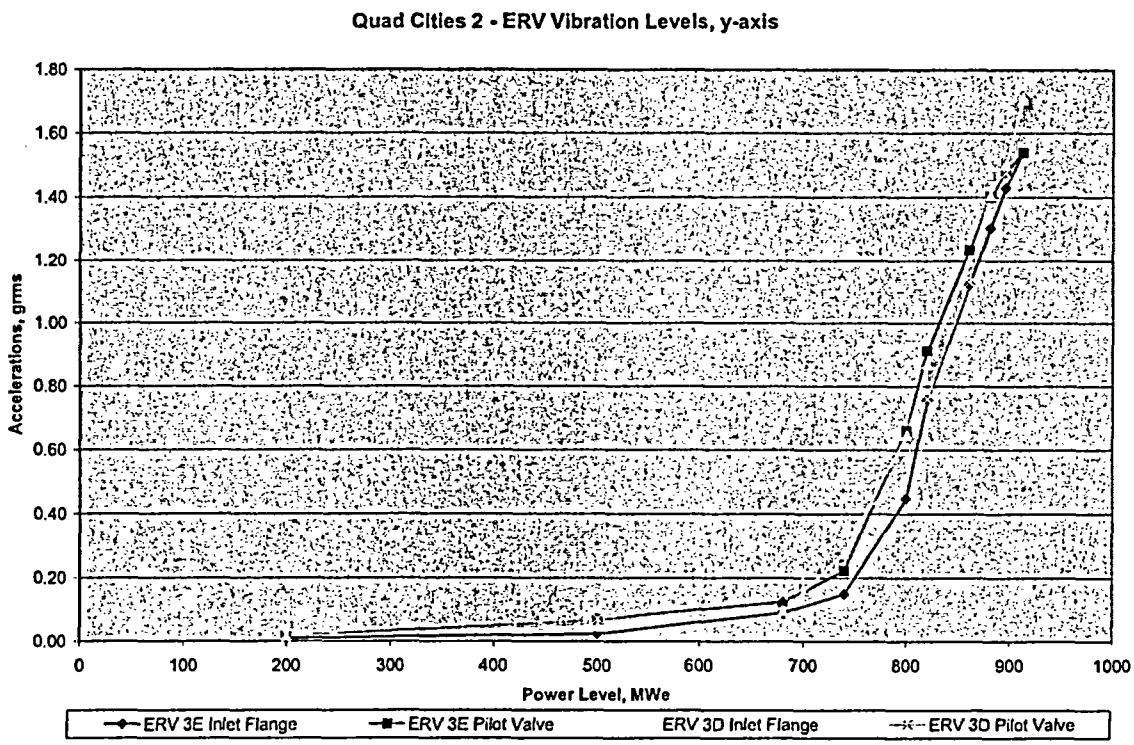

Figure 10: ERV y-axis Vibration Trends to 912 MWe

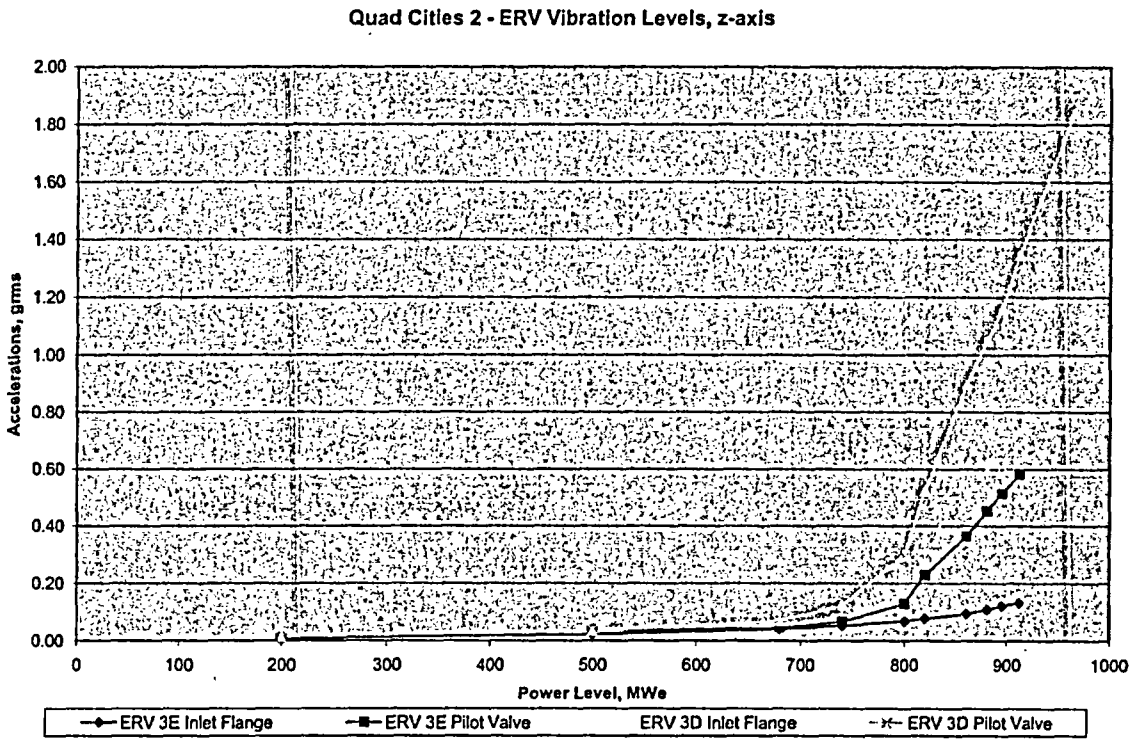

Figure 11: ERV z-axis Vibration Trends to 912 MWe

ATTACHMENT 3

Transmittal of Design Information QDC-04-16 – Inspection Information from Q2R17 MSIV Internal Inspections

EXELON TRANSMITTAL OF DESIGN INFORMATION

X SAFETY-RELATED __ NON-SAFETY-RELATED __REGULATORY RELATED	Originating Organization X Exelon __Other (specify)	TODI No. QDC-04-16
Station Quad Cities Unit(s) Two System Designation: RX		Page 1 of 2 To General Electric

Subject MSIV inspection results during Q2R17

Pat Yost Preparer	Pat Yost Preparer's Signature	4-15-04 Date
Tom Wojcik Approver	Approver's Signature	04.15.04 Date

Status of Information: X Approved for Use __Unverified

Method and Schedule of Verification for Unverified TODIs: N/A

Description of Information:

Observations made by preparer (AOV Program Engineer) during disassembly and visual inspection of 1A, 2B, 2C and 1D MSIV's. Resulting inspection notes are provided on page 2.

Purpose of Issuance:

Support evaluation of MSIV's for acceptability for full EPU power operation.

Limitations:

None.

Source Documents:

Personal observations made by TODI preparer.

Distribution: Nancy Finnicum (Records Management), Sharon Eldridge

Q2R17 inspection results from 1A, 2B, 2C and 1D MSIV's

- The 1A lower liner had no wear and no evidence of steam erosion through flow windows. No issue with the 1A upper liner though the area where the piston ring is located showed rubbing/polishing (no obvious metal loss). No evidence of leak by past the main plug seat or pilot plug seat. The pilot assembly retainer plate had minor wear on side in contact with pilot assy when valve open.

- The 2B lower liner had no wear and no evidence of steam erosion through flow windows. The 2B upper liner ID had a minor groove being worn into the ID surface in the area where the plug piston ring rides when full open. The wear area was about the thickness of the piston ring, was about 3 inches long and about 0.005" - 0.008" deep. No evidence of leak by past the main plug seat or pilot plug seat. The pilot disc plate had wear indications from the pilot plug. The wear areas were heaviest 180 degrees opposite each other indicating that the main plug was likely rocking on the pilot plug when full open. The wear was about 0.015" deep at the deepest.

- The 2C lower liner had no wear and no evidence of steam erosion through flow windows. The 2C upper liner had no measurable wear on the ID but was polished by the piston ring in one location. No evidence of leak by past the main plug seat or pilot plug seat. The pilot disc plate had wear indications from the pilot plug. The wear areas were heaviest 180 degrees opposite each other indicating that the main plug was likely rocking on the pilot plug when full open. The wear was about 0.015" deep at the deepest.

- The 1D lower liner had no wear and no evidence of steam erosion through flow windows. The 1D upper liner had a section (~ 40% of circumference) where the piston ring wore on ID. Wear depth was about 0.005 - 0.010" deep. No evidence of leak by past the main plug seat or pilot plug seat. The pilot assembly retainer plate had minor wear on side in contact with pilot assy when valve open.

ATTACHMENT 4

MPR Independent Review Letter

MPR
ASSOCIATES INC.
ENGINEERS

April 30, 2004

Ms. Sharon Eldridge
Lead Project Engineer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

Subject: Review of EC-EVAL #348316, "Evaluation of Quad Cities Unit 2 Main Steam Line Vibrations at EPU Power Levels," Revision 1

Dear Ms. Eldridge:

Exelon recently provided MPR with EC-EVAL #348316, "Evaluation of Quad Cities Unit 2 Main Steam Line Vibrations at EPU Power Levels," Revision 1, for review. The EC-EVAL summarizes technical analyses that assess the impact of operating Quad Cities Unit 2 (QC2) at power levels up to 2957 MWth. MPR has completed a review of the EC-EVAL and concluded that Exelon has taken effective compensatory actions and completed analyses to demonstrate that components susceptible to accelerated vibration damage will maintain their integrity to the end of the current operating cycle.

Exelon has restricted operation of QC2 to power levels no higher than 2511 MWth until the root cause analysis of a Target Rock valve is completed. Although this action is warranted, MPR experience with these valves indicates that the cause of the test failure may not be apparent even after the analyses are concluded. There may not be an obvious connection between higher vibration levels and the increased actuation pressure observed in the test failure. Therefore, Exelon may have difficulty eliminating the power level restriction for QC2.

Please call me or Bill McCurdy if you have any questions regarding this letter.

Sincerely,

Phillip J. Rush, P.E.

320 KING STREET ALEXANDRIA, VA 22314-3230 703-519-0200 FAX: 703-519-0224 http:\\www.mpr.com

ATTACHMENT 5

Stevenson and Associates Independent Review Letter

Stevenson & Associates

A structural-mechanical consulting engineering firm

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E-MAIL: info@vecsa.com
WWW: www.stevenson-associates.com

BOSTON•CLEVELAND•CHICAGO•SANTA CRUZ•FT. WORTH

April 22, 2004

Ms. Sharon Eldridge
Exelon Corporation
4300 Winfield Road
Warrenville, IL 60555

Subject: Review of Quad Cities Main Steam ERV Vibrations at EPU Levels

References:
1) Report QC-16Q-303: Quad Cities Unit 2 ERV Vibration Data Reduction
2) Report SIR-03-136 (Rev 3): Main Steam Line Vibration for QC2 PORV
3) Report SIR-040048 (Rev 0): QC2 Main Steam Vibration Assessment
4) GE Report DRF 0000-0023-4260: Eval. QC2 MSIV & SRV Vibration
5) Target-Rock SRV Vibration Aging Test Requirements
6) Report SIE-04-23: ERV Vibration Testing Assessment
7) Report QDC-0200-M-1380: Evaluation of Components for Vibration Effects

Dear Ms. Eldridge:

I have reviewed the referenced reports. My overall conclusion is that you have correctly diagnosed the primary vibration problem of the ERVs and that your proposed solutions should allow you to successfully operate through an entire fuel cycle at EPU power levels.

With regard to the individual reports, I offer the following comments, conclusions and recommendations for the record:

<u>1) Report QC-16Q-303: Quad Cities Unit 2 ERV Vibration Data Reduction</u>
It is a presentation of the vibration data collected in 2003 and 2004, and the observations are valid. The vibration levels do indeed increase significantly above 800 MWe power levels and this should remain a concern going forward. It appears that the 160 Hz tone increases more, and more frequently, than the 138 Hz tone. These vibration levels remain high (excessive?) for some of the components and may continue to pose problems for some components (e.g., limit switches, MSL and valves, and the leak-off line) in the future.

2) Report SIR-03-136 (Rev 3): Main Steam Line Vibration for QC2 PORV
a)The recommendations for short-term and long-term improvements are reasonable.
b)I think it is interesting that the leak-off piping has a significant mode at 160 Hz (see Table 4-5). A different supporting scheme has been installed to address this vulnerability..
c)As a general comment the vibration stresses, while not high, are also not small (2-3 ksi). The high number of cycles that the system undergoes, however, may cause some future – by that I mean years out - high-cycle, low stress fatigue failures, particularly at Quad Cities at EPU levels. I do not conclude that these stress levels pose a current operating cycle concern at EPU levels, but extended exposure (operating cycle after operating cycle) may lead to cumulative fatigue damage. To that end, it is suggested that the recommendations be investigated to ultimately achieve lower vibration levels at Quad Cities and I understand that it is being done.

3) Report SIR-04-0048 (Rev 0): QC2 Main Steam Vibration Assessment
I concur with the primary conclusion in Section 2.3 that negligible actuator wear is forecast and that the use of the improved bushing materials and posts will be successful. Regarding the MSL piping assessment in Section 4, I refer back to the previous report (SIR-03-136) and simply offer that the stresses are not insignificant and, thus, may add up over time to cause high-cycle albeit low stress cumulative damage. As such, reducing vibration levels is a desirable goal.

4) GE Report DRF 0000-0023-4260: Eval. QC2 MSIV & SRV Vibration
I concur with GE's overall conclusions that the valves will function without significant in-cycle fatigue or wear. "Seismic aging" may not be much of an indicator given that seismic aging is a low-cycle test by definition. The wear aging that is occurring a high cycle phenomenon and can best be measured by operating experience, detailed inspection and analysis, and explicit wear-aging testing. My understanding is that the SRVs are being tested and I concur that this is prudent course of action.

5) Target-Rock SRV Vibration Aging Test Requirements
As alluded to immediately above, I doubt that it will be possible to identify wear aging in a 90-minute aging test. I believe a much longer test would be needed to get to the cycle equivalency required. However, if there is valve sensitivity at the discrete tones to be tested, it should become evident. Once again, the testing that Exelon is undertaking is the preferable route.

6) Report QDC-0200-M-1380: Evaluation of Components for Vibration Effects
I generally do not believe that seismic test amplitudes being higher than ambient vibration levels, even with cycle equivalencies, will predict any kind of invulnerability of the component. For temperature elements and other monolithic components with no moving parts it is probably fine to establish that they should not be vibrationally vulnerable. With respect to limit switches and valve operators, the seismic test is not much of predictor of vibrational sensitivity. As with the ERV bushings, the subcomponent responses are the key elements. A seismic test will not

necessarily have energy content at the higher frequencies - like 138 Hz – and, thus, will not excite the modes of interest. Inspections, operating histories, and shake table testing sine sweeps over the frequency range of concern, like that conducted for the ERVs, are better indicators. As an example, if there are concerns about limit switches, then they should be placed on a shake-table and be subjected to a slow swept sine input and some dwell testing at discrete tones. My understanding is limit switches and valve operators are, in fact, being tested and I concur with this course of action for components with moving subassemblies.

7) Report SIR-04-23: ERV Vibration Testing Assessment
I completely agree with the Summary and Conclusions, and believe this test report has correctly identified the immediate ERV vibration problem and the resulting solution path.

If there are any questions or comments regarding my assessments or conclusions please contact me directly at the letterhead phone number at extension 21.

Very truly yours,

Walter Djordjevic
Senior Consultant

EC 348316, Revision 2
Attachment 6
Summary of Walkdown Scope

Components	Commodities	What	How	Existing Activity to Credit
All Four MSLs	Venturi Small Bore	Attachment and First Support	Visual	
All Four MSLs	Drain Lines	Attachment and First Support	Visual	
All Four MSLs	Penetrations	Bellows, Rods, Pins	Visual	
All Four MSLs	Supports	Bolts, Nuts, Rubbing	Visual	
All Four MSLs	Snubbers	Damage, Wear	Visual	
MO 2-0220-1	LLRT Taps	Attachment	Visual	
MO 2-0220-2	LLRT Taps	Attachment	Visual	
HPCI -4 Valve	Penetration	Bolts, Pins	Visual	
HPCI -4 Valve	LLRT Taps	Attachments/Supports	Visual	
HPCI -5 Valve	Penetration	Bolts, Pins	Visual	
HPCI -5 Valve	LLRT Taps	Attachments/Supports	Visual	
RCIC Inboard PCI	Penetration	Bolts, Pins	Visual	
RCIC Inboard PCI	LLRT Taps	Attachments/Supports	Visual	
RCIC Outboard PCI	Penetration	Bolts, Pins	Visual	
RCIC Outboard PCI	LLRT Taps	Attachments/Supports	Visual	
All Four MSLs	Supports	Bolts, Nuts, Rubbing	Visual	
All Six CIVs	Hangers/Supports	Damage, Wear	Visual	
SJAE Steam Supply	Hangers/Supports	Damage, Wear	Visual	
MSL D	RCIC tie-in	Attachment and First Support	Visual	
MSL C	HPCI tie-in	Attachment and First Support/Vents/Drains	Visual	
MS Small Bore Lines		Attachment and First Support – all supports on these lines should be inspected to be sure they carry load and are intact	Visual	
All Four PORVs	Vacuum Breakers	Appear closed/Stroke?	Visual	Walkdowns to be performed prior to removal per Installers walkdown of 597866

EC 348316, Revision 2
Attachment 6
Summary of Walkdown Scope

Components	Commodities	What	How	Existing Activity to Credit
All Four Inboard MSIV	Limit Switches	Mounting, Lever Arm, Wires	Visual	2-0203-1A, B, C, and D inboard limit switch inspections will take place per 00481703-02D
All Four PORVs	Acoustic Monitors	Mounting, Wires	Visual	Walkdowns to be performed prior to removal per Installers walkdown of 597866
All Four PORVs	TE	Mounting, Wires	Visual	Walkdowns to be performed prior to removal per Installers walkdown of 597866
All Four PORVs	Junction Box	General Inspection	Visual	Walkdowns to be performed prior to removal per Installers walkdown of 597866.
All Four inboard MSIV	IA Leaks Snoop	Air Leak	Snoop	Inboard MSIV accumulator leak test will be per 00481476-01
Level 2 Equipment		Signs of Vibration-Related Degradation	Visual	
All Four Inboard MSIV	TE	Mounting, Wires	Visual	Inboard T/E inspections will take place per 00481573-01B
RPS Limit Switches		Mounting, Wires	Visual	
Drywell Level Indication		Attachment and First Support	Visual	
B and C Outboard MSIVs	Limit Switches and Actuators	Mounting, Lever Arm, Wires, Springs, Bolts	Visual	B and C outboard actuator and limit switch inspections will take place per (00448532-01, 00481582-01, 00481581-01, 99253318-C02)

EC 348316, Revision 2
Attachment 6
Summary of Walkdown Scope

Components	Commodities	What	How	Existing Activity to Credit
Target Rock S/RVs	Flex Hose	Attachment and Any Wear	Visual	
Target Rock S/RVs	PS 2-0262-37A, B and C	Switch, Tubing and First Support	Visual	
Target Rock S/RVs	PS 2-0203-34A	Switch, Tubing and First Support	Visual	
Target Rock S/RVs	Accumulator Tubing	Attachment and First Support	Visual	
Target Rock S/RVs	SO 2-0203-3A	Mounting, Wires	Visual	
Target Rock S/RVs	Acoustic Monitor	Mounting, Wires	Visual	
Target Rock S/RVs	Vacuum Breaker	Appear Closed/Stroke?	Visual	
All Four Outboard MSIV	IA Leaks Snoop	Air Leak	Snoop	Inboard MSIV accumulator leak test will be per 00481476-01
All Four Outboard MSIV	TE	Mounting, Wires	Visual	Outboard T/E inspections will take place per 00481573-02B
All Eight Safety Valves	Small Bore Lines	Attachment and First Support	Visual	
All Eight Safety Valves	Acoustic Monitors	Mounting, Wires	Visual	
All Eight Safety Valves	TE	Mounting, Wires	Visual	
Electrohydraulic Control (EHC)	TCV Accumulators	Mounting	Visual	Per existing activity 545268-03
EHC	Lines Near TCVs, TSVs, and BPVs	Attachments and First Support	Visual	Per existing activity 545268-03
All Four Inboard MSIV	Actuators	Springs, Bolts	Visual	Inboard MSIV 1A,C,D actuator inspections will be per 00596979-01, 596980-01, and 596982-01. Yost will still perform Visual inspection for 0203-1B actuator/manifold
All Four Inboard MSIV	Air Lines/Small Valves	Attachment and First Support	Visual	

EC 348316, Revision 2
Attachment 6
Summary of Walkdown Scope

Components	Commodities	What	How	Existing Activity to Credit
MO 2-0220-1	Operator	Non-intrusive	Stroke	Unit 1 walkdowns during Q1R17 identified no vibration related issues on this equipment. This equipment will not be specifically inspected during Q2R17
MO 2-0220-1	Limit Switch	Mounting, Wires	Visual	Unit 1 walkdowns during Q1R17 identified no vibration related issues on this equipment. This equipment will not be specifically inspected during Q2R17
All Four Inboard MSIV	Solenoids	Mounting, Wires	Visual	
MO 2-0220-2	Operator	Non-intrusive	Stroke	VOTES testing per 609587, no additional walkdowns required
MO 2-0220-2	Limit Switch	Mounting, Wires	Visual	VOTES testing per 609587, no additional walkdowns required
All Four Inboard MSIV	Flex Line	Attachment and Any Wear	Visual	
All Four Inboard MSIV	Junction Box	General Inspection	Visual	
A and D Outboard MSIV	Limit Switches and Actuators	Mounting, Lever Arm, Wires, Springs, Bolts	Visual	A and D outboard actuator and limit switches may take place (contingent on LLRT) per (99149021-A02, 00481584-01, 00539221-01,0048158-01). Yost will complete walkdowns if contingencies not worked

EC 348316, Revision 2
Attachment 6
Summary of Walkdown Scope

Components	Commodities	What	How	Existing Activity to Credit
HPCI Four Valve	Operator	Non-intrusive	Stroke	Testing per 646225, no additional walkdowns required
HPCI Four Valve	Limit Switch	Mounting, Wires - the four-rotor assembly for this valve should be removed and replaced with the as-found switch quarantined and inspected per criteria to be provided by Sargent and Lundy	Visual	Testing per 646225, no additional walkdowns required
All Four Outboard MSIV	Air Lines, Small Valves	Attachment and First Support	Visual	
All Four Outboard MSIV	LLRT Taps	Attachment	Visual	
HPCI -5 Valve	Operator	Non-intrusive	Stroke	Unit 1 walkdowns during Q1R17 identified no vibration related issues on this equipment. This equipment will not be specifically inspected during Q2R17
HPCI -5 Valve	Limit Switch	Mounting, Wires	Visual	Unit 1 walkdowns during Q1R17 identified no vibration related issues on this equipment. This equipment will not be specifically inspected during Q2R17
All Four Outboard MSIV	Solenoids	Mounting, Wires	Visual	
All Four Outboard MSIV	Flex Line	Attachment and Any Wear	Visual	
All Four Outboard MSIV	Junction Box	General Inspection	Visual	
Drywell Coolers		Non-intrusive	Visual	

EC 348316, Revision 2
Attachment 6
Summary of Walkdown Scope

Components	Commodities	What	How	Existing Activity to Credit
RCIC Inboard PCI	Operator	Non-intrusive	Stroke	Unit 1 walkdowns during Q1R17 identified no vibration related issues on this equipment. This equipment will not be specifically inspected during Q2R17
RCIC Inboard PCI	Limit Switch	Mounting, Wires - the four-rotor assembly for this valve should be inspected and replaced if any degradation noted with the as-found switch quarantined and inspected per criteria to be provided by Sargent and Lundy	Visual	Unit 1 walkdowns during Q1R17 identified no vibration related issues on this equipment. This equipment will not be specifically inspected during Q2R17
RCIC Outboard PCI	Operator	Non-intrusive	Stroke	Testing per 587694, no additional walkdowns required
RCIC Outboard PCI	Limit Switch	Mounting, Wires	Visual	Testing per 587694, no additional walkdowns required
RWCU 2-1201-2	Limit Switch	Mounting, Wires	Stroke	Testing per 99272081, no additional walkdowns required.
MO 2-1201-2	Operator	Non-intrusive	Stroke	Testing per 99272081, no additional walkdowns required
All Four MSLs	Snubbers	Damage, Wear	Visual	To be completed per snubber program

EC 348316, Revision 2
Attachment 6
Summary of Walkdown Scope

Components	Commodities	What	How	Existing Activity to Credit
Level 3 Equipment		Signs of vibration-related degradation	Visual	Unit 1 walkdowns during Q1R17 identified no vibration related issues on this equipment. This equipment will not be specifically inspected during Q2R17.
Level 4 Equipment		Signs of vibration-related degradation	Visual	Unit 1 walkdowns during Q1R17 identified no vibration related issues on this equipment. This equipment will not be specifically inspected during Q2R17.

ATTACHMENT 1D

Engineering Change Evaluation 347006, Revision 1 "Evaluation (EVAL) of Dresden Unit 3 Main Steam Line (MSL) Vibrations at Extended Power Uprate (EPU) Power Levels"

Engineering Change 347006, Revision 1
Evaluation (EVAL) of Dresden Unit 3 Main Steam Line (MSL) Vibrations at Extended Power Uprate (EPU) Power Levels

Reason for Evaluation / Scope:

Background

During the forced outage on Quad Cities Unit 1 in November 2003, the 3B Electromatic Relief Valve (ERV) actuator was discovered with significant damage. ERV actuators were refurbished, reinforcing plates (i.e., tieback modifications) were installed, and vibration instrumentation was added to the ERVs and other MSL components. Quad Cities EC EVAL 346515, "Evaluation of Quad Cities Unit 1 Main Steam Line Vibrations at Extended Power Uprate (EPU) Power Levels," concluded that components were vulnerable to accelerated aging due to the increased vibration levels from EPU. Therefore, it was necessary to determine the extent of condition of the observed issues. The instrumentation of Dresden Unit 3 components and this evaluation are part of the extent of condition review for the Quad Cities issue.

The components evaluated are listed in the table provided in the section titled, "Summary: Long-Term Component Responses to EPU Vibration Levels," on page 6. All components were found acceptable for full-cycle operation. Recommendations for future inspections are made to ensure continued acceptable component performance. The total scope of the walkdown/ inspections performed is contained in EC EVAL 346402.

Purpose

This EC EVAL provides a technical evaluation supporting continuous operation of Dresden Unit 3 up to EPU power levels. This evaluation confirms the conclusions reached by the evaluation of walkdown results, as documented in EC 346402, "D3M10 Dryer Repair, Main Steam Walkdown Results and Evaluation for Return to Extended Power Uprate (EPU) Power."

Approach

Vibration data throughout the range of power operation, component failure history, inspection information and analytical modeling were used to assess component and system responses for full EPU power operation. Vibration data were obtained during the ramp up to 912 MWe (2851 MWth) on December 29, 2003. Additional data was taken on January 23, 2004 to replace erroneous data for MSL A. This data was extracted and then utilized in evaluations. The purpose of these evaluations was to provide assurance that full EPU power operation will pose no threat to continued equipment operation throughout a complete 24-month fuel cycle.

Results

Required actions are included to ensure acceptable equipment performance beyond the current cycle, which culminates in the Dresden Unit 3 refueling outage scheduled for November 2004 (i.e., D3R18). Additional actions to be completed during the subsequent shutdown for D3R18 are also provided. Actions are tracked under AR 203507

Engineering Change 347006, Revision 1
Evaluation (EVAL) of Dresden Unit 3 Main Steam Line (MSL) Vibrations at Extended Power Uprate (EPU) Power Levels

Required Actions:

Prior to D3R18

1. Obtain a full set of vibration data when/if the unit is moved to full thermal power of 2957 MWth for the first time. This data will be used to confirm the assumptions made on expected maximum vibration levels. (Assignee: A8326CMO, due date September; 2004, Analysis action assignee: A8064MW-DR, due date November, 2004)
2. Evaluate the current Preventive Maintenance (PM) scope and frequency for all components included in this evaluation. The expectation is that some MSL components will need increased PM frequencies for inspections, rebuilds, refurbishments and/or replacements until sufficient EPU operational experience proves that accelerated aging is not occurring. (Assignee: A8330NESTP, due date May 21, 2004)
    a. Consideration should be given to performing a post-EPU baseline inspection during the D3R18 outage.
3. Evaluate the snubber inspection plan. Ensure that sufficient MSL snubbers are functionally tested during D3R18 such that degradation is not occurring. Adjust inspection plan as necessary. At Quad Cities Unit 2, this review resulted in the addition of two MSL snubbers to the inspection scope. (Assignee: A8351NESPR, due date June 1, 2004)
4. Compare results of Quad Cities NAMCO switch and limitorque testing to Dresden vibration levels to ensure that testing bounds Dresden performance. (Assignee: A8064MW-DR, due date July 15, 2004)
5. Monitor the thermocouples on the Target Rock Safety/Relief Valve (S/RV) for potential pilot valve leakage throughout the cycle and confirm that locations meet General Electric (GE) Service Information Letter (SIL) 196, Thermocouple Location for Safety/Relief Valve Discharge Lines," Supplement 11. (Monitoring done as part of normal operator panel walkdowns, confirmation assigned to A8330NESTP (DRN92), due date April 15, 2004)
6. Monitor MSL flows until one main steam isolation valve (MSIV) is inspected to ensure early warning capability for detecting degradation. (Assignee: A8351NESPR, due date November 2, 2004)
7. Create activity to repeat walkdown as documented under EC EVAL 346402 each outage until performance is validated as acceptable. (assignee: A8330NESSC, due date 11/5/04)

During D3R18 Outage

8. Inspect the components internal to the ERV actuators. (Assignee: A8351NESPR, due date December 15, 2004)
9. Inspect one MSIV during the next refuel outage (D3R18 to confirm that no degradation has occurred, especially in the area of disk to stem. If degradation is found, evaluate the need for expanding inspection to other MSIVs. (D3R18 – Assignee: A8351NESPR, due date December 15, 2004)

Detailed Evaluation:

Component Damage Summary (Dresden Unit 3)

EC 346402 contains a detailed scope description and results of the walkdowns conducted during D3M10. There was no EPU vibration related damage identified during these walkdowns. Condition of the ERVs was found to be satisfactory with the exception of minor installation problems such as binding on 3E ERV upper guide bracket (**CR 189474/WR 123341**), which were corrected.

Effects of Increase to Full EPU Power 2957 Megawatts thermal (MWth)

All of the component evaluations utilized data obtained during the December 29, 2003, ramp up to 912 MWe and the steady state data from January 23,2004. These values represent the plant response to 2851 (December 29, 2003) and 2860 (January 23, 2004) MWth. In the future, power levels may reach full licensed power of 2957 MWth. Each component evaluation includes a discussion of any potential effects of increased vibration levels as the unit approaches full thermal power. Actual vibration levels will be measured if/when the unit achieves full thermal power to confirm the assumptions made.

Acceptability of ERV Component Operation at EPU Levels

The four ERVs have virtually identical assemblies consisting of the main ERV valve body, pilot valve, and solenoid actuator. The pilot valve is connected to the ERV by means of a turnbuckle (a threaded pipe coupling arrangement) and a pilot valve tube. Each valve has small diameter leak off piping that is routed back to the ERV discharge line. The 3B ERV was chosen for modeling due to its similarity in spatial orientation to the damaged Quad Cities 3B ERV. The key differences between the Dresden and Quad Cities configurations are the actuator rotation (90 degrees) and the discharge piping configuration. A detailed finite element analyses, using the finite element program ANSYS, was completed for a representative ERV (i.e., 3B), and is documented in Reference 2. In addition, significant testing of an ERV assembly has been completed at Wyle Laboratories. Although the focus of this testing was to validate the modifications for Quad Cities, they are directly applicable to Dresden. Documentation of the testing and results is included in the documents related to EC (DCP) 343933,"Replace the Current PORVs with ERVs," calculation SIR-04-023.

The finite element analyses determined natural frequencies and their corresponding mode shapes. By comparing the natural frequencies and mode shapes to the frequency content of the measured vibration data, it was determined that there was no concern for accelerated wear of the actuator internals as seen at Quad Cities. The 157 Hz problematic pendulum mode found at Quad Cities only existed at a much lower frequency for Dresden. The subsequent testing performed at Wyle Laboratories for Quad Cities

demonstrated that the mode of concern is an independent structural mode of the plunger assembly. Further reviews are in progress to determine differences between Dresden and Quad Cities and are tracked under AR 194877

Correlation of Bump Test Data to Analytical Models

Bump test and dynamic data for the 3B ERV were obtained during plant start-up on December 29, 2003, from 0% to 100% power (2851 MWth). The 3B ERV was subjected to impact tests while recording the accelerometers on the pilot valve. The magnitude of the data was not controlled, so the spectra plots were assessed for frequency content only. The model frequencies compare well to the bump test data. Thus, the analytical model is determined to be representative of the installed ERVs and can be used to evaluate the vibration loads and response.

Evaluation of Vibration Test Data

The vibration data was reviewed for amplitude versus power level to identify key frequencies and to observe amplitude versus power level trends. The overall Dresden acceleration magnitudes were approximately 10–20 times lower than for similar locations at Quad Cities Unit 1. The maximum RMS acceleration was 0.2 and 0.17 grms for TC16 and TC32, separate recorder designations, respectively. The tables in Attachment 1 show the RMS acceleration values for the Dresden ERVs and MSIVs. Typically, the Dresden accelerations increased proportionately to operational power level.

There were no strong frequencies observed consistently in the data. Multiple discrete frequencies were evidenced in the spectra and at several power levels, but the magnitudes were less than 0.068 grms at any discrete frequency. The maximum discrete vibration magnitudes at the 3B ERV inlet flange were 0.018, 0.026 and 0.025 grms in the X, Y, and Z axes, respectively (see Attachment 4).

Configuration Assessment – Dresden 3B ERV and Quad Cities 3B ERV

The Dresden 3B ERV has a different geometry when compared to some Quad Cities assemblies. The Quad Cities actuator is oriented radially outward from the ERV, whereas the Dresden actuator is mounted tangentially to the ERV for most assemblies. The effect of this design difference substantially affects the modal response. However, the completed testing validated that this difference is not related to the degradation modes observed for the bushings and guide rods. Therefore, no further evaluation of these configuration differences was deemed necessary.

ERV Modal Assessment – Dresden 3B ERV Compared to Quad Cities 3B ERV

All modal data was reviewed for frequency and mode shape dynamic behavior. The intent was to identify frequencies similar to the Quad Cities Unit 1 ERV

model pendulum mode shapes on the pilot valve at 139 and 157 Hz. Although some of the natural frequencies were the same between Dresden Unit 3 and Quad Cities Unit 1, the mode shapes are different. This is due to the orientation of the actuator. The pendulum mode shapes for the Dresden 3B ERV pilot do not occur at or near 157 Hz.

The Dresden pilot valve pendulum modes and active actuator modes occur at frequencies between 25 and 80 Hz. Since the measured dynamic amplitudes are very low at any discrete frequency within this range, the dynamic response of the actuator would also be low. Therefore, there is no expectation of any increase in the wear rate of the internal actuator components at Dresden.

Summary – Operation at Full EPU Thermal Power – 2957 MWth

A projection of the data was performed based on the maximum slope of channel 9 (on TC16). With this assumption, the maximum level expected at full thermal power is 0.25 grms. Since this magnitude is low, there is no expectation of any future problems at full EPU power

Evaluation of Piping

Piping models were generated and evaluated for Quad Cities Unit 1 and are documented in EC 346515. These models include main steam (MS) piping from the reactor vessel nozzle to the drywell penetration and include the relief valve discharge piping to the drywell penetration. Leak off piping for the ERVs was also assessed. The maximum calculated stress for the ERV drain piping was 3214 pounds per square inch (psi) versus an allowable stress of 3846 psi, per the criteria of OM-3, "Requirements for Preoperational and Initial Start-up Vibration Testing of Nuclear Power Plant Piping Systems." Since the magnitude of the measured dynamic data is lower for Dresden than for Quad Cities by a factor of 10 – 20 times, the resultant stresses would also be expected to be lower by that amount. That will result in a maximum pipe stress of less than 321.4 psi for Dresden, which is well below the OM-3 allowable (Attachment 4). Therefore, operation at EPU power levels will not impact piping structural integrity.

Evaluation of MSIVs

The evaluation of the MSIVs was performed by comparing measured values to seismic aging qualification test data for the Quad Cities Unit 1 actuators (see Attachment 3). The seismic aging testing consisted of sinusoidal variation of frequency between 25 and 200 Hz at 0.75 g applied to the valve bonnet in accordance with IEEE-344, "Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," for vibration endurance testing. Measured values for the B and D inboard MSIVs, and extrapolated values for the A and C MSIVs, were well bounded by this aging evaluation and were, therefore, determined to be acceptable for continuous full EPU power operation (Attachment 3).

The wear aging criteria does not imply that the components are qualified for the full five-year recommended maintenance life of the components with that steady state vibration level. In addition, degradation of various internal components may not be identified through external inspection. Therefore, preventive actions to perform the following two monitoring/inspection activities are recommended.

1. Prior to D3R18, monitor MSL flows for any change that could be indicative of degradation of an MSIV.
2. During D3R18, inspect one inboard MSIV for any signs of wear-related degradation in the area of the stem including pneumatic control components, limit switches, electrical leads, and fasteners.

Evaluation of Target Rock S/RVs

The Target Rock S/RV is evaluated in Attachment 3, with the result that there is low risk that future acceptable performance will be impacted. The most likely result of any degradation would be a leaking pilot valve. The downstream thermocouples should be closely monitored during the fuel cycle to ensure that degradation (i.e., evidence of leakage) has not occurred. The location of the thermocouples should be confirmed to comply with GE SIL 196, Supplement 11

Summary: Long Term Component Responses to EPU Vibration Levels

An assessment of the component response to EPU power levels has been performed and is summarized in the following table:

Component	2910 MWt Assessment Results	Expected Long-Term EPU Performance	Comments
MSIV Actuators	Sinusoidal aging test at 0.75 g from 25 to 200 Hz -	Acceptable at EPU levels based on existing margin. Recommendation includes Internal inspection during D3R18 to inspect for stem to disk degradation	Attachment 3
Inboard and Outboard MSIV Limit Switches	Seismically rugged 21.06 g, 29 Minute test at 7.5 g – Acceptable margin.	Acceptable at EPU levels based on existing margin,	Attachment 2
Inboard and Outboard MSIV Temperature Elements	Passive Item, very rigid subjected to small g forces acceptable	Acceptable	Attachment 2
MSIV LLRT Taps	Very low frequency based on analysis – no significant excitation measured at low frequency Acceptable for EPU levels	Acceptable	Attachment 2
Safety Valve Acoustic Monitors	Sine beat test accelerations 6.0 g compared to 4.226 g	Acceptable at EPU levels based on existing margin	Attachment 2

Component	2910 MWt Assessment Results	Expected Long-Term EPU Performance	Comments
Safety Valve Temperature Elements	Pipe clamp style thermocouple, passive element without any extended masses	Acceptable	Attachment 2
Pressure Switches	Seismic qualification tested at 20 g subjected to less than 0.2 g – Acceptable based on margin	Acceptable	Attachment 2
ERVs, Actuators, Drain Lines, etc	Analyzed using 2851 MWth vibration data and Finite Element Model (FEM)	All components acceptable for EPU levels based on margins and operating experience	Attachment 4
Safety Valves	Seismic qualification 11 g horizontal and 9 g vertical compared to ERV measured values of 1.52 g horizontal and 4.17 g vertical – Acceptable based on margin	Acceptable – Based on existing margin and no indication of degradation after 1 year at EPU levels	Attachment 4
Target Rock S/RV (3A)	Compared to qualification information on new type two-stage valves	There is some potential for pilot leakage and thread wear on the main piston/stem joint. Pilot valve leakage would be detectable by increased tailpipe temperatures	Attachment 3
MSL Drain Valves 1-0220-1 and -2, Operators, and Limit Switches	Estimated life duration based on the seismic testing information and extrapolated measured vibration levels is 38 years	Acceptable for full power operation. Valves are closed after turbine warm-up and left closed during normal operation, hence position change is only required during a unit shutdown	Attachment 2

Conclusions / Findings:

This EC EVAL provides an engineering evaluation supporting operation up to 2957 MWth. It has been determined that this full EPU power operation will not result in imminent failure or unacceptable degradation levels of any components. The conclusion is provided by evaluation of the empirical data, including operational data, measured vibration data, endurance test reports, and ERV testing during the outage to determine natural frequencies.

To confirm the conclusion, an action item to measure the actual vibration at full EPU thermal power will be performed. The acceptance level is less than 0.25 grms. Additionally, two action items have been created to monitor the MS flow rate and to inspect MSIV internals during the next refueling outage. These actions should confirm that the wear aging of the components is acceptable and the recommended five-year inspection interval is acceptable.

Engineering Change 347006, Revision 1
Evaluation (EVAL) of Dresden Unit 3 Main Steam Line (MSL) Vibrations at Extended Power Uprate (EPU) Power Levels

Attachments:

Attachment 1: Summary of Current MSL Vibration Levels – Dresden Unit 3 – for complete data information, see DRES-07Q-302 (Calculation)

Attachment 2: Evaluation of Components for Vibration Effects at 2851 MW

Attachment 3: "Evaluation of Dresden Unit 3 MSIV and SRV Vibration," General Electric Report No. DRF-0000-0025-0645, dated February 5, 2004

Attachment 4: Structural Integrity Associates Vibration Assessment

References:

1. DRES-07Q-302, Revision 1, Dresden Unit 3 Main Steam Line Vibration Data Reduction.
2. DRES-07Q-301, Revision 0, ERV Finite element Analysis and Vibration Evaluation.
3. DRE04-0002, Revision 0. Evaluation of Components for Vibration Effects.

EC# 347006
ATTACHMENT 1
Vibration Data

Summary of Dresden Unit 3 Vibration Data for TC16PWR Files Taken December 29, 2003

5 to 200 Hz Bandwidth

		80% Power Level			86% Power Level			90% Power Level		
		2317 MWth			2470 MWth			2538 MWth		
Channel	Location	Grms	Gmax	Gmin	Grms	Gmax	Gmin	Grms	Gmax	Gmin
1	3-0203-3D NS	0.0783	0.3442	-0.3264	0.0748	0.3367	-0.3375	0.0755	0.3599	-0.3569
2	3-0203-3D Vertical	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
3	3-0203-3D EW	0.1483	0.6882	-0.6682	0.1503	0.7286	-0.6411	0.1263	0.5754	-0.5585
4	B MS RV Header NS	0.0469	0.2113	-0.2292	0.0448	0.2124	-0.1940	0.0448	0.1951	-0.2179
5	B MS RV Header Vertical	0.0548	0.2455	-0.2701	0.0564	0.2572	-0.2734	0.0589	0.2601	-0.2815
6	B MS RV Header EW	0.0966	014341	-0.3861	0.0920	0.4035	-0.4052	0.0878	0.3886	-0.4233
9	D MS Riser to RPV NS	0.0813	0.3402	-0.3585	0.0821	0.3515	-0.3649	0.0835	0.3565	-0.3940
10	D MS Riser to RPV Vertical	0.0993	0.3660	-0.3511	0.0792	0.3305	-0.3534	0.0775	0.3357	-0.3623
11	D MS Riser to RPV EW	0.1112	0.6638	-0.5088	0.1385	0.616	-0.6326	0.1548	0.6746	-0.6999
12	B MS Riser to RPV NS	0.0600	0.2612	-0.2695	0.0679	0.3072	-0.3112	0.0717	0.3738	-0.3225
13	B MS Riser to RPV Vertical	0.0559	0.2574	-0.2737	0.0600	0.2774	-0.2822	0.0630	0.2764	-0.2885
14	B MS Riser to RPV EW	0.0955	0.4789	-0.4477	0.1095	0.5618	-0.5153	0.1199	0.6370	-0.5617

		95% Power Level			100% Power Level		
		2694 MWth			2851 MWth		
Channel	Location	Grms	Gmax	Gmin	Grms	Gmax	Gmin
1	3-0203-3D NS	0.0738	0.4206	-0.4401	0.0893	0.4170	-0.4703
2	3-0203-3D Vertical	N/A	Na	N/A	N/A	N/A	N/A
3	3-0203-3D EW	0.0859	0.4955	-0.4809	0.0891	0.4630	-0.4443
4	B MS RV Header NS	0.0491	0.2346	-0.2092	0.0543	0.2579	-0.2668
5	B MS RV Header Vertical	0.0640	0.3137	-0.2941	0.0703	0.3423	-0.3566
6	B MS RV Header EW	0.0903	0.4545	-0.4469	0.0979	0.4297	-0.4513
9	D MS Riser to RPV NS	0.1014	0.4495	-0.4598	0.0986	0.4723	-0.4705
10	D MS Riser to RPV Vertical	0.0850	0.3678	-0.3932	0.0720	0.3639	-0.3810
11	D MS Riser to RPV EW	0.1876	0.9299	-0.8391	0.2005	0.9725	-0.8844
12	B MS Riser to RPV NS	0.0879	0.4173	-0.4194	0.0959	0.4475	-0.4537
13	B MS Riser to RPV Vertical	0.1305	0.4357	-0.4706	0.0771	0.3568	-0.3663
14	B MS Riser to RPV EW	0.1491	0.6816	-0.6499	0.1492	0.6731	-0.7106

EC# 347006
ATTACHMENT 1
Vibration Data

Summary of Dresden Unit 3 Vibration Data for TC32PWR Files Taken December 29, 2003

5 to 200 Hz Bandwidth

		80% Power Level			86% Power Level			90% Power Level		
		2317 MWth			2460 MWth			2538 MWth		
Channel	Location	Grms	Gmax	Gmin	Grms	Gmax	Gmin	Grms	Gmax	Gmin
1	3-203-1B IB MSIV NS	0.046	0.2066	-0.2178	0.0535	0.2336	-0.2456	0.0603	0.2586	-0.2953
2	3-203-1B IB MSIV Vertical	0.1384	0.1632	-0.1709	0.0423	0.2081	-0.1993	0.0468	0.2074	-0.2076
3	3-203-1B IB MSIV EW	0.1588	0.0275	-0.2749	0.0619	0.2972	-0.2964	0.0682	0.2928	-0.3132
4	3-203-3B Pilot Valve Inlet (1) NS	0.1671	0.3106	-0.3283	0.07	0.2932	-0.3089	0.0661	0.2917	-0.2987
5	3-203-3B Pilot Valve Inlet (1) Vertical	0.1044	0.5477	-0.4973	0.1061	0.5082	-0.5571	0.0967	0.4683	-0.4408
6	3-203-3B Pilot Valve Inlet (1) EW	0.0784	0.3596	-0.3599	0.0838	0.3817	-0.3897	0.0883	0.3884	-0.4431
9	3-203-1D IB MSIV NS	0.0613	0.2654	-0.2791	0.0656	0.3141	-0.3089	0.0746	0.3696	-0.3328
10	3-203-1D IB MSIV Vertical	0.0427	0.2083	-0.2489	0.0463	0.2963	-0.2859	0.0505	0.3512	-0.3236
11	3-203-1D IB MSIV EW	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
12	3-203-3D Pilot Valve Inlet (1) NS	0.0433	0.2214	-0.22	0.0465	0.2118	-0.2119	0.0495	0.2249	-0.2149
13	3-203-3D Pilot Valve Inlet (1) Vertical	0.133	0.6395	-0.6345	0.1106	O.4886	-0.5016	0.1048	0.4808	-0.4501
14	3-203-3D Pilot Valve Inlet (1) EW	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
17	3-203-3A Target Rock NS	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
18	3-203-3A Target Rock Vertical	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
19	3-203-3A Target Rock EW	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
20	3-203-3B Inlet flange NS	0.06O6	0.252	-0.2625	0.0613	0.276	-0.2836	0.0601	0.3084	-0.304
21	3-203-3B Inlet flange Vertical	0.091	0.375	-0.3833	0.0989	0.621	-0.587	O.1085	0.6296	-0.6162
22	3-203-3B Inlet flange EW	0.0488	0.2106	-0.2298	0.053	0.2745	-0.2846	0.0613	0.373	-0.3751
25	3-203-3C Inlet flange NS	0.0712	0.2799	-0.2747	0.0893	0.3672	-0.355	0.0669	0.309	-0.2906
26	3-203-3C Inlet flange Vertical	0.1707	0.8791	-0.8212	0.1721	0.88	-0.9027	0.156	0.7384	-0.7401
27	3-203-3C Inlet flange EW	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
28	3-203-3E Inlet flange NS	0.0641	0.2813	-0.3046	0.0717	0.334	-0.3667	0.0735	0.3308	-0.3739
29	3-203-3E Inlet flange Vertical	0.067	0.2994	-0.2987	0.0702	0.3522	-0.2988	0.0685	0.3396	-0.3313
30	3-203-3E Inlet flange EW	0.0479I	0.2107	-0.2074	0.053I	.025	-0.2286	0.0577I	0.2492	-0.2432

Note (1): Accelerometers are attached to the pilot valve turnbuckle

EC# 347006
ATTACHMENT 1
Vibration Data

Summary of Dresden Unit 3 vibration Data for TC32PWR files Taken on December 29, 2003 (continued)

5 to 200 Hz Bandwidth

		95% Power Level			100% Power Level		
		2694 MWth			2851 MWth		
Channel	Location	Grms	Gmax	Gmin	Grms	Gmax	Gmin
1	3-203-1B IB MSIV NS	0.0713	0.3338	-0.3523	0.0825	0.4261	-0.5022
2	3-203-1B IB MSIV Vertical	0.0531	0.3865	-0.3708	0.0596	0.4801	-0.7379
3	3-203-1B IB MSIV EW	0.0803	0.408	-0.3862	0.0921	0.6858	-0.7303
4	3-203-3B Pilot Valve Inlet (1) NS	0.0738	0.3155	-0.3487	0.0741	0.3572	-0.3195
5	3-203-3B Pilot Valve Inlet (1) Vertical	0.0985	0.4889	-0.4517	0.101	0.4903	-0.4844
6	3-203-3B Pilot Valve Inlet (1) EW	0.0946	0.4095	-0.4175	0.1028	0.4614	-0.5388
9	3-203-1D IB MSIV NS	0.0897	0.4417	-0.4158	0.1042	0.5256	-0.4774
10	3-203-1D IB MSIV Vertical	0.0579	0.2908	-0.3016	0.0661	0.4604	-0.4999
11	3-203-1D IB MSIV EW	N/A	N/A	N/A	N/A	N/A	N/A
12	3-203-3D Pilot Valve Inlet (1) NS	0.0551	0.2617	-0.2558	0.0599	0.2815	-0.284
13	3-203-3D Pilot Valve Inlet (1) Vertical	0.1123	0.5107	-0.5358	0.1204	0.5208	-0.5206
14	3-203-3D Pilot Valve Inlet (1) EW	N/A	N/A	N/A	N/A	N/A	N/A
17	3-203-3A Target Rock NS	N/A	N/A	N/A	N/A	N/A	N/A
18	3-203-3A Target Rock Vertical	N/A	N/A	N/A	N/A	N/A	N/A
19	3-203-3A Target Rock EW	N/A	N/A	N/A	N/A	N/A	N/A
20	3-203-3B Inlet flange NS	0.0651	0.2991	-0.2963	0.0723	0.3409	-0.3774
21	3-203-3B Inlet flange Vertical	0.1204	0.6371	-0.6804	0.1335	0.6622	-0.65
22	3-203-3B Inlet flange EW	0.0664	0.3842	-0.384	0.078	0.4503	-0.3969
25	3-203-3C Inlet flange NS	0.0706	0.3347	-0.3663	0.0738	0.3319	-0.3547
26	3-203-3C Inlet flange Vertical	0.1639	0.904	-0.8969	0.1698	0.8773	-0.8388
27	3-203-3C Inlet flange EW	N/A	N/A	N/A	N/A	N/A	N/A
28	3-203-3E Inlet flange NS	0.08	0.3705	-0.3916	0.0876	0.4219	-0.4068
29	3-203-3E Inlet flange Vertical	0.0708	0.3298	-0.3176	0.0777	0.3515	-0.3799
30	3-203-3E Inlet flange EW	0.0642	0.2654	-0.3179	0.0698	0.2991	-0.3427

Note (1): Accelerometers are attached to the Pilot valve turnbuckle

EC# 347006
ATTACHMENT 1
Vibration Data

Summary of Dresden Unit 3 Vibration Data for Data Taken on January 24, 2004

5 to 200 Hz Bandwidth

	100% Power Level (09:29:07) 2860 MWth			100% Power Level (09:35:18) 2860 MWth		
Location	Grms	Gmax	Gmin	Grms	Gmax	Gmin
3-0203-3D Vertical	0.027	0.0788	-0.0802	0.0264	0.0786	-0.0797
3-0203-3A Target Rock NS	0.0856	0.3979	-0.371	0.0834	0.3677	-0.3687
3-0203-3A Target Rock Vertical	0.2206	0.8176	0.49381	0.2214	0.8502	-0.8686
3-0203-3A Target Rock EW	0.0844	0:3855	-0.4635	0.0845	0.3675	-0.4203
3-0203-3D Pilot Valve Inlet (1) EW	6.55E-04	0.0032	-0.0051	0.0007	0.0035	-0.0053

Note (1): Accelerometers are attached to the Pilot valve turnbuckle

EC# 347006
ATTACHMENT 2
MSL Component Evaluation Summary

Summary of Evaluation (Calc. No. DRE04-0002)

Component	Manufacturer/ Model	Vibration Data Points	Evaluation
Inboard and Outboard MSIV Limit Switches	NAMCO/EA740	32 Channel Tape Deck; Channels 1, 2, 3, 7, and 8	a) SRSS of peak accelerations measured = 1.488 g < SRSS of peak accelerations from seismic qualification test (Ref. 1) = 21.06 g. b) Based on a test duration of 30.5 minutes and input acceleration varying from 7.34 g to 15.07 g (Ref. 1), the equivalent duration, based on a fatigue exponent of 4.3 and using the SRSS of the measured RMS accelerations (0.168 g), is 13.86 years. The limit switches are mounted on a bracket; therefore, an amplification factor of four in the x-direction was applied to the RMS acceleration based on computed frequencies of the bracket. c) Using IEEE-344 as guidance, 10 cycles of strong motion effect per 15-second duration is considered. Using a fatigue exponent of 4.3, based on 30.5 minutes of test at input acceleration varying from 7.34 g to 15.07 g (Ref. 1), will yield an equivalent number of cycles of 4.4×10^9 for the SRSS of the measured RMS accelerations of 0.168 g. The limit switches are mounted on a bracket; therefore, an amplification factor of four in the x-direction was applied to the RMS acceleration based on computed frequencies of the bracket. For typical steels, this number of cycles is in the flat region of the S-N curve. Comparing the measured accelerations with the qualified accelerations provides reasonable assurance that at such a low level of acceleration, the corresponding alternating stress intensities will be below the fatigue endurance limit.

EC# 347006
ATTACHMENT 2
MSL Component Evaluation Summary

Component	Manufacturer/ Model	Vibration Data Points	Evaluation
Inboard and Outboard MSIV Temperature Elements	Omega/BT-000-T-7.125-72-1	32 Channel Tape Deck; Channels 1, 2, 3, 7, and 8	This thermocouple is a passive element consisting of a 7.125" long X 3/16" diameter stainless steel tube containing two thermocouple wires. It is attached with a spring-loaded quick connect bayonet-style cap. The lead wires are protected with glass braid insulation and stainless steel cover (20 gauge stranded). The extended mass of the thermocouple is very small (less than 1 ounce) and the cantilevered length is only 7.125". Considering the low level of accelerations (SRSS RMS acceleration = 0.168 g) the alternating stress intensities in the thermocouple components will be well below their fatigue endurance limit.
Safety Valve Acoustic Monitors	NDT International, Inc./NDT-838-1	32 Channel Tape Deck Channels 13, 14, 15 (Channels 2, 3, 4 on page 17 of Ref. 5)	a) SRSS of peak accelerations measured = 1.478 g < sine beat test accelerations from seismic qualification test (Ref. 2) = 6.0 g. b) The SRSS of the measured RMS accelerations is 0.275 g. Comparing this acceleration value with the qualified acceleration provides reasonable assurance that at such a low level of acceleration the corresponding alternating stress intensities will be below the fatigue endurance limit.

EC# 347006
ATTACHMENT 2
MSL Component Evaluation Summary

Component	Manufacturer/ Model	Vibration Data Points	Evaluation
Safety Valve Temperature Elements	Weed/N9343-T-2-.375-U	32 Channel Tape Deck; Channels 13, 14, and 15 (Channels 2, 3, and 4 on page 17 of Ref. 5)	a) SRSS of peak accelerations measured = 1.478 g < min. SRSS peak accelerations from SRV aging test (Ref. 4) = 3.76 g. b) Based on a test duration of 120 minutes and input acceleration varying from 3.76 g to 8.86 g (Ref. 6), the equivalent duration, based on a fatigue exponent of 4.3 and using the SRSS of the measured RMS accelerations (0.275 g), is 135.3 years. c) Based on 10 cycles per second and using a fatigue exponent of 4.3, the 120 minutes of test at an input acceleration varying from 3.76 g to 8.86 g (Ref. 3) will yield an equivalent number of cycles of 4.27×10^{10} for the SRSS of the measured RMS accelerations of 0.275 g. For typical steels, this number of cycles is in the flat region of the S-N curve. Comparing the measured accelerations with the qualified accelerations provides reasonable assurance that at such a low level of acceleration, the corresponding alternating stress intensities will be below the fatigue endurance limit.

Component	Manufacturer/ Model	Vibration Data Points	Evaluation
MS Drain Valves Operator	Limitorque/ SMB-000	32 Channel Tape Deck; Channels 1, 2, 3, 7, and 8 Note: Channels 1, 2, 3, 7 and 8 are on MSIVs. These values are increased by a factor of 3 for margin	d) SRSS of peak accelerations measured = 1.488 g < SRSS peak accelerations from dynamic qualification test (Ref. 3) = 37 g. e) Based on test duration of 351 minutes and input acceleration varying from 3.29 g to 37g (Ref. 3), the equivalent duration, based on a fatigue exponent of 4.3 and using the SRSS of the measured RMS accelerations (0.504 g), is 180 years. f) Using IEEE-344 as guidance, 10 cycles of strong motion effect per 15-second duration is considered. Using a fatigue exponent of 4.3, based on 351 minutes of test at and input acceleration varying from 3.29 g to 37 g (Ref. 3) will yield an equivalent number of cycles of 5.68×10^{10} for the SRSS of the measured RMS accelerations of 0.504 g. For typical steels, this number of cycles is in the flat region of the S-N curve. Comparing the measured accelerations with the qualified accelerations provides reasonable assurance that at such a low level of acceleration, the corresponding alternating stress intensities will be below the fatigue endurance limit.

Component	Manufacturer/ Model	Vibration Data Points	Evaluation
MS Drain Valves Operator Limit Switch	Limitorque/ SMB-000	32 Channel Tape Deck; Channels 1, 2, 3, 7, and 8 Note: Channels 1, 2, 3, 7, and 8 are on MSIVs. These values are increased by a factor of 3 for margin	See evaluation for the operator above.

References

1. Wyle Report No. 47824-2
2. Wyle Report No. 45638-1 (EQ-01Q)
3. SDRC Report No 10959
4. NTS Report No. 548-8854-2
5. SIA Calc. No. DRES-07Q-302

EC# 347006
ATTACHMENT 3
Evaluation of MSIV and SRV

DRF 0000-0025-0645

February 5, 2004

To: Sharon Eldridge
From: PD Knecht

Subject: Evaluation of Dresden Unit 3 MSIV and SRV Vibration Data

References:

1. "Assessment Scope for Potential Vibration Induced Problems Quad Cities Unit 1", Exelon Test Plan, November 26, 2003
2. "Quad Cities Nuclear Power Station Units 1 and 2 Environmental Qualification Report - MSIV Actuator", NEDC-31886P, Class III, December 1990.
3. "Dresden Unit 3 Main Steam Line Component Vibration Assessment", Structural Integrity Associates, DRES-07Q-302r1, January16, 2004
4. "Quad Cities Unit 1 Assessment of High Frequency Acoustic Vibration on MSL Components", GE-NE 0000-0008-1763-02, December 2002.
5. "Target Rock Relief Valve failure to Fully Open", SIL 646, December 20, 2002
6. "Hope Creek Generating Station Unit 1 Environmental Qualification Report, Book No. S20, NEDC 30942, November 1985

Background

Dresden Unit 3 is currently licensed to operate at an extended power uprate value of 2957 MWt. Following two hood failures of the steam dryer assembly at Quad Cities Unit 2 and a hood failure and observed failure of the 3B ERV at Quad Cities Unit 1, operation of both Dresden Units at EPU power has been of concern due to the potential for flow induced vibration impacts on the main steam line (MSL) components.

Due to this concern, vibration data on various components external to the reactor pressure vessel at Dresden Unit 3 were taken during the power ascension to CLTP power on December 30, 2003. Data were evaluated at power levels between approximately 78.3% current licensed thermal power (CLTP) and 96.47% CLTP. Due to instrumentation problems, additional data was taken on the A SRV on January 23, 2004 at approximately 96.7% CLTP. The plant is currently operating at its maximum thermal power consistent with the generator capability.

GENE was asked in Reference 1 to evaluate the vibration data associated with the inboard MSIVs on the "B" and "D" MSLs and the Target Rock Safety Relief Valve and provide recommendations regarding long term operation at CLTP. This letter provides the GENE response to this request.

Methodology

Vibration data for MSIVs were compared against seismic aging qualification test data for the Quad Cities Unit 1 MSIV actuators (Reference 2). Although Dresden does not have these exact same actuator, engineering judgment was used to derive a conclusion regarding the similarity of the components. The seismic aging test (Reference 2) consisted of sinusoidal variation of frequency between 25 hz and 200 hz at 0.75g applied to the valve bonnet. No evidence of damage was observed during these qualification tests.

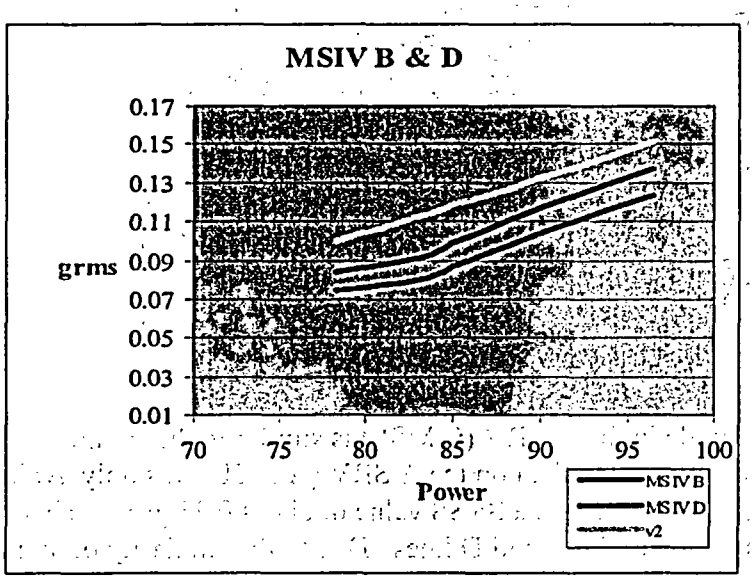
No similar data exists for the 3-stage Target Rock SRVs. However, it was assumed that seismic wear aging would have been successful for these valves at the standard IEEE 382-1980 test value of 0.75g. Furthermore, there are other BWRs that have operated at above original rated power that have not reported evidence of wear or stuck open valve events.

Evaluations

1. Main Steam Isolation Valves

B and D MSIVs

Square Root Sum of Squares (SRSS) values of the three monitored vibration axes corresponding to the total grms value from reference 3 are shown in the figure below. For reference the velocity squared relationship also is shown.



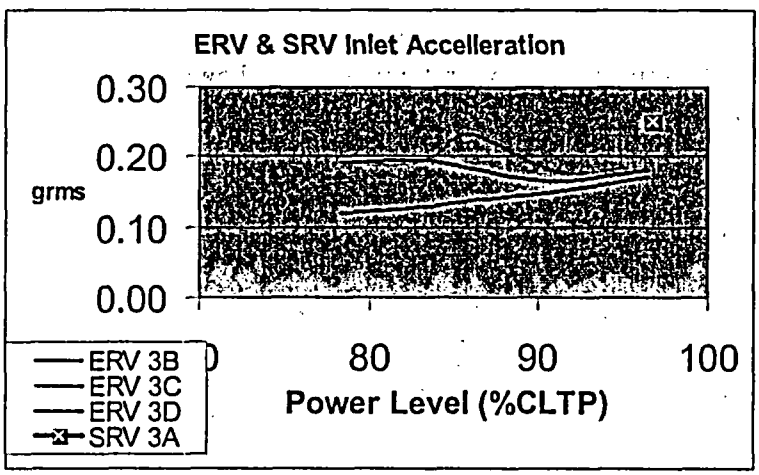
In all cases, the total amplitudes are relatively low in comparison with the qualification basis of 0.75g and the displacement velocities are below 0.1 in/sec. No damage is expected for these conditions.

Minor acoustic resonant peaks were observed at about 145 hz and 152 hz in the reduced data (Reference 3). However, the peak amplitudes for these frequencies were not significant and did not exceed 0.03 grms. The peak values decreased as power approached EPU power.

It should be noted that the MSIV actuators at Dresden are different than those at Quad Cities Unit 1 and no vibration aging qualification basis is documented. However, based on similarity of the designs and the relatively low amplitude of the vibrations, the MSIVs should be acceptable based on engineering judgment.

A and C MSIVs

Because no specific vibration data were obtained for the A or C MSIVs, the total SRSS vibration amplitudes at the ERV were reviewed as shown in the figure below to determine if those MSIVs would have a similar response as the B and D MSIVs.



As can be seen the amplitude in the C ERV (C MSL) is similar to the B and D lines especially at EPU power. Vibration data on the A SRV (A MSL) was only available at approximately 96.7% CLTP and shows a SRSS value of about 0.25 grms. Therefore it too shows a similar amplitude to the B and D lines. Due to the similarity of amplitudes

and geometry of the piping, the A and C MSIV likely will have lower amplitude than the ERVs or SRVs and be well within the 0.75g wear aging criterion and displacement velocity will be low.

Because the walkdowns also concluded that there was no significant increase in vibration related issues due to operation at EPU, it is concluded that all MSIVs will operate successfully at EPU power.

Discussion

It should be noted that the wear aging criteria does not imply that the components are qualified for the full five-year recommended maintenance life of the components with that steady state vibration level. If this amount of vibration occurs, there can be long term degradation of the valve internals. The higher steam flow rate can cause agitation of the internal parts causing accelerated fretting corrosion wear in comparison with pre-EPU conditions (refer to SIL 568). Accelerated wear of other subassemblies on the MSIVs such as pneumatic control components, limit switches, electrical leads and fasteners also may not be identified through external inspection. Because this failure mode cannot be detected through external inspection alone, an internal inspection should be made at the next refueling outage to assess the rate of wear.

In Reference 4, the impact of MSL vibrations was discussed. Because the main disk is free to move, high vibration could potentially result in degradation of the lower disk guide liner. If significant, this could slow or possibly prevent valve motion. Since Dresden 3 has completed approximately one year without evidence of degradation and vibration levels are well within the IEEE wear aging standard the risk of component failure is considered low to medium consistent with Reference 4.

Because no degradation of the MSIVs was shown during the walkdowns and to date operability has been demonstrated during the most recent outage, it is concluded that operation at CLTP is acceptable. However, it is recommended that individual MSL flows be monitored during the operating cycle to detect any variations between steam lines that could be due to MSIV degradation and an internal wear inspection should be conducted at the next refueling outage.

2. Target Rock Safety Relief Valves

Due to instrumentation problems during the power ascention on December 30, 2003, additional data was taken on the A SRV on January 23, 2004 at approximately 96.7% CLTP. The total vibration is about 0.25 grms. No significant evidence of acoustic

vibrations is shown. A peak amplitude at about 72 hz is indicated with a SRSS magnitude of 0.06 grms.

Discussion

The Target Rock valve at Dresden is a 3-stage design that has been replaced with a 2-stage design in later BWRs. No qualification data exist for the 3-stage SRV because they were qualified through the Seismic Qualification Users Group (SQUG) methodologies that did not address wear aging. Qualification aging of 2-stage valve has been successful with wear aging at the same 0.75 g acceleration in a similar manner to that which has been applied to the MSIV actuators. In comparison with the 2-stage Target Rock, the actuators on the 3-stage valve are lighter and it should display higher natural frequencies. Reference 6 identified natural frequencies of 128, 152 and 160 hz in a 2-stage Target Rock SRV.

Based on the similarity of design, the existing Dresden experience without observed degradation other than some loose hose fittings, and the vibration data taken at CLTP, it is concluded, based on engineering judgment, that there is only a low to medium risk of degradation occurring at CLTP and continuous operation is acceptable subject to monitoring for evidence of degradation.

As discussed in Reference 4 and 5, vibration could cause thread wear on the main piston/stem joint. In addition vibration could cause the pilot valve to begin to leak. Extended operation with leaking pilot valves has been shown to cause steam cutting and eventual inadvertent opening of the SRV. Because of these concerns, the downstream thermocouples on the SRVs should be closely monitored during the next operating cycle for evidence of leakage. The location of the thermocouples should be confirmed to comply with SIL 196 supplement 11.

EC# 347006

ATTACHMENT 4

Structural Integrity Associates Vibration Assessment

Structural Integrity Associates, Inc.

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January 21, 2004
SIR-04-007
KJO-04-001

Ms. Sharon Eldridge
Lead Project Engineer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

Subject: Dresden Unit 3 Main Steam Line and Component Vibration Assessment

Dear Sharon:

This letter report contains the Dresden Unit 3 main steam vibration assessment for the Electromatic Relief Valves (ERVs), main steam piping, and the safety valves, due to operation at Extended Power Uprate. The assessment is contained in Attachment 1.

If you have any questions, please call me.

Prepared by

Kevin J. O'Hara
Associate

Reviewed by

Karen K. Fujikawa, P.E.
Associate

Approved by

Karen K. Fujikawa, P.E.
Associate

kjo
Attachment: 1. Evaluation of Dresden Unit 3 Main Steam Line Vibration Assessment

cc: Karen K. Fujikawa (SI)
M. Qin (SI)
G. L. Stevens (SI)
DRES-07Q-401

Evaluation of Dresden Unit 3 Main Steam Line Vibration Assessment

Quad Cities Unit 1 main steam system has experienced significant vibration loads due to a combination of flow turbulence and acoustic resonance, especially at the Electromatic Relief Valve (ERV) branch piping. The vibration levels have increased as a direct result of the flow rate increase realized by Extended Power Uprate (EPU) operation. Additionally, this plant has experienced both hardware failures and discovered excessive component wear during its most recent outages (Q1F51). SI report SIR-04-001 documents this assessment [7].

This report provides a dynamic assessment of Dresden Unit 3 (U3). This is a sister facility to Quad Cities Unit 1 (QC1), in that both of these facilities are both BWR3s. The concern is whether the Dresden facility has similar dynamic issues associated with the main steam line (MSL) system. The intent is to review the Dresden main steam system dynamic behavior, including a detailed analysis of the 3B ERV and compare its dynamic behavior to Quad Cities.

Vibration measurements were taken at 36 accelerometer locations on the Dresden U3 main steam piping at various operating levels up to 2851 MWth. Most of the accelerometers were located on the four ERVs and two adjoining pilot valves, which were expected to be the locations of high vibration. The vibration time histories were converted to frequency domain spectra and grms (root-mean-square) values. The time histories were then filtered and maximum peak values determined. Reference [2] contains details of the vibration data reduction, and Table 1 summarizes the RMS and peak accelerations for the ERVs at 2851 MWth.

The vibration data obtained at EPU conditions (2851 MWth) was used to evaluate the ERVs (1-0203-3B, -3C, -3D, and -3E), the main steam piping inside primary containment and its associated branch lines, and the safety valves. The following summarizes the results of the vibration evaluation.

Evaluation of the ERV Assemblies

Dresden ERV 3B was assessed similar to Quad Cities ERV 3B for comparative purposes, since QC1 ERV 3B sustained significant wear at the last outage. These ERVs have virtually identical assemblies, which consists of the main ERV valve body, pilot valve, and solenoid actuator (Figure 1). The pilot valve is connected to the ERV by means of a turnbuckle and a pilot valve tube. Each valve has a small diameter drain pipe that is routed back to the ERV discharge line. The key differences are that Dresden ERV 3B actuator is rotated 90° (Figure 2) to the Quad Cities ERV 3B actuator (Figure 3) and the discharge pipe routing. A detailed finite element analysis, using the finite element program ANSYS [3] was performed for Dresden ERV 3B, Reference [1].

The finite element analyses determined natural frequencies and their corresponding mode shapes. The natural frequencies and mode shapes were compared to the frequency content of the measured vibration data. For QC1 ERV 3B, it was determined that a pilot valve pendulum mode is the primary cause of the excessive actuator wear at 157 Hz, whereas, for the Dresden Unit 3 ERV 3B, pilot valve pendulum mode occurred at a much lower frequency.

Correlation of Bump Test Data to Analytical Models

Bump test data for ERV 3B and dynamic data were obtained during power ascension on December 29th to EPU power (2851 MWth). ERV 3B was subjected to impact tests while

recording the accelerometers on the pilot valve. The magnitude of the data was not controlled, so the spectra plots were assessed for frequency content only. The bump test results were then compared to the modal analysis results to the Dresden ERV 3B data. A review of the bump test data shows the frequencies of the Dresden 3B models compares well to the bump test data. Thus, the analytical model is assessed to be representative of the installed ERVs and can be used to evaluate the valve dynamic behavior.

Evaluation of Vibration Test Data

The vibration data was reviewed for amplitude versus power level to identify key frequencies and to observe amplitude versus power level trends. Review of the time history data, indicated that the Dresden acceleration magnitudes (grms) were very low, approximately 10 to 20 times lower than QC1 data. The maximum RMS acceleration was 0.2 and 0.17 grms for TC16 and TC32, respectively. Table 1 shows the time history RMS acceleration values for the Dresden ERVs and MSIVs. Typically, the Dresden accelerations increased proportionate to operational power level.

There were no strong frequencies observed consistently in the data. Multiple discrete frequencies showed up in the spectra and at several power levels, but the vibration magnitudes were very low (less than 0.068 grms at <u>any</u> discrete frequency). The maximum discrete vibration magnitudes at the ERV 3B inlet flange were 0.018, 0.026 and 0.025 grms in the X, Y and Z axes, respectively.

Configuration Assessment – Dresden ERV 3B and Quad Cities ERV 3B

The Dresden ERV 3B has a key difference in the geometry, the Quad Cities actuator is oriented radially outward from the ERV (Figure 3), whereas, the Dresden actuator is mounted tangentially to the ERV (Figure 2). The effect of this design difference, substantially affects the modal response.

ERV Modal Assessment - ERV 3B and Quad Cities ERV 3B

All modal data was reviewed for frequency and mode shape dynamic behavior. The intent was to identify frequencies at 139 and 157 Hz (similar to QC1 3B ERV model) that had pilot valve pendulum modes shapes. A frequency comparison was performed between Dresden Unit 3 3B ERV and QC1 3B ERV. Although some of the natural frequencies were the same between the Dresden and QC1 ERVs, the mode shapes are quite different. This is due to the orientation of the Dresden 3B ERV actuator (oriented 90° out-of-phase with respect to QC1 3B ERV), the pilot pendulum mode shapes for Dresden 3B ERV <u>do not</u> occur at or near 157 Hz.

The Dresden pilot valve pendulum modes and active actuator modes occur at frequencies between 25 and 80 Hz. Since the dynamic amplitudes are very low at any discrete frequency within this range, then the dynamic response would also be very low.

Evaluation of Piping – Quad Cities and Dresden Comparison

<u>Quad Cities Unit 1</u>

Piping models were generated for Quad Cities MSLs A, B, and C [4]. These models include the main steam piping from the reactor vessel nozzle to the drywell penetration, and include the

relief valve discharge piping to the drywell penetration. In addition, piping models were generated for the ERV assembly and associated drain piping. The piping models were generated using the computer program, PIPESTRESS [5]. The drain piping is very similar for the four valves. Therefore, the valve assemblies and the connecting drain piping were evaluated by applying the highest measured acceleration on each ERV for the appropriate MSL.

The analysis was done using the response spectrum modal superposition method. The vibration loading was input as response spectra in the three orthogonal directions. The shape of the input spectra is obtained from the frequency domain acceleration curves generated from QC1 data. Peak acceleration values are used instead of rms, because even though the peak acceleration may occur in only a small percentage of the cycles, at 139 hz they can occur often enough to apply thousands of load cycles in a relatively short time. The analysis conservatively assumes that all vibration cycles are occurring at the peak levels. Use of peak rather than rms values conservatively assumes that all vibration cycles are at the peak value, whereas the majority of the cycles are at a much lower level. The input spectra was conservatively applied to the entire piping system.

Stresses calculated for the piping were compared to the endurance limit as specified by O&M Standard Part 3 [6]. The allowable stress is 3846 psi, based on the OM-3 criteria. The maximum calculated stress for the ERV drain piping was 3214 psi, which is within the OM-3 allowable. It should be noted that the OM-3 criteria contains a number of safety factors which, when taken together, produce a very conservative result.

Dresden, Unit 3

Since Dresden U3 dynamic data is a factor of 10-20 lower then the stresses in the Dresden piping would lower by the same factor ($\leq$ 321.4 psi versus Quad Cities at 3214 psi, models behave linearly for dynamic stresses). Thus, the Dresden dynamic stresses are well below the OM-3 allowable, so operation at EPU power levels will not affect the integrity of the piping.

Evaluation of the Safety Valves

The safety valve evaluations are based on the measured accelerations at the ERV inlet flanges. The maximum measured acceleration is 0.2 g rms in any direction. These vibration amplitudes are substantially lower than the Quad Cities data (1.52 horizontally and 4.17 vertically). The safety valves are Dresser relief valves. Similar valves at another nuclear power plant have seismic qualification levels of 11 g horizontal and 9 g vertical. Based on the number of cycles that the Dresden safety valves have already experienced at these vibration levels, plus the results of the walkdown which identified no visual wear, the valves at Dresden U3 are considered to have operability margin.

Conclusions and Recommendations

Based on the results of the various analyses, it is acceptable to operate at 2851 MWth for any period of duration. The dynamic data for the MSL system, including the Dresden 3B ERV are low. The ERV 3B analysis indicates that the ERV mode shapes would not respond to these low acceleration magnitudes. Additionally, no acoustic modes are observed in the MSL, therefore, the Dresden Unit 3 MSL and associated valve and branch line have adequate margin to operate at 2851 MWth indefinitely.

References

1. Structural Integrity Associates Calculation No. DRES-07Q-301, Revision 0, "3B ERV Finite Element Analysis and Vibration Evaluation."
2. Structural Integrity Associates Calculation No. DRES-07Q-302, Revision 0, "Dresden Unit 3 Main Steam Line Vibration Data Reduction."
3. ANSYS/Mechanical, Revision 5.7, ANSYS, Inc., December 2000.
4. Structural Integrity Associates Calculation No. QC-11Q-303, Revision 0, "Analysis of Main Steam Piping."
5. PIPESTRESS 2000 Solver, Version 3.5.0 +67, June 2001, DST Computer Services, S.A.
6. ASME OM-S/G Standard, Part 3, 1994 Edition.
7. Structural Integrity Associates Report Number SIR-04-001, "Quad Cities Unit 1 Main Steam Vibration Assessment," dated January 13, 2004.

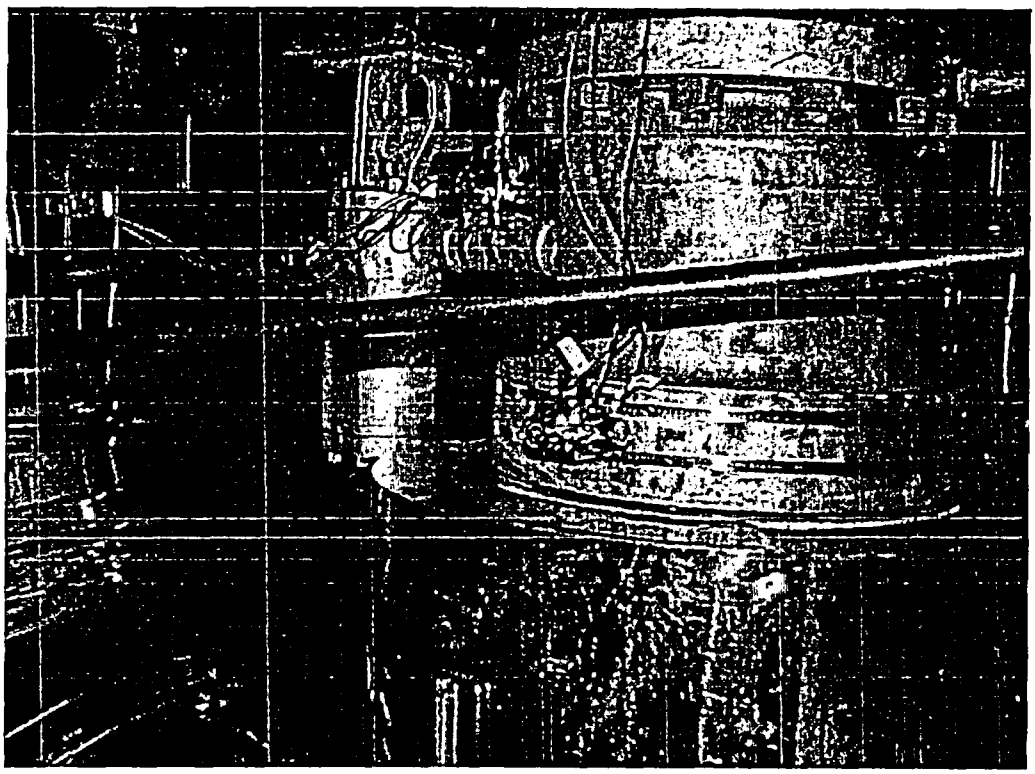


Figure 1: ERV-3B Assembly

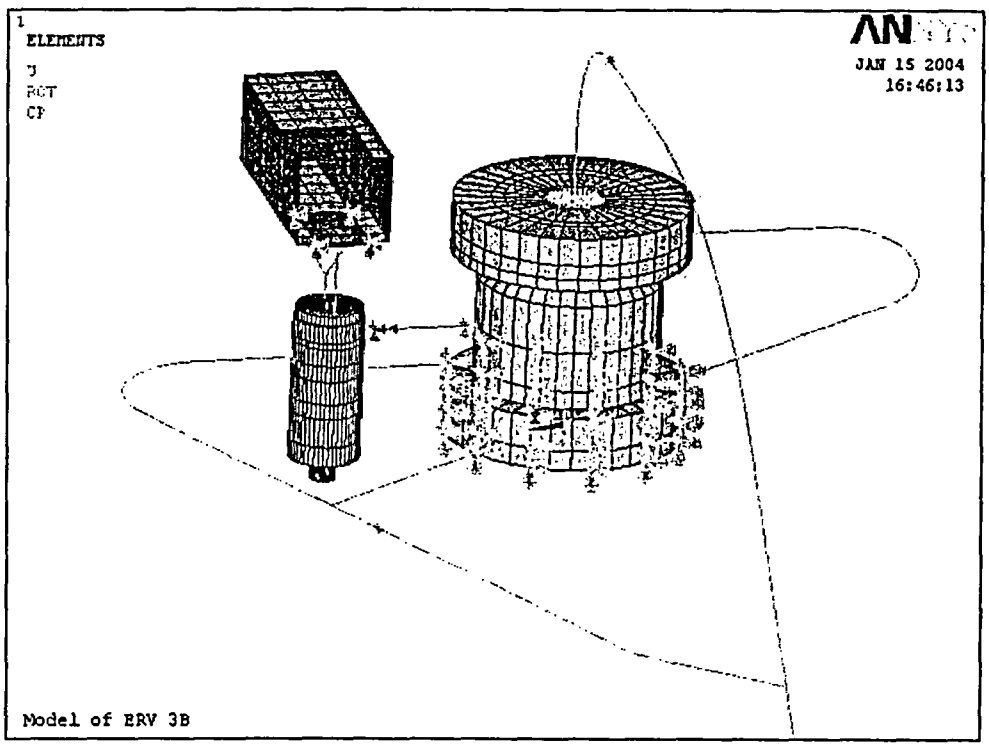


Figure 2: Dresden ERV 3B Assembly Model (Actuator is tangential to ERV axis)

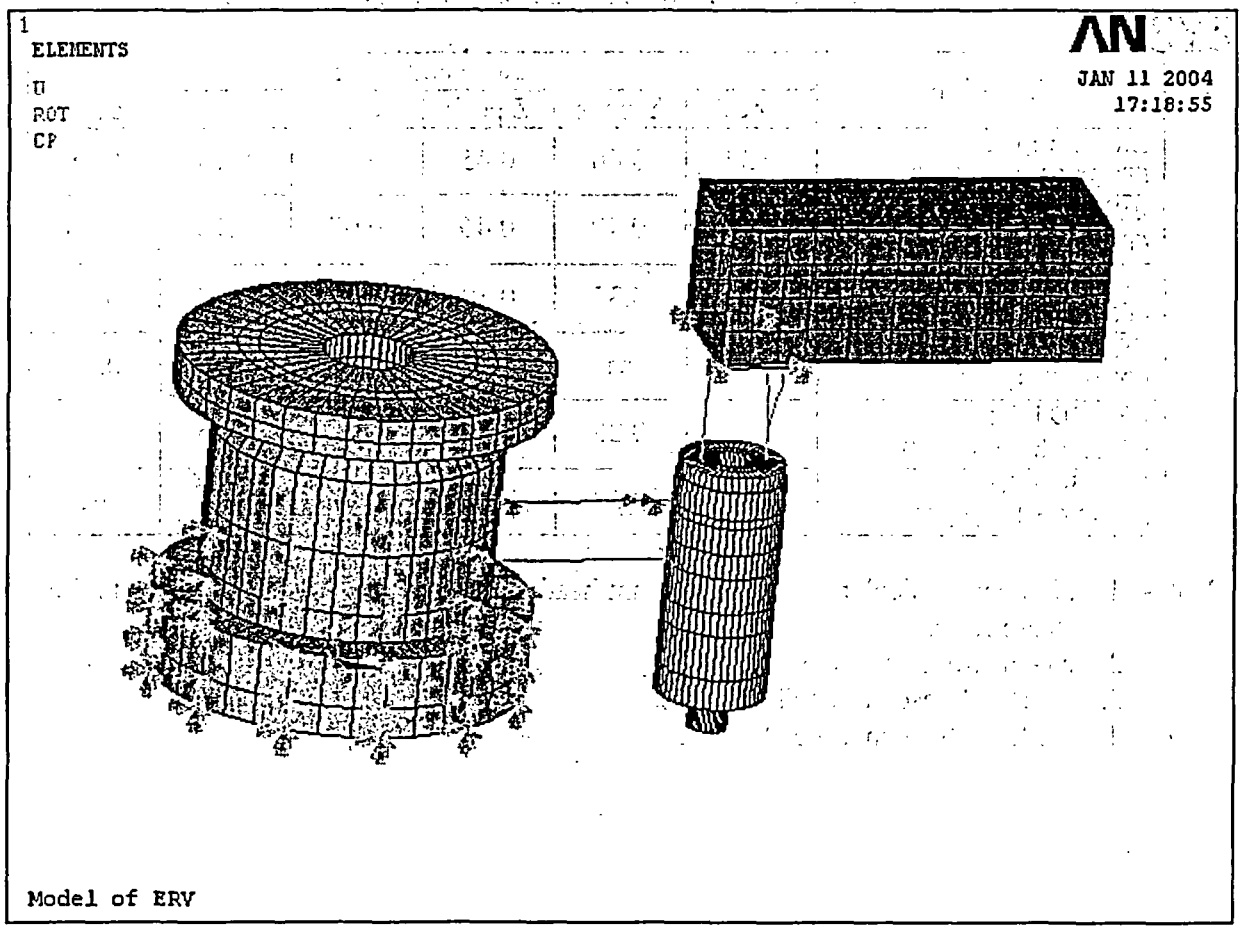


Figure 3: Quad Cities ERV 3B Assembly Model (Actuator is radial to ERV axis)

Table 1. ERV Measured Accelerations at 2851 MWth

Location	Acceleration (g)					
	X peak	Y peak	Z peak	X rms	Y rms	Z rms
ERV 3B Inlet (TC32 Ch 16, 17, 18)	0.34	0.66	0.45	0.07	0.13	0.08
ERV 3B Pilot (TC32 Ch 4, 5, 6)	0.36	0.49	0.46	0.07	0.10	0.10
ERV 3E Inlet (TC32 Ch 22, 23, 24)	0.42	0.35	0.30	0.09	0.08	0.07
ERV 3D Inlet (TC16 Ch 1, 2, 3)	0.42	na	0.46	0.09	na	0.09
ERV 3D Pilot (TC32 Ch 10, 11, 12)	na	0.28	0.52	na	0.06	0.12
ERV 3C Inlet (TC32 Ch 19, 20, 21)	0.33	0.88	na	0.07	0.17	na

Notes:
1. X is perpendicular to the main steam header, Y is vertical and Z is parallel to the steam flow.
2. na - instruments were bad
3. TC16 – Dresden Tape Deck 16
4. TC32 – Dresden Tape Deck 32

ATTACHMENT 1E

Engineering Change Evaluation 346402
"D3M10 Dryer Repair, Main Steam Walkdown Results and Evaluation for Return to Extended Power Uprate (EPU) Power"

EC Eval 346402
D3M10 Dryer Repair, Main Steam Walkdown Results and Evaluation for Return to Extended Power Uprate (EPU) Power

Reason for Evaluation/Scope:

The 2003 failures at Quad Cities Unit 1 and Unit 2 upper hood areas were a result of fatigue due to low frequency pressure pulsing. In view of these failures, Dresden Unit 3 was preemptively repaired during Dresden Unit 3 Maintenance Outage 10 (i.e., D3M10). This review documents the completion of the repair and acceptability to return Unit 3 to EPU operation.

Additionally, a comprehensive walkdown of the Unit 3 main steam (MS) system was performed on December 6, 7, and 8, 2003. These walkdowns were performed as part of the extent of condition review resulting from the damage discovered at Quad Cities Unit 1 when it was shut down to repair damage to the steam dryer. The purpose of this walkdown was to identify whether or not Dresden Unit 3 had similar damage. This evaluation provides a listing of all issues identified and their disposition. The walkdowns included inspection of specifically identified components and a general inspection of all related areas. Corporate and Site Engineering developed the scope of the walkdown after a review of the issues identified at Quad Cities in Reference 1.

Detailed Evaluation:

The scope of the dryer modification is intended to address all of the applicable repairs necessary to prevent the Quad Cities Unit 1 and Unit 2 failures from occurring at Dresden Unit 3. The specific items addressed in the modifications included:

- Removal of the existing ½" thick vertical and top horizontal plates from the outer hood at 90 and 270 degree and replacement with 1" thick plates.
- Removal of the four diagonal struts in the 90 and 270 degree outer hood and, if required, removal of the startup instrumentation brackets, pipe, or fittings from the outer hood on both sides.
- Install six gussets to the outer hood at 90 and 270 degrees. These gussets stiffen the connection between the side of the outer hood and the lower cover plate and also reduce bistable flow between the two MS nozzles, thus reducing the overall load on the dryer outer hood.
- Install thirteen mitigation bars on all tie bars.
- Drill holes in the steam dryer skirt and support channel to arrest the propagation of the fatigue cracks and minimize susceptibility to future damage.

The steam dryer modification was performed in accordance with General Electric Nuclear Engineering (GENE) Field Deviation Disposition Requests (FDDRs) EB3-0076, Rev. 0, EB3-0077, Rev. 0, and EB3-0081, Rev. 0., and GENE Specifications 26A5733, Rev. 2, "Reactor Internals Modifications Materials Requirements," and 26A5734, Rev. 2, "Reactor Internals Modifications." These modifications address the effects of the failure

mechanism identified in the Quad Cities Unit 2 root cause report (RCR) and Nuclear Event Report (NER) QC-03-042.

Regarding the MS piping issues identified at Quad Cities Unit 1 during the recent forced outage for dryer repairs, the recommendations from Corporate Engineering provided in Attachment A were implemented as detailed in Attachment B. Attachment B contains the inspection results. In general, there were no significant discrepancies. The discrepant items identified are similar to those normally found after a long run of the unit. Each discrepant item identified was either repaired or evaluated for acceptability.

The following list describes each discrepant item identified and provides a reference to the disposition of each item in bold type.

- Binding on 3E Electromatic Relief Valve (ERV) upper guide bracket **CR 189474/WR 123341**
- Wrench teeth marks on 3E ERV pilot nozzle **Non EPU, see Att. B**
- 4D safety valve acoustic sensor has gap **WR 123321**
- Target Rock Safety/Relief Valve (S/RV) bellows alarm hose fitting clamp loose **WR 123320**
- Target Rock S/RV acoustic sensor improperly mounted **CR 189466/WR 123449**
- Disconnected spring can on main steam safety valve (MSSV) F leakoff **WR 123326**
- Bolt/nut missing on MSSV G leakoff **WR 123327**
- Minor air leak on 1C main steam isolation valve (MSIV) manifold **Eng Eval 346212**
- Minor air leaks on the 2D and 2A MSIVs **Eng Eval 346212**
- Heating, ventilation, and air conditioning (HVAC) ductwork near MSIVs damaged **WR 123312**
- Flow restrictor sensing lines – six supports with loose nuts **WR 123429**
- Flow restrictor sensing lines – one missing bolt at base-plate **WR 123481**
- Head vent line – two loose bolts on clamp **WR 123487**
- Head vent line – U-bolt nut backed off **WR 123493**
- B main steam line (MSL) X-area – spring can support missing nut **WR 123241**
- C MSL X-area – wire in spring can **WR 123437**
- Reboiler steam line elbow insulation damage **WR 123250**
- Rod hanger for pipe support off MS bypass valve drain header loose **WR 123331**
- Support downstream of 3-0261-30C loose **WR 123333**
- Support downstream of 3-5641-12A,B sheared bolts. If this were vibration induced, the bolts would be expected to have pulled out of the wall instead of shearing as they did. **WR 123335**
- Turbine Stop Valve (TSV) -1 linear variable differential transformer mechanical damage **CR 189618/WR 123421**
- TSV-4 missing side cover plate bolt **WR 123257**

- Electrohydraulic Control System (EHC) support rubber grommet degradation **WR 123240/123243/123248**
- EHC piping touching grating **WR 123242/123245/124711**
- Light hanging from wires near TSV-4 **WR 123251**

Additionally, as mentioned in Attachment A, the 2M MS flow switch was replaced with the removed switch sent for analysis.

Conclusion/Findings:

The dryer has been upgraded and is completely acceptable for EPU operation with at least an interim performance monitoring program to evaluate and trend performance.

The MS system issues identified above were compared to those identified at Quad Cities Unit 1. The Quad Cities findings are documented in Condition Reports (CRs) 188050, 188052, 188128, 188185, and 188202 (Reference 1). While multiple items were identified on these walkdowns, the extent of the issues does not rise to the magnitude of those identified at Quad Cities. For example, the Quad Cities issues included sheared piping on the ERV pilot discharge line, broken welds on the ERV actuator covers, damage to the actuator internals and a damaged pipe snubber (i.e., seismic support). Since the issues identified on these walkdowns are not unlike those normally found after a long run of the unit, it is concluded that Dresden Unit 3 has not seen a significant increase in vibration-related issues post EPU. Therefore, there is no significant increase in the risk of vibration-related damage to MS piping and components associated with returning to EPU power levels.

References:

1. Quad Cities Unit 1 CRs 188050, 188052, 188128, 188185, and 188202
2. Letter from S. Eldridge to A. Shahkarami, dated December 3, 2003 (Attachment A)
3. Dresden Unit 3 CR 189633
4. NER QC-03-042 Supplement 1, BWR Steam Dryer Integrity
5. Quad Cities 2 Dryer Failure RCR GENE-0000-0018-3359, July 2003

ATTACHMENT A

December 3, 2003

To: A. Shahkarami

From: S. Eldridge, Vibration Assessment Team Lead

Subject: Recommendations for D3M10 Actions Based on Quad Cities EOC Reviews

Corporate Engineering and the Vibration Assessment Team recommend the following actions for Dresden Unit 3 during the upcoming D3M10. These recommendations are based on the findings from Quad Cities Unit 1 walkdowns and assessments. The vibration assessment team was formed to evaluate the adequacy of EPU affected components for continued EPU power operation, particularly in areas that are affected by the increased steam flows generated by the EPU power levels.

1. Perform walkdowns as detailed in Attachment 1 – D3M10 – Unit 3 Main Steam System Inspections.
    a. Drywell walkdown teams should be directed to look at all equipment in the areas so that the results include a general walkdown of each elevation for all accessible supports and components.
        i. The reason for this general look is due to the general vibration transfer that occurs through the attachments to the drywell steel structure.
    b. If any snubber supports are identified with degraded attachment hardware, the related snubber should be functionally tested to ensure no snubber degradation has occurred.
2. Perform additional activities that are detailed as in Attachment 2 – Recommendations for D3M10 Main Steam System Extent of Condition Scope From QC Event
    a. The three activities that are currently not in scope are discussed below:
        i. Disconnect to inspect for cold spring – This action is only recommended if damage is noted on the pilot line connections or if the actuators show signs of significant degradation of internal sub-components.
        ii. The need for the addition of the tiebacks on the ERVs will be determined based on the results of the Quad Cities evaluations still in progress and the D3M10 ERV examination results.
        iii. Replacement of the ERV pilots is a contingency.
3. Install the minimum vibration monitoring points on components as detailed below.
    a. All points described need to have three orthogonal directions monitored. The actual location on each component should be mounted to match those locations on the Quad Cities components.
        i. All four ERV inlet flanges
        ii. Two ERV pilots

iii. Two MSIVs
iv. The A Target Rock Valve
v. One location monitored during original EPU power ascension.

b. It is recommended that an outside expert in installation and testing of accelerometers be brought in to assist in the mounting and pre-power ascension testing of the installed equipment.

4. The Focused Assessment Team has determined that the additional actions below need to be taken related to determining wear degradation on vulnerable components:
   a. Remove and replace one of the MSL high flow switches and send for analysis to determine if wear related degradation is occurring.
   b. Inspect EHC support rubber grommets and evaluate for increased wear.

ATTACHMENT 1
D3M10 Unit 3 Main Steam System Inspections

Area/Components	Inspection Team Members	Scope of Inspection	Results of Inspections
Drywell Large Steam Piping Supports	Wu/Reda	Bolting appears tight Support configuration normal Support shifting/rubbing present	
Drywell ERV Relief Valves	Testin/Poppe/ Harvey	Actuator box physical condition Actuator mounting normal Pilot lever normal configuration Accessible bolting appears tight	
Drywell ERV Relief Valves Actuators	EMD	Remove cover of actuators and inspect internals for damage and wear W/O 642663	
Drywell Target Rock S/RV	Testin/Poppe/ Harvey	Accessible bolting appears tight Connected flex lines condition Instrument rack looks normal	
Drywell ERV Pilot Lines	Testin/Poppe/ Harvey	No signs of vibration damage No surface crack visible TE installation normal	
Drywell Relief Valve Tailpipe Supports	McGivern/ McAllister	Bolting appears tight Support configuration normal Support shifting/rubbing present	
Drywell Relief Valve/Safety Valve Acoustic Monitors	Brewer/Zelinko	Sensors installed normally, no vibration damage on cabling	
Drywell MS Safety Valves and Leakoff lines	Testin/Poppe	Accessible bolting appears tight, rams horns look normal Telltale rupture disc normal	
Drywell MSIVs	Lintakas/Winter	Actuator mounting looks normal, no vibration damage on electrical conduits, TE mounted normally, actuator J-box mounting normal, limit switches appear normal, air line supports normal, springs in good condition, air manifold components normal, no vibration damage on air lines	
Drywell Flow Restrictor Sensing Lines	Lazowski/Reuter	Piping and reservoirs appear normal Pipe supports appear normal Pipe support bolting tight Support shifting/rubbing present	
Drywell MS Drain Valves/Supports	Lintakas/Winter	Drain valve actuator appears normal Connected conduit in good condition Accessible bolting appears tight Piping supports in good condition	
Drywell Head Vent Line Supports	Lazowski/Reuter	Pipe supports appear normal Pipe support bolting tight Support shifting/rubbing present	
Drywell MS Relief Valve Vacuum Breakers	Poppe	Appears closed, physical damage	

ATTACHMENT 1
D3M10 Unit 3 Main Steam System Inspections

Area/Components	Inspection Team Members	Scope of Inspection	Results of Inspections
Large Steam Containment Isolation Valve Actuator (Isolation Condenser (ISCO), High Pressure Coolant Injection (HPCI))		Valve actuator appears normal Connected conduit in good condition Accessible bolting appears tight	
X-Area MSIVs		Actuator mounting looks normal Springs in good condition Air manifold components normal No vibration damage on air lines No vibration damage elect conduits TE mounted normally Actuator J-box mounting normal Limit switches appear normal Air line supports normal	
X-Area Large Steam Piping Supports	Wu/Reda	Bolting appears tight Support configuration normal Support shifting/rubbing present	
X-Area Drain Lines Valves		Drain valve actuator appears normal Connected conduit in good condition Accessible bolting appears tight	
X-Area Drain Line Supports	Wu/Reda	Bolting appears tight Support configuration normal Support shifting/rubbing present	
Turbine Building Steam Line Wall Penetration Supports	Poppe	Support lugs appear normal with welds intact No evidence of significant impact or wear	
Turbine Building Large Pipe Supports		Bolting appears tight Support configuration normal Support shifting/rubbing present	
Turbine Building Branch Line Supports		Bolting appears tight Support configuration normal Support shifting/rubbing present	
Turbine Building Lead Drain Tieback Supports	Wu/Reda	Bolting appears tight Support configuration normal Support shifting/rubbing present	
Turbine Building TSVs	Mourikes/Baxa	Evidence of vibration damage to valve components Connected piping in good condition Connected elect conduits in good condition	
Turbine Building Turbine Control Valves (TCVs)	Mourikes/Baxa	Evidence of vibration damage to valve components Connected piping in good condition Connected elect conduits in good condition	

ATTACHMENT 1
D3M10 Unit 3 Main Steam System Inspections

Area/Components	Inspection Team Members	Scope of Inspection	Results of Inspections
Reactor Building Corner Rooms Flow Instrument Racks	Harhoff/Borman	Rack in good condition No evidence of vibration damage to rack or piping	
Reactor Building Snoop Check All MSIV Air Lines in Drywell and X-Area Prior to Startup	Lintakas/Poppe	Snoop check does not indicate a major problem caused by excessive vibration	

ATTACHMENT 2

Recommendations for D3M10 Main Steam System Extent of Condition Scope from QC Event

Potential Activity	Include in D3M10 (Yes/No)	Reason for Decision	Scope of Work	Potential Scope Expansion
Inspect ERV Solenoid Actuators	Yes	Quad Cities has identified damage and wear in multiple actuators	Open actuators cover and perform visual inspection. Actuate limit switches and verify proper function W/R 122044	If damage or abnormal wear is discovered, the actuator will need to be removed, repaired and the standard surveillance performed
Inspect ERV Pilot Piping for Surface Discontinuities	Yes	Experience at Quad Cities has shown that surface damage to ERV pilot line piping may introduce piping failure under MS vibration conditions	Perform visual inspection of ERV pilot line piping from ERV to tailpipe W/R 122064	If surface problems are found, perform NDE as required and repair as required
Inspect All 5 S/RV and S/RV Tailpipe Supports	Yes	Quad Cities found S/RV tailpipe support non-conformances	Visually inspect supports to identify any damage, loose bolting or non-conformances W/R 122064	Pipe support repair if required. Potential for snubber test if as-found conditions warrant
Disconnect the ERV Pilot Line Piping at ERVs and Check for Cold Spring	No	This is not seen as a critical element by itself. Even if some cold spring is present imparting residual stress in the pilot piping, piping failure is not expected to occur without increased vibration and an initiating surface discontinuity	None Suggest contingency packages be prepared to perform this activity	If damage is found in the ERV actuators or supports, scope expansion to include this item may occur
Add Additional ERV Actuator Supports	No	Past history from Unit 2 has not demonstrated the need for this modification	None Suggest modifications and contingency packages be prepared to implement this activity	If damage is found in the ERV actuators, scope expansion to include this item may occur
Add Vibration Sensors to MS System	Yes	Based on damage found at Quad Cities, Management has decided vibration sensors will be added on D3 MS	Implement modfifications to install vibration monitoring sensors on MS system	None

ATTACHMENT B

D3M10
Unit 3 Main Steam System Inspections

Area/Components	Scope of Inspection	Inspectors	Date	Results of Inspection
Drywell 1, 2, and 3 Floor Large Steam Piping Supports	Bolting appears tight Support configuration normal Support shifting/rubbing present	Lazowski/Reutter	12/7	General area walkdowns on 1,2,3,4 floors did not see any major support problems or signs of significant vibration damage. All large pipe supports looked good with no discrepancies
Drywell 2 Floor ERV Relief Valves	Actuator box physical condition - Actuator mounting normal Pilot lever normal configuration Accessible bolting appears tight	Testin/Poppe/ Swart	12/7	The ERVs looked good with no evidence of vibration damage. The ERV actuator boxes and mountings were in good condition. All accessible bolting found tight. Pilot levers found normal. No vibration induced problems found
Drywell 2 Floor ERV Relief Valves Actuators	Remove cover of actuators and inspect internals for damage and wear W/O 642663	EMD / Swart / Testin	12/7	The general condition of all ERV actuators was good with no abnormal wear. The "E" ERV has an issue with potential binding. One of the spring guideposts did not extend above the top of the bushing, as it should, due to previous repairs made to the actuator. This configuration caused a small lip to be created in the brass bushing as the post wore into the bushing due to normal wear. This lip created some resistance when slight pressure was applied to depress the plunger. The experienced EMD FLS present during the inspection felt strongly the ERV would actuate if requested (WO 644843 to repair the 3E ERV actuator) Work complete

D3M10
Unit 3 Main Steam System Inspections

Area/Components	Scope of Inspection	Inspectors	Date	Results of Inspection
Drywell 2 Floor Target Rock S/RV	Accessible bolting appears tight Connected flex lines condition Instrument rack looks normal	Testin/Poppe	12/7	All accessible portions of the TR valve looked good including tailpipe support and all connected flex hoses and conduits. There was no evidence of any vibration damage. All bolting looked tight. The acoustic sensor was not in the correct location. It was located downstream of the tailpipe pipe support. The sensor should be move closer to the valve per the IM procedure. It is believed that it would function properly in the current location
Drywell 2 Floor ERV Pilot Lines	No signs of vibration damage No surface crack visible TE installation normal	Testin/Poppe/ Swart	12/7	No vibration damage or surface cracks were found on the pilot and main disc leakoff lines. All pilot leakoff nozzles looked excellent except the E ERV had a wrench teeth mark near the union thread area. Multiple wrench marks and other indentations were detected on the leakoff piping downstream of the unions. Engineering reviewed wrench marks and determined the tool marks were acceptable as is All ERV leakoff piping contained threaded unions and not flanges. Dresden piping and instrumentation diagram shows flanges. Quad Cities has flanges installed. Modifications M12-2 (3)-80-038 (EC 2357/2407) revised the design of this piping and installed bends and threaded unions. Engineering generated a calculation to justify unions and threaded joints. There was a cross-threaded joint that was repaired under WO 645413
Drywell B, 1, and 2 Floor Relief Valve Tailpipe Supports	Bolting appears tight Support configuration normal Support shifting/rubbing present	Lazowski/Reutter	12/7	The Target Rock support near the valve was in excellent condition. All welds were intact and all bolts were tight. All other supports on all relief valve tailpipes looked normal with no degradation noted

D3M10
Unit 3 Main Steam System Inspections

Area/Components	Scope of Inspection	Inspectors	Date	Results of Inspection
Drywell 2 Floor RV/SV Acoustic Monitors and TEs	Sensors installed normally No vibration damage on cabling	Zelinko	12/7	All acoustic monitor sensors were verified tight to the pipe and all cables were free of damage from the sensor to the penetration. All acoustic sensors were verified flat to the pipe except the 4D, which was slightly elevated due to being mounted over a weld (WO 645035 work complete) All thermocouples were verified free of damage and all t/c and lead wire conduit connections were verified hand tight The Target Rock pressure switches were verified hand tight except the bellows pressure switch mounting on the local rack was loose due to poor installation. (WO 645031 work complete) All pressure switch wiring and seal tights were verified secured and free of damage It was noted that the Target Rock acoustic sensor was not mounted in the location specified in the DIS procedure. It was mounted downstream of the pipe support on the tailpipe at the location of the t/c. In this location the acoustic would function properly and did when the valve was actuated on startup from D3R17. (WO 645035 work complete) There was no evidence of vibration damage
Drywell 2 Floor MSSVs and Leakoff Lines	Accessible bolting appears tight Rams horns look normal Telltale rupture disc normal	Testin/Poppe	12/7	All accessible parts of all MS safety valves looked good with no signs of any vibration related damage. All rams horns looked good and all rupture discs were intact. Generally all leakoff lines looked good except poor workmanship on two small piping supports. These problems were not vibration related. They were: - A disconnected spring can on the F MSSV leakoff (WO 645038 work complete) - A bolt/nut missing from a clamp on the G MSSV

D3M10
Unit 3 Main Steam System Inspections

Area/Components	Scope of Inspection	Inspectors	Date	Results of Inspection
				leakoff line (WO 645039 work complete)
Drywell 1 Floor MSIVs	Actuator mounting looks normal Springs in good condition Air manifold components normal No vibration damage on air lines No vibration damage elect conduits TE mounted normally Actuator J-box mounting normal Limit switches appear normal Air line supports normal	Harvey/Poppe	12/7	No evidence of vibration damage was noted. All springs were in good condition. The actuators were in good condition with no signs of degradation. Valve insulation looked normal. The air accumulators and piping to the valves looked good. A snoop check of the pneumatic lines/manifolds was performed. The 1C MSIV manifold has minor leak at the threaded connection to the block. Historically, this type of leak has been found and it is not indicative of excessive vibration. Engineering generated EC Eval 346212 to document acceptability
				A vertical HVAC duct near the MSIVs showed signs of damage due to a missing external stiffener. The HVAC system engineer walked it down and advised is it historical and not due to recent vibration effects (WO 646184 to repair)
Drywell 1 Floor Flow Restrictor Sensing Lines	Piping and reservoirs appear normal pipe supports appear normal Pipe support bolting tight Support shifting/rubbing present	Lazowski/Reutter	12/7	On the flow restrictor instrument sensing lines the following problems were identified: - Identified 6 supports utilizing u-bolts, where a single nut was loose on one or two of the u-bolts. One u-bolt was missing a nut. These supports were in the vicinity of the containment penetration and not near the flow restrictors (W/R 123429 tool pouch complete) - Base- plate on 1 support had 3 tight bolts installed through grating, 1 bolt was missing at base-plate (W/R 123481 tool pouch complete) These problems appear to be a workmanship issue, as certain ones are loose and others are not
Drywell 1 Floor MS Drain Valves/Supports	Drain valve actuator appears normal Connected conduit in good condition Accessible bolting appears tight Piping supports in good condition	Lazowski/Reutter	12/7	No abnormalities or vibration related damage was discovered in the drain line supports. All drain valves looked normal with no visible signs of vibration damage

D3M10
Unit 3 Main Steam System Inspections

Area/Components	Scope of Inspection	Inspectors	Date	Results of Inspection
Drywell 1, 2, 3, and 4 Floors Head Vent Line Supports	Pipe supports appear normal Pipe support bolting tight Support shifting/rubbing present	Lazowski/Reutter	12/7	In general the supports on the normal head vent line were in good condition, especially near the steam line. There were 2 loose bolts on one support of the Head vent valves 3-220-46/47. The support appears to have "walked" away from its normal installed position (WR 123487 tool pouch complete) There was also a U-bolt (2 feet below 3rd floor grating) on the normal head vent line where a nut was backed off 2". This looks like a historical issue due to the rusty appearance. It appeared to be a workmanship issue (WR 123493 tool pouch complete)
Drywell 1 Floor MS Relief Valve Vacuum Breakers	Appears closed, physical damage	Testin/Swart	12/7	The relief valve vacuum breakers were in good condition. The vacuum breakers stroked smoothly with no excessive play or wear detected. They appeared in mint condition
Drywell 4 Floor Large Steam Containment Isolation Valve Actuator (ISCO, HPCI)	Valve actuator appears normal Connected conduit in good condition Accessible bolting appears tight	Poppe	12/7	Both the ISCO and the HPCI valves were in excellent condition. There was no evidence of vibration damage.. All flex conduits were in excellent condition. There were no signs of packing leakage. All bolting was tight

D3M10
Unit 3 Main Steam System Inspections

Area/Components	Scope of Inspection	Inspectors	Date	Results of Inspection
X-Area MSIVs	Actuator mounting looks normal Springs in good condition Air manifold components normal No vibration damage on air lines No vibration damage elect conduits TE mounted normally Actuator J-box mounting normal Limit switches appear normal Air line supports normal	Lintakas	12/6	Overall, all MSIVs in the X-Area looked in good shape. The actuators, springs, spring mounting posts, packing area, manifolds, temperature elements mounted to manifold, air lines, limit switches, sealtite to junction boxes, and junction boxes were all inspected. All looked good, with no signs of vibration or other degradation. MSIVs were open when inspected, and there were no leaks from packing A snoop check was performed of the air lines from the accumulators to the manifolds. Generally, looked good. However, there were air leaks on the 2D and 2A. 2D had minor bubbling of snoop at the union connection by the accumulator. 2A had big bubbling at the union by the manifold connection and small bubbling at the actual connection to the union. These manifolds are being replaced D3R18 per the normal PM. Historically, these types of leaks have been found and it is not indicative of excessive vibration. Engineering generated EC Eval 346212 documenting the acceptability. The rubber boots that provide secondary containment where the MSLs go to the turbine pipeway were visually inspected. Boots were in good condition. The equalizing header drain line penetration through the boot looked undisturbed and in good condition

D3M10
Unit 3 Main Steam System Inspections

Area/Components	Scope of Inspection	Inspectors	Date	Results of Inspection
X-Area Large Steam Piping Supports	Bolting appears tight Support configuration normal Support shifting/rubbing present	Wu/Chhablani	12/6	In general no issues related to the EPU were identified. All the Large Steam Piping Supports (except support with a missing nut covered below) including supports at the north and south walls of the X-Area, Steel Structure Supports and Rod Supports are acceptable. Some small angle supports underneath the drain lines that are not attached to the basement floor (as they are dead weight supports) are also found acceptable. The following two issues were identified: - A nut on the 3B MS line rod support is missing. This line in next to the East wall of the X-Area and the missing nut is on the east side of the riser. (WR 123241 tool pouch complete) - A spring can support on the west side of the 3C MS line (closest to the west wall) is found to have an area with rust marks. Also a red color thin wire is found hanging from the spring can (WR 123437 tool pouch complete)
X-Area MS Drain Line Valves	Drain valve actuator appears normal Connected conduit in good condition Accessible bolting appears tight	Lintakas/Winter	12/8	The 3-0220-90s and the 3-0220-2 were inspected. No abnormalities were noted. Bolting appeared tight, and conduit was in good condition
X-Area MS Drain Line Supports	Bolting appears tight Support configuration normal Support shifting/rubbing present	Wu/Chhablani	12/8	Covered in X-Area MS Large Piping Supports (See Above)

D3M10
Unit 3 Main Steam System Inspections

Area/Components	Scope of Inspection	Inspectors	Date	Results of Inspection
Turbine Building MSL G Wall Penetration Supports	Support lugs appear normal with welds intact No evidence of significant impact or wear	Poppe	12/6	The two MSL guide supports in the G wall looked good. There was no evidence of damage or abnormal wear. All accessible welds were intact and no signs of rubbing or damage was noted A general area walkdown identified 3 issues. These were: - A misaligned stanchion support under bypass valves 7/4/1 to condenser piping. Design Engineering walked it down and concluded it was acceptable as is - The reboiler steam line elbow just off the equalizing header insulation was damaged. All pipe supports on the reboiler steam line looked in good condition. The elbow insulation had been taped up previously. This could be a poor workmanship issue combined with normal piping vibration (WO 645065 scheduled for D3R18) - A light hanging from the ceiling was hanging by its wires. The cover of the box that supports the light has come disconnected from the box (WO 644824 scheduled D3R18)
Turbine Building Large Pipe Supports	Bolting appears tight Support configuration normal Support shifting/rubbing present	Reda/Guerrero	12/7	All pipe support looked acceptable with no deficiencies or vibration damage noted

D3M10
Unit 3 Main Steam System Inspections

Area/Components	Scope of Inspection	Inspectors	Date	Results of Inspection
Turbine Building MS Branch Lines Supports	Bolting appears tight Support configuration normal Support shifting/rubbing present	Reda/Guerrero	12/7	In general, no significant vibration related damage was found similar to what Quad Cities experienced. Three deficiencies were noted on small bore line pipe supports. - A rod hanger for a pipe support off the MS bypass valve drain header is loose (WR 123331 tool pouch complete) - A unistrut pipe support downstream of valve 3-0261-30C was found loose (WR 123333 tool pouch complete) - A pipe support located downstream of valves 3-5641-12A,B was found with its anchor bolts sheared off. This does appear to be historical as the sheared end of the anchor bolt is rust colored (WO 645042 work complete)
Turbine Building Lead Drain Tieback Supports	Supports M-1212D-4, -5, -6, and -7 Bolting appears tight Support configuration normal Support shifting/rubbing present W/O 642633	Reda/Guerrero	12/7	The tie back supports are tightly insulated. There were no external signs of vibration damage or movement. It was noted that the supports were tight when physically moved with the insulation in place

D3M10
Unit 3 Main Steam System Inspections

Area/Components	Scope of Inspection	Inspectors	Date	Results of Inspection
Turbine Building TSVs	Evidence of vibration damage to valve components Connected piping in good condition Connected elect conduits in good condition	Mourikes/Chiou	12/6	The valves and connected piping were in good condition. The connected electrical conduits were in good condition. Prior to the outage position indication had been lost to TSV#1. This walkdown also focused on identifying the cause of this problem. The following issues were identified: - The TSV#1 LVDT ball joint rod end shoulder bolt (pin) was missing, the metal ears that had been holding the pin were severely gouged, and the LVDT rod dropped to its lowest position onto the housing. (WO 627786 work complete) - The hanger by the #4 TSV is missing the lower left side cover plate bolt. It appears to be historical (WO 645600 scheduled D3R18)
Turbine Building TCVs	Evidence of vibration damage to valve components Connected piping in good condition Connected elect conduits in good condition	Mourikes/Chiou	12/6	Inspected CV accumulators, electrical connections, limit switch rods, LVDT rods/hime joints connections and EHC piping connections to the actuators on TCV and TSVs. No vibration damage present. Connected piping in good condition. Connected electrical conduits in good condition
Turbine Building EHC Support Rubber Grommets	Grommets appear normal No abnormal wear of grommets No vibration damage present	Chiou/Guerrero	12/8	Found grommet support degradation 1@ below CIV #4, 2@ below CIV #5, and 1@ below CIV #2. The degradation does not appear to be EPU related (WO 647373, 647375, 647376) Also found EHC piping touching grating at CIV #2, CIV #3, and CIV #5. No vibration damage present (WO 647377, 647378; WR 124711)

D3M10
Unit 3 Main Steam System Inspections

Area/Components	Scope of Inspection	Inspectors	Date	Results of Inspection
Reactor Building Corner Rooms Flow Instrument Racks	Rack in good condition No evidence of vibration damage to rack or piping	Haarhoff/ Bormann	12/8	Racks and piping inspected – no evidence of damage to rack or piping
Snoop Check All MSIV Air Lines in Drywell and X-Area Prior to Startup	Snoop check does not indicate a major problem caused by excessive vibration W/O 642647 – Cancelled – done under the W/D plan. No WO needed	Lintakas/ Harvey	During Outage	Snoop checks discussed in above MSIV items. EC 346212 justifies as-found air leaks
Reactor Building MS Flow Switches	Replace the 2M flow switch and provide removed switch to Zelinko for analysis	IMD	During Outage	WR 122819/WO 644001 – Complete

ATTACHMENT 2

Extended Power Uprate Extent of Condition Reviews

Report Summary and Project Instruction, Main Steam, Feedwater, and Reactor Internals and Recirculation Systems Reports

2A) Report Summary and Project Instruction

2B) Main Steam System Report

2C) Feedwater System Report

2D) Reactor Internals and Recirculation System Report

ATTACHMENT 2A

Report Summary and Project Instruction

Report Summary for the Extent of Condition Review

During a meeting with the NRC technical staff on September 23 and September 24, 2004, questions were asked by the NRC technical staff regarding the EPU Vulnerability Review conducted by Exelon Generation Company, LLC (EGC). The following is a summary of the questions posed by the NRC, and the response to each question.

What was the scope/depth of review?

As stated in the Project Instructions for this review (i.e., Attachment 4), the purpose of the review was to:

- Identify EPU-related vulnerabilities for the Dresden Nuclear Power Station (DNPS) and Quad Cities Nuclear Power Station (QCNPS), and

- Develop recommended actions to mitigate potential failures resulting from these vulnerabilities.

For the purpose of this review, an EPU-related failure was defined as a failure attributed to changes in process parameters caused by EPU, and which results in one or more of the following:

- Licensee event report
- Engineered safety features actuation
- Reactor scram
- Plant power derate
- Unplanned entry into technical specifications
- Operator work-around and operator challenge
- Unexpected accelerated degradation
- Loose part generation

Note that although it is not listed as in item in the Project Instructions, loose part generation (caused by higher post EPU flow) was also a failure mode considered during the review. In fact, page 30 of the project instruction provides a recommendation or output of the vulnerability review for feedwater sample probe failure. During the Boiling Water Reactor Owners Group review and challenge of the project, it was recommended that loose part generation to be added to the above list as a unique failure mode.

Who performed the reviews?

To conduct the review, EGC formed a team composed of EGC Engineering, DNPS Engineering, QCNPS Engineering, General Electric, MPR Associate, Inc., Sargent and Lundy Engineers, and other organizations on an as needed basis. Several challenge reviews were held at various milestones of the project to incorporate management and industry input into the process. After development of the process flow chart and the Project Instruction (i.e., Attachment 4), the Boiling Water Reactor Owners Group (BWROG) convened to review the documents and provide feedback. In addition to the BWROG, the Institute for Nuclear Power Operation (INPO) also attended the challenge meeting. After completion of the first group of Balance Of Plant (BOP) systems, the BWROG reviewed the results and assessed the effectiveness of the process. When the Safety Systems and second group of BOP systems reviews were

completed, the BWROG again reviewed and critiqued the final results. In addition to the BWROG, each system and groups of systems were subject to rigorous reviews by the members of the team and Senior Management Challenge Boards.

Methodology/Approach

As stated in the Project Instruction in Attachment 4, the overall structure of the review for each system is depicted in Figure 1, "Identification and Mitigation of EPU Vulnerabilities". The review was structured to identify and evaluate:

- Process parameters, such as flow, temperature, pressure, moisture and fluid state for those systems, which have been affected by EPU operation. Data and information were gathered for each of the four plants at DNSP and QCNPS.
- If the plant operating data or the interviews with site personnel (e.g., System Engineer, Control Room Operator, Non-Licensed Operator, Maintenance personnel or other designated individual) indicated that the system performance was affected by EPU, the system was considered vulnerable to potential failure.
- If a given system was assessed to be potentially vulnerable to EPU related failure, the review was expanded to cover the affected components and subcomponents in that system.
- Components were then grouped and evaluated for the changed parameter (flow, vibration, impact velocity, pressure pulsation, heat, radiation, etc.) and increased vulnerability. Operating experience at DNPS and QCNPS, along with that in the industry, was also incorporated into the component reviews.
- Component reviews for various failure modes, such as high cycle fatigue, wear, erosion, and aging, resulted in recommendations for inspections, preventive maintenance, modification, and analyses or studies.
- Each vulnerability was then assessed for level of risk (i.e, high, medium, and low). Note that since the vulnerability prioritization was subjective, the team did not remove any of the vulnerabilities or the corresponding recommendations from the list.
- Potential vulnerabilities and the corresponding recommendations were then reviewed and challenged by the team, Senior Management during the challenge board presentation, and by the BWROG.

Findings

At the conclusion of the review, over one hundred recommendations were developed and categorized in the following nine areas to address potential vulnerabilities resulting from EPU operation:

- Increased feedwater flow has increased the fatigue loading on some vessel internals, which may require more frequent inspections of susceptible components.

- Increased core differential pressure has changed jet pump flow and consequently the loading on jet pump support components. These components will require accelerated inspections.

- Changes in operating conditions have increased component wear, which will require implementation of enhanced preventive maintenance.

- Increased feedwater flow, steam flow, and recirculation pump speed resulted in increased vibration on system piping and components.

- Elimination of the standby feedwater and condensate pumps, and operation of these pumps at non-optimum flow conditions, has introduced gradual component degradation.

- Effect of increased feedwater and condensate flow on BOP valves and internal components were not addressed vigorously as part of EPU implementation.

- Increased feedwater flow has increased the effects of flow-accelerated corrosion (FAC). EPU assumptions for FAC must be validated.

- Known system deficiencies were not corrected prior to EPU implementation, resulting in more pronounced operational challenges.

- Post-extended power uprate operating and analytical margins have been reduced in some areas.

Corrective actions and planned implementation dates

After review and acceptance by the EGC leadership team, approximately three hundred-seventy corrective actions were created and entered into the EGC action tracking systems for the four units at DNPS and QCNPS. The assigned implementation dates were aggressive and were based on outage schedules and funding approval processes. Immediately after approval of the proposed corrective actions by EGC leadership, all inspections and walkdowns, and some of the modifications, were added to the scope of the Fall 2004 DNPS Unit 3 outage. Authorization has been provided to start the analytical work proposed by the team and some are currently underway. The matrix accompanying this document (i.e., Attachment 4) provides a listing of the corrective actions, the corresponding tracking numbers and assigned implementation dates.

INSTRUCTIONS FOR IDENTIFICATION AND MITIGATION OF EPU VULNERABILITIES

1.0 PURPOSE

The purpose of this instruction is to define the methodology to be used by the Extended Power Uprate (EPU) Extent of Condition Review Team for:

- Identifying EPU related vulnerabilities for the Dresden and Quad Cities (D/QC) plants power systems and
- Recommending actions to mitigate potential failures resulting from these vulnerabilities.

Overall Project Objectives are defined in the Project Charter (Reference 1).

2.0 SCOPE OF REVIEW

2.1 Significant EPU Related Failures

For the purpose of this review, a significant EPU-related failure is one that can be attributed to changes in process parameters resulting from EPU and which results in one or more of the following:

- Licensee Event Report
- Engineered Safety Features Actuation
- Reactor Scram
- Plant Power Derate
- Unplanned Entry into Technical Specification
- Operator Work-Around and Operator Challenge
- Unexpected Accelerated Degradation
-

2.2 In-scope Plant Conditions

2.2.1 The plant conditions to be included in this review will be limited to normal full EPU power conditions and those plant lineups and configurations that are not considered unusual, e.g., small load reductions to perform routine turbine valve testing.

2.2.2 Evaluation of safety and standby system will be performed separately in accordance with Section 5 of this Project Instruction, which will be established at a later date.

3.0 OVERALL REVIEW STRUCTURE FOR POWER SYSTEMS

The overall structure of the review for each system is depicted in Figure 1, Identification and Mitigation of EPU Vulnerabilities. The review is structured to identify and evaluate:

- Process parameters, such as flow, temperature, pressure, moisture and fluid state which have been affected by EPU
- Affected components and subcomponents
- Potential of a component/subcomponent failure to result in unacceptable consequences such as scram, tech spec entry, etc.
- Affected characteristics of components and subcomponents, such as vibration levels, stress, impact velocity, etc.
- Vulnerabilities and failure modes, such as high cycle fatigue, wear, erosion, aging, etc., resulting from the affected characteristics
- Priority of identified vulnerabilities
- Recommendations

Data gathered throughout the review process will be entered into a database in a consistent format across all systems reviewed. Presentation and reporting of the data will include:

- Table of system parameters that change as a result of EPU. For each such parameter, pre- and post-EPU analytic and measured values will be presented. Whenever available, the acceptable limits for these values will also be presented. Sample format is presented in Attachment 1.
- Table of components and subcomponents affected by the change in parameter(s) resulting from EPU. The EPU related vulnerability for each component/subcomponent will be evaluated and summarized in this table. Sample format for this Table is provided in Attachment 1.
- Tables documenting the results of the vulnerability assessments on a parameter/component basis; format of these tables is provided in Attachment 4.
- Recommendations and their bases as shown in Attachment 5.

Data and information will be gathered for each of the four plants at D/QC. For the evaluation phase, including determination of vulnerabilities, maximum use will be made of

bounding or limiting components and subcomponents across systems and across all four D/QC plants. Presentation of results will use bounding and combined results for both sites and all four plants when appropriate.

4.0 INSTRUCTIONS FOR REVIEW OF POWER SYSTEMS

Step 1 Collect Required Input for System

Required inputs for the system review will be collected and provided to the system review team. These inputs will include the following:

- P&ID drawings
- List of system components and subcomponents from the Passport database, with components designated as "run to failures" eliminated.
- Results of EPU reviews previously conducted by Exelon including the November 2003 Focused Review, EPU Margin Lists Report, Pinch Points, Common Cause Analysis and EPU Evaluation Changes.
- Results of GE comparison of its latest methodology and analysis for design of EPU to those used for D/QC.
- D/QC EPU Task Reports
- Other documents (such a original system and design specifications, UFSAR information, etc.) as appropriate.

Step 2 Review System EPU Documentation Assembled in Step 1

System team members will familiarize themselves with the documents assembled in Step 1 in order to effectively and efficiently conduct the system review. Appropriate information from these documents will be entered into the database in later steps.

Identify major differences between the four D/QC units in order to determine the most effective way to combine the reviews and document results in subsequent steps.

Define the physical and functional system boundaries for the system being reviewed consistent with the System Boundary Documents. Identify any significant deviations from these boundaries in the final system report. As interface issues are defined in the review process, document these in the appropriate field in the database and communicate these issues directly to the affected system review team.

Step 3 *Identify Significant Parameters that have changed under EPU Conditions*

Note: Early participation from personnel knowledgeable about the system and about EPU related changes is important to efficiently identify appropriate information.

Identify normal operating parameters important to system performance, and the limiting operational conditions for each parameter identified.

In a team brainstorm session, identify all the physical parameters important to system operation, i.e., flow rates, pressures, temperatures, etc. By examination of changes in these parameters that have or will occur, and as functions of external factors (Summer or Winter operation, for example), and other expected operational conditions (beginning or end of fuel cycle, or number or trains running, for example) determine the limiting circumstances for each physical parameter identified above. Document the basis for the limiting conditions of operation in the System Folder.

In addition, identify any system parameters that are expected to change as a result of EPU, based on review of the EPU Task Report or the Engineering Report, as appropriate, and as modified by GE's latest methodology.

Example entries for this step are provided in Table 1 of Attachment 1 for the pilot review of the Isophase Bus system.

Document results in the appropriate data fields as depicted in Attachment 1.

Step 4 *Incorporate previously identified vulnerabilities from system documentation*

The previous EPU-related reviews assembled in Step 1 may have already identified EPU-related vulnerabilities that apply to the system being reviewed. Enter these previously identified vulnerabilities for components/subcomponents in the database.

Step 5 *Obtain plant operating data from D/QC for the parameters of interest*

Obtain and review recorded data at D/QC for the parameters of interest, as defined in Step 3, for pre-EPU and post-EPU operation. Determine if data is available at the limiting operational conditions established in Step 3. If data is available, consider as appropriate estimates averages over a 7-day period. If data is not available, extrapolate to the expected limiting conditions based on available data, date trends, and analysis as appropriate. (For example, if a unit has not operated at increased core flow at the end-of-cycle, determine the likely flow rate, pump speed, amps, etc., required to reach the licensed basis – or maximum the unit expects to reach.) If extrapolation is performed, document the basis for the extrapolation in the System Folder.

For each parameter, identify the analytic pre-EPU and post-EPU values. Enter this data into the database.

For each parameter, identify the current design basis range, based on the system documentation. Enter this data into the database.

STEP 6 *Identify additional concerns and known vulnerabilities based on lessons learned at D/QC and other EPU plants*

On the basis of the interviews and information gathered below, identify any added parameters (new rows in Attachment 1, Table 1) that provide an EPU related concern, but which had not already been identified in Step 3. Similarly, identify any components/subcomponents (rows in Attachment 1, Table 2) that provide an EPU related concern but which had not been included in the Passport provided list of identified components in Step 1.

As part of the data review, identify any unexpected changes in parameter values and consider whether these could be EPU related. Add any such changes to the list of parameters to be evaluated further.

Step 6a. ***Interview plant personnel at Dresden and Quad Cities.***

Interview system engineers, operators and maintenance personnel at Dresden and Quad Cities as well as the GE system engineer. Utilize the Questionnaire in Attachment 2.

Step 6b. Obtain input from BWR Owners Group and INPO.

Obtain input from BWROG regarding EPU experience and concerns, using the questionnaire in Attachment 3.

Step 6c. Interview personnel at other EPU plants.

Interview personnel at other plants that have undergone EPU, using the questionnaire presented in Attachment 3.

Step 7 Evaluate whether changes in significant parameters are consistent with EPU evaluations

Determine whether the changed parameter is as expected by the EPU evaluations, i.e., whether the change is within the expected range.

If the operational parameter reported by the plants exceeds the expected range, proceed with the evaluation on a component/subcomponent basis, in Step 10.

Otherwise, proceed to Step 8.

Step 8 Determine if the plant operational range is consistent with the original design basis

Determine whether the changed parameter is still within the original design basis, e.g., based on the system specification. Determine the limit for an acceptable value for this parameter on a system basis, if possible and enter this information in the worksheet in Attachment 4.

If the operational parameter exceeds the original design basis, proceed with the evaluation on a component/subcomponent basis, in Step 10.

Otherwise, proceed to Step 9.

Step 9 ***Evaluate parameters that are of potential concern that are Non EPU Related***

If a concern exists that has passed through the filters of Steps 7 and 8, then the parameter of concern is NOT EPU RELATED. These conditions should be documented and the associated vulnerabilities defined; however, their basis should be identified. Typical reasons for further evaluation may be:

- Unexplained data trend
- Operation near a "cliff" such that small changes in the operating parameter may result in a problem
- Plant or industry operating experience
- Potential interactions between the system being reviewed and other systems

Perform this evaluation using the worksheet in Attachment 4. Determine whether further evaluation on a component/subcomponent basis is required. Identify and document on the worksheet the basis for why the vulnerability needs to be analyzed (i.e., plant material condition, lack of perceived margin, failure history, etc.).

Step 10 *Proceed with vulnerability evaluation on a component/subcomponent basis; group components by type and/or function*

Sort the list of components from Passport by the EPN (or other appropriate field) to group components and subcomponents by type or function. This would facilitate evaluation of like components.

Eliminate component and subcomponent types that would not be affected by EPU.

Step 11 *Validate component/subcomponent list by comparison to P&ID*

Review the components and subcomponents provided from Passport. The run-to-failure components with no PMs have been eliminated from this list. Review the P&ID. Determine whether EPU-related failure of additional components or subcomponents shown on the P&ID could meet the criteria of a significant EPU-related failure as defined in Section 2 above. Add any such additional components/subcomponent to the database and carry forward to further evaluations.

NOTE: Obtain a review of this list of components /subcomponents by the system engineer and/or EPU-engineer as a check on the completeness and appropriateness of the compiled list of to be evaluated components/subcomponents.

Step 12 *For each parameter identified in Step 11, identify the components and subcomponents that are affected.*

Determine which components and subcomponents are affected by each changed parameter.

Identify the parameter of concern in the PARAMETER field of the database for each component and subcomponent to which this parameter change applies. Any components and subcomponents that are NOT affected by any of the changed parameters will be eliminated from further consideration in Step 13.

Further evaluations are performed for each parameter/component combination.

Step 13 *Determine whether the failure of the components and subcomponents identified in Step 11 lead to unacceptable failure effects*

For each component and subcomponent not eliminated in Step 12, determine which (if any) of the following unacceptable consequences can result from a failure of the component or subcomponent:

- LER
- ESF Actuation
- Reactor Scram

- Tech Spec Entry
- Power De-Rate
- Operator Work-Around Challenges
- Unexpected Accelerated Degradation

Document the results of this evaluation in the FAILURE VULNERABILITY field of the database.

NOTE: EPU parameters that do not result in one of the defined unacceptable failures will not be considered further in this review.

Step 14 *Identify the critical characteristic and associated potential failure mode and determine the effect of the changed parameter on the critical characteristic of the component/subcomponent.*

For each parameter/component/subcomponent of concern with failure vulnerability as identified in Step 13, determine the *critical characteristic* of concern and the *associated failure mechanism*.

NOTE: Examples of critical characteristics include stress, vibration level, temperatures for EQ evaluations, etc – i.e., any characteristic, resulting from the change in the EPU parameter, which has the potential for inducing a failure in the component or subcomponent.

Examples of failure mechanisms associated with the critical characteristic include wear, fatigue, erosion, aging, etc. A typical difference between a critical characteristic and a failure mechanism is that the critical characteristic can usually be measured or calculated. However, in many cases a numerical value for a critical characteristic will not be available. It is not the intention of the review team to calculate critical value parameters or associated limits if these do not already exist in available calculations.

Determine the effect of the changed parameter on the critical characteristics of the component and subcomponent being evaluated. Use the guidance provided in Attachment 4.

When practical, the review documented in Attachment 4 should result in quantitative values of critical characteristics that can be compared to acceptance criteria.

Where available, obtain actual values for the critical characteristic based on testing at plants (e.g., vibration testing of Electromatic relief valves, internal inspection results for SRV, and NDT inspections for heat exchanger tubing).

For each such critical characteristic determine, when possible, a limiting value or acceptance criteria such as analytic design limits, component specification, industry guidance, experience at other EPU plants or operating experience at D/QC.

Document the results in the following fields in the database:

CRITICAL CHARACTERISTIC
FAILURE MECHANISM
EXPECTED VALUE FOR CRITICAL CHARACTERISTIC
ACTUAL VALUE FOR CRITICAL CHARACTERISTIC
LIMIT

Identify all instances where the expected or actual characteristic is outside a defined limit. Specifically,

- Compare the "Expected Characteristic" to the "Actual Characteristic". If the Actual Characteristic is worse (less margin to limit) then identify this as a GAP by entering YES in the GAP field, with an explanation of the identified gap.
- Compare the "Expected Characteristic" and the "Actual Characteristic" to the Limit. If either the "Expected Characteristic" or the "Actual Characteristic" does not meet the limit, identify this as a GAP by entering YES in the GAP field, with an explanation of the gap.
- For all items requiring "action" as documented on the worksheet in Attachment 4, enter a YES in the database field labeled GAP.

Step 15 *Prioritize vulnerabilities based on severity, probability of occurrence and likelihood of detection*

Note: The system review team will perform this step by constructing a recommended prioritization; the full EPU Margin team will review this prioritization to ensure consistent application of prioritization across the various systems.

Incorporated in the database is the following Severity Number for each of the following Failure Effects:

LER	10
ESF Actuation	10
SCRAM	10
De-rates	5
Tech Spec Entry	5
Operator Work Around Challenge	1
Unexpected Accelerated Degradation	1

For each GAP identified, assign a Probability Number of 10, 5 or 1 for failure, with high probability at 10, medium 5, and low probability at 1. Consider:

- Margin available to the identified limit,
- Operating experience at D/QC and other plants,
- Expected time of operation at the condition of concern and
- Reliability data from PRA or other sources.

Enter this probability into the database.

Determine the potential for the failure remaining undetected, e.g., a controller on an AOV may have been damaged by continuous vibration but its failure is not apparent until the valve is asked to operate to perform a critical function.

For incipient failures that would be detected by regularly scheduled PM or other planned inspections, assign a Detectability Number of 1. For failures that would be detectable immediately after failure (e.g., a leak in a Main Steam line) assign a Detectability Number of 5; for failures that would remain undetected until the component is called on to act, assign a Detectability Number of 10.

Determine an overall Risk Priority by multiplying the Severity Number by the Probability Number by the Detectability Number.

Assign the following categories for the calculated Risk Priority Numbers:

Risk Priority Number	Priority
250-1000	High
25-125	Medium
1-10	Low

Step 16 Provide Recommendations

Note: The system review team will perform this step by defining recommended actions. The full EPU Extent of Condition Review team will review these recommendations to take advantage of the insights and experience of the entire team.

For low priority vulnerabilities define the risk, justify continued safe operation and provide the recommended basis for no repair to Exelon Management.

Summarize this recommendation in the RECOMMENDATION field of the database and if needed provide a more detailed justification in the final system report.

For the medium and high priority failures with unacceptable consequences identified in Step 15, identify potential plant modifications or changes to plant operating strategies at D/QC that will reduce or eliminate the probability of failure.

Consider the associated need for monitoring, inspection programs, and enhanced Preventive Maintenance.

Document the basis for each recommendation using the format provided in Attachment 5.

Summarize the recommendation in the RECOMMENDATION field of the database.

Step 17 *Considering the nature of the recommendations, the current EPU process should be reviewed by GE for potential changes or enhancements. (Not an EPU EOC team effort.)*

5.0 EVALUATIONS OF SAFETY SYSTEMS

This section will be established at a later date.

6.0 REFERENCES

1. GENE/Exelon Corp EPU EOC Project Charter

7.0 ATTACHMENTS

1.0 EPU Extent of Condition Database Fields
2.0 Questionnaire for Interviews of Dresden and Quad Cities Personnel
3.0 Questionnaire for Interviews of Other EPU Plant Personnel
4.0 Guidance for Evaluating Vulnerabilities by Identifying Potential Failure Mechanisms and Determining Effect of Changed Parameter on Critical Characteristics of Components and Subcomponents
5.0 Sample Summary Table of Recommendations and Bases

8.0 FIGURES

Figure 1 – Identification and Mitigation of EPU Vulnerabilities

Attachment 1

EPU Extent of Condition Database Fields *(example from Isophase bus pilot review*

System Parameters that Change as a Result of EPU

System	Parameter	Analytic Pre-EPU level	Analytic Post-EPU level	System Design Basis Value	Limiting Pre-EPU Plant Data	Limiting Post-EPU Plant Data
Isophase Bus Duct	Current in phase conductors	31,100 A @920MVA, 17.1kV or 29,500 A @ 920 MVA, 18kV (ref 2)	32,413 A @ 960MVA, 17.1kV or 30,800 A @960MVA 18kV. (ref. 2)	32,000 amps	29,006 amps (ref. note 3)	31,227 amps (ref. Note 4)

EPU-Related Vulnerabilities of Components/Subcomponents

Component/ Subcomponent	Failure Vulnerability	Critical Characteristic	Failure Mechanism	Expected Value for Critical Characteristic	Actual Value for Critical Characteristic	Limit	GAP	Risk Priority	Recommendation	Comments
Phase bus conductor	Scram due to inadequate current carrying capacity	Current carrying capacity without cooling	Conductor overheats from excess current	32,413 A (ref 2)	31,227 A (ref 4)	33,000 A (ref 2)	NO	NA	NA	

Attachment 2

Questionnaire for Interviews of Dresden and Quad Cities Personnel

D/Q EPU EOC INTERVIEWS

Date: _____

Interviewer(s): _____

_____

PARTICIPANTS	TITLE

Topic(s): _____

The purpose of this interview is to identify potential EPU-related vulnerabilities that could result in one or more of the following:

- License Event Report
- Engineered Safety Features Actuation
- Reactor Scram
- Plant Power Derate
- Unplanned entry into Technical Specification
- Operator "Work Around" that increases the risk to one of the above events.
- Unexpected accelerated equipment that increases the risk to one of the above events.

Suggested Interview Groups:

- **D/Q Plant Personnel:**
  - System Engineers
  - Reactor Engineers
  - Program Engineers
  - Design Engineers
  - Maintenance Personnel
  - Operations Personnel
- **Exelon Corporate Technical Staff**
- **Plant personnel** for utilities that are currently in the EPU evaluation process or have completed partial or full EPU implementation
- **NSSS and AE Personnel**

Note: Responses to the interview questions should consider different operating modes/configurations: normal operation (summer and winter); components out of service (e.g., FWH string, FW pump); ECCS conditions.

Systems:		
	Question	**Response**
1	How has system operation changed following the EPU? What parameters have changed (pressure, temperature, flow, fluid state, moisture content)?	
2	Is system operation since the EPU as expected? Have there been any unexpected problems, disappointments, concerns or issues?	
3	Was system operation post-EPU affected by changes in the performance of other systems?	
4	What components were affected by the EPU—either directly by changes in system parameters or indirectly by changes in interfacing systems?	
5	What modifications were made to the system to support operation at the EPU conditions? • What failure modes or vulnerabilities were addressed by the modifications? • Did the modifications (as implemented) create any new failure modes or vulnerabilities? • Were you satisfied with the modification as implemented? • Did the modifications (as implemented) adequately address the existing material condition issues? • Has system performance post-modification been satisfactory	

Procedures:		
	Question	**Response**
1	Were operating procedures changed to support operation at EPU conditions? • What failure modes or vulnerabilities were addressed by the procedure changes? • Did the procedure changes create any new failure modes, vulnerabilities or operator challenges? • Were you satisfied with the procedure changes as implemented? • Has system operation with the revised procedures been satisfactory?	
2	Were PM procedures for systems, components and subcomponents revised to reflect , with regards to frequency and scope, accelerated wear and degradation? • Which PMs were revised? • What was the basis for the changes? • Are there additional PM changes necessary to ensure system and component reliability at the EPU conditions?	
3		
4		
5		

Material Condition:		
	Question	**Response**
1	What are the material condition issues associated with the components in the system?	
2	Does operation at EPU conditions exacerbate any of these material condition issues?	
3	Was accelerated degradation of components with material condition issues adequately addressed by the EPU Project?	
4		
5		

Attachment 3

Questionnaire for Interviews of Other EPU Plant Personnel

Utility: _____

Plant(s): _____

Completed by: _____

Phone Number: _____

E-mail Address: _____

Date of EPU implementation(s): _____

Current Percent Power Increase(s) from Original Licensed Power _____

Background and Purpose of EPU Component Failure Experience Survey

In response to questions from NRC senior management, the BWROG Executive Oversight Committee has authorized formation of a generic Extended Power Uprate Committee. The BWROG has made a commitment to the NRC to survey BWR plants that have substantial EPU (greater than 5% power increase) operating experience in order to determine the material condition of system components and subcomponents that may be impacted by increased flow rates and changes in flow-induced vibration frequencies and magnitudes (main steam, feedwater, and BOP components in particular). The NRC is concerned with the potential for latent component failures. The BWROG has also committed to review and summarize the INPO power uprate database.

The following 13 BWRs are requested to complete the following survey as these plants have substantial experience operating under EPU conditions:

- Dresden 2,3
- Quad Cities 1,2
- Monticello

- Hatch 1,2
- KKM
- KKL
- Duane Arnold
- Brunswick 1,2
- Clinton

Since the BWRVIP is focusing on steam dryer issues, this committee will focus on other components such as piping supports, instrument probes, valves, and internal mechanisms of components. Steam dryer failures will be tabulated only in a cursory manner.

Participating utilities are also requested to carefully review recent periodic maintenance experience in order to assess whether there has been indications of increased component wear rates and/or increased parts replacement during preventative maintenance of plant equipment that can be attributed to EPU.

Please provide your responses to the following questions at your earliest opportunity:
(double click boxes to select your answer)

1) Have you experienced steam dryer component failures that have been attributed to EPU?
   - ☐ Yes
   - ☐ No

2) Please list component / subcomponent failures for which you believe were caused by implementation of EPU. Include also time of failure detection in months following the extended power increase. If the component was redesigned following failure please explain briefly what design changes have been made:

	Component / Subcomponent	Failure Detection Time (Months after final implementation of EPU)	Brief Description of Component Redesign (if applicable)	Describe Previous Component Failures That Occurred Prior to EPU

	Component / Subcomponent	Failure Detection Time (Months after final implementation of EPU)	Brief Description of Component Redesign (if applicable)	Describe Previous Component Failures That Occurred Prior to EPU
1				
2				
3				
4				
5				
6				
7				
8				
9				
10				
11				
12				

3) Have you experienced increased component wear rates and/or increased parts replacement during preventative maintenance of plant equipment that that can be attributed to EPU?

☐ Yes

☐ No

If yes please provide the following information:

	Component Identified That Has Experienced Increased Wear or Replacement Frequency Following Implementation of EPU	Provide Specific Information That Explains the Basis for This Assessment
1		
2		
3		
4		
5		
6		
7		
8		
9		
10		
11		
12		

4) Has your forced outage power loss rate been adversely affected following implementation of EPU?

Yes ☐ No ☐

Average forced outage loss rate for 24-month period prior to implementation of EPU: _____%

Average forced outage rate for _____ month period following EPU implementation:

_____%

5) Other comments of interest regarding the impact of EPU at your plant(s):

_____

_____

_____

_____

Attachment 4

Guidance for Evaluating Vulnerabilities by Identifying Potential Failure Mechanisms and Determining Effect of Changed Parameter on Critical Characteristics of Components and Subcomponents

This attachment contains the following:

- A Vulnerabilities Evaluation worksheet, which is used to document the evaluation of the effect of a changed system parameter on a component or group of components. This worksheet is used for the system level evaluation to determine if the changed parameter may adversely affect one or more components, such that a component level evaluation is required. For the component level evaluation, this worksheet is used to determine if there are any component-level vulnerabilities resulting from EPU that need to be further evaluated.
- A table of component issues for a parameter change. This table provides guidance for how system parameter changes may affect various types of components. This table should be used when completing the Vulnerabilities Evaluation worksheet to help assure that all possible vulnerabilities are identified and evaluated.

EXTENT OF CONDITION - EVALUATION CRITERIA
VULNERABILITY REVIEW

System: [System Name]
Parameter(s): [Parameter description – can be one or more parameters]
Component(s): [Component identification – can be one or more components; N/A for system level evaluation]
Subcomponent(s): [Subcomponent – optional; can be one or more subcomponents; N/A for system level evaluation]

Instructions: The purpose of this form is to assess potential vulnerabilities related to a changed system parameter or parameters. For each attribute, the associated questions should be evaluated relative to the effect of the changed parameter on a system component or components. For the system level evaluation, if "Yes" is indicated in the "Action?" column, a component level evaluation is required for that changed parameter. For the component level evaluation, if "Yes" is indicated, the potential vulnerability must be evaluated for potential failure mechanisms and a risk ranking must be assigned.

Attribute	Evaluation*	Action?
Component ratings: Does the changed parameter exceed a component rating or design value, e.g., with regard to capacity, duty, pressure, flow, temperature, fluid state, fluid chemistry or voltage?		Yes or No
Operating margin: Does the changed parameter reduce the operating margin for a component to an unacceptable level? Does it require that standby equipment now operate during normal conditions such that the margin is reduced to an unacceptable level?		Yes or No
Material condition: Does the changed parameter have an adverse effect on the material condition of a component such that the performance of the component is significantly affected or the margin for degradation is reduced to an unacceptable level? Consider whether the material condition was degraded pre-EPU and whether EPU could aggravate this degradation unacceptably.		Yes or No
Load and load combinations: Does the changed parameter result in loads on a component (e.g., thermal or dynamic) such that the margin to failure is reduced to an unacceptable level?		Yes or No
Environmental conditions and environmental qualification: Does the changed parameter have an adverse effect on a component relative to EQ, for example, due to changes in pressure, temperature, humidity, corrosiveness, EMI or nuclear radiation (including duration exposure)?		Yes or No
Interface considerations: Does the changed parameter have a potentially adverse effect on an interfacing system (including functional and physical interfaces)? [If "Yes," this information must be communicated to the EPU review team reviewing the affected system.]		Yes or No
Mechanical/structural considerations: Does the changed parameter have an adverse effect on component vibration, shock, reaction forces, equipment foundations or pipe supports?		Yes or No
Hydraulic considerations: Does the changed parameter have an adverse effect on pump net positive suction head, allowable pressure drop, allowable fluid velocity, fluid state, margin to flashing or steam quality?		Yes or No
Electrical considerations: Does the changed parameter have an adverse effect on component power, voltage, raceway requirements, fault current, electrical insulation or motor requirements? Does the changed parameter adversely affect interfacing electrical systems (e.g., motor control center)?		Yes or No
Instrumentation and control considerations: Does the changed parameter affect component setpoints, instrument ranges or calibrations?		Yes or No

Attribute	Evaluation*	Action?
Reliability and service life considerations: Does the changed parameter result in additional component cycles, changes in maintenance requirements (e.g., online maintenance changed to outage maintenance) or increased wear? Is the current PM frequency adequate?		Yes or No

* This evaluation should consider previous evaluations such that vulnerabilities that have already been identified and dispositioned (as verified by the review team) are not identified as vulnerabilities on this worksheet.

Common Table of Component Issues for a Parameter Change

Component	Changed Operating Parameter				
	Pressure	Process Stream Temperature	Ambient Temperature	Flow	Fluid State or Steam Quality
Piping	• Design pressure • Relief valve setting • Margin to flashing	• Design temperature • Thermal expansion • FAC • Margin to flashing • Insulation requirements	• Thermal pressurization	• Vibration • FAC • Pipe supports • Margin to flashing • Relief valve capacity	• FAC • Vibration • Water/steam hammer
Pumps	• Design pressure • Relief valve setting • NPSH	• Design temperature • NPSH • Service temperature for elastomeric materials • Nozzle loads	• Motor degradation • EQ rating for motor	• New operating point vs. BEP • Flow capacity/ discharge pressure • NPSH • Vane passing frequency • Vibration • Motor rating, current, horsepower • Motor heating • Relief valve capacity	•
Heat Exchangers—Tube Side	• Tube side design pressure • Relief valve setting • Thermal performance	• Tube side design temperature • Thermal performance	•	• Tubesheet/inlet end erosion • Tube erosion • Partition plate DP • Thermal	•

Component	Changed Operating Parameter				
	Pressure	Process Stream Temperature	Ambient Temperature	Flow	Fluid State or Steam Quality
				performance • Relief valve capacity	
Heat Exchangers—Shell Side	• Shell side design pressure • Relief valve setting • Thermal performance	• Shell side design temperature • Thermal performance	•	• Tube FIV in steam zone (e.g., condenser, FWH condensing zone) • Tube FIV in liquid zone (e.g., FWH subcooling zone) • FAC at inlets (nozzle, shell) • FWHs: flashing at subcooling zone inlet • End plate wear • Internal support erosion • Drains capacity • Vent capacity • Thermal performance • Relief valve capacity	• FAC at inlets (nozzle, shell)
Valves—AOVs & MOVs	• Design pressure • Pressure locking	• Design temperature • Thermal-induced pressure locking	• MOVs: actuator capability degradation and EQ rating for motor	• Flow capacity • Control margin • Pipe vibration (valve and actuator) • Stroke time/	• Appropriate trim configuration and materials

Component	Changed Operating Parameter				
	Pressure	Process Stream Temperature	Ambient Temperature	Flow	Fluid State or Steam Quality
			• EQ rating for actuator and controller • Thermal binding • Thermal-induced pressure locking	water hammer • Margin to cavitation	
Valves—Check	• Design pressure	• Design temperature	•	• Pipe vibration	• Appropriate trim configuration and materials
Valves—Relief	• Design pressure • Relief valve setting	• Design temperature	•	• Pipe vibration • Relief capacity	• Appropriate trim configuration and materials
Tanks	• Design pressure • Relief valve setting	• Design temperature	•	• FAC at inlets • Pipe vibration • Capacity	• FAC
Turbine Drives	• Design pressure	• Design temperature	•	• Capacity horsepower requirements • Pipe vibration	• FAC
I&C	• Design pressure (I&C subject to process stream) • Measurement range/setpoint adequacy (for pressure and flow instrumentation)	• Design temperature (I&C subject to process stream) • Measurement range/setpoint adequacy (for temperature instrumentation)	• EQ rating of controllers and electronics	• Measurement range/setpoint adequacy (for flow instrumentation) • Pipe vibration • Vortex shedding for instrumentation projecting into flow stream	•

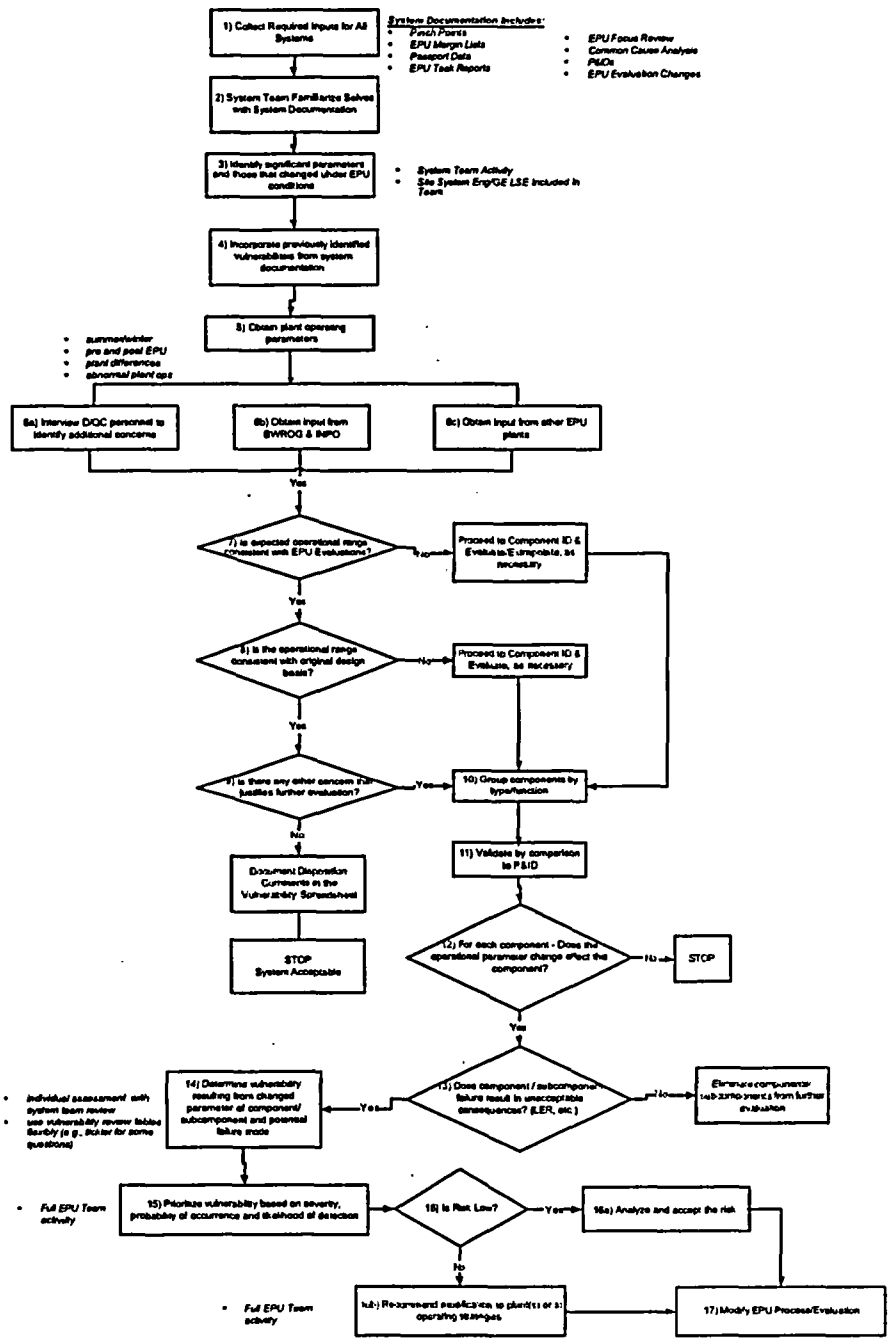
Attachment 5

Example of Recommendations and Bases

Recommendation for Q2R17	Basis	Risk
Replace Feedwater Sample Probe 2-0632 to Eliminate Concern with Vortex Shedding Failure Vulnerability	• Dresden - Three FW probe failures due to high cycle fatigue induced by flow-induced vibration • Quad Cities - Operability Evaluation CR 190513 addresses the consequences of a probe failure as a result of Dresden's three FW probe failures. CA#2 requires the inspection of this probes during Q2R17 and develop corrective actions to repair probe based upon inspection results and Dresden's root cause. • GE-NE-000-0024-2731 determined ratio of Vortex Frequency to Probes Natural Frequency as 1.026 • ASME Appendix N - Below 0.76 acceptable however other documents more restrictive • GE documents that vortex shedding frequency is sufficiently close to sample probe first natural frequency during various operating conditions that lock-on is very likely (i.e., resonant condition) • Continued successful operation of existing probe for an entire operating cycle cannot be assured	Probe failure could occur after start-up from Q2R17 and enter feedwater sparger. Damage could occur similar to Dresden Unit 2 Feedwater Sparger damage identified during D2R18. Vulnerability Prioritization Based upon ER-AA-410-1004 10 x 10 x 10 = 1000 (High) Note: The Condensate Sample Probe 2-0633 is being removed during Q2R17 per EC345445, AT180600.
Inspect Condensate Pump 2-3302B for Recirculation Cavitation Damage	• Dresden Unit 3 'C' Condensate Pump impeller found with cavitation damage due to low flow – recirculation cavitation • New impeller installed less than 2 years • Cavitation noise heard in 3C pump prior to disassembly • Two other pumps at Dresden were overhauled and no cavitation damage found. However, this type of cavitation damage was not looked for nor	Based upon the rate of degradation found on the Dresden Unit 3C condensate pump, pump performance and impeller's integrity may not be assured between refuel cycles. Significant pump performance degradation may result in foreign material generation or unit derates. Vulnerability Prioritization Based upon ER-AA-410-1004 5 x 10 x 5 = 250 (High)

Recommendation for Q2R17	Basis	Risk
	were these pumps noisy prior to overhaul. • Quad Cities Condensate pump noise is worse than Dresden's 3C and occurs on all condensate pumps • Damage areas with through wall holes indicative of recirculation cavitation due to operating away from best efficiency point (BEP) ❑ Recommendations for 2-3302B o The B condensate/condensate booster train is being worked this outage o Inspections may be performed by borescope or casing removal. However, need to inspect back side of impeller vanes (opposite side of inlet eye) and view all vanes o Pictures of Dresden's cavitation damage sent to system engineer o Terry Hoffman (Dresden) should be contacted to aid in identification of recirculation cavitation damage.	

Figure 1.0 ~ Identification and Mitigation of EPU Vulnerabilities



ATTACHMENT 2B

Main Steam System Report

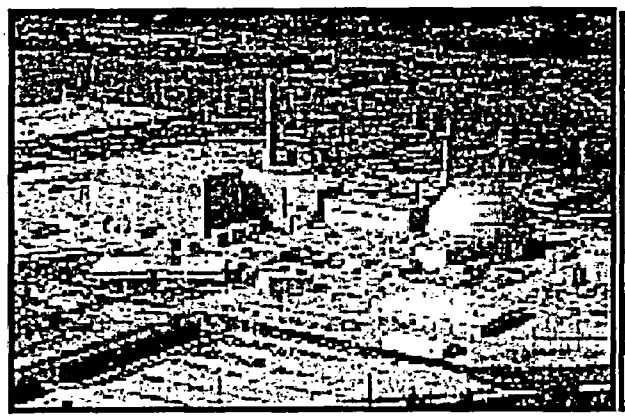
Exelon Corporation

&

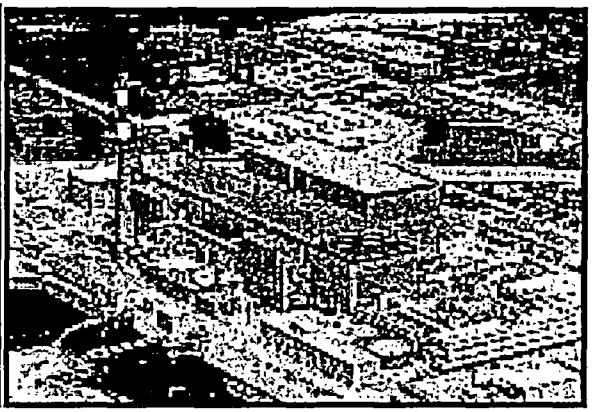
GE Energy

Extended Power Uprate

Extent of Condition
System & Component Evaluation



Dresden



Quad Cities

Main Steam System

SYSTEM NAME:	Main Steam
STATION(S):	☒ Dresden ☒ Quad Cities
SYSTEM REVIEWERS:	Bob Geier, Paul Doverspike, Don Knecht

Table of Contents

1. Executive Summary

EPU Task Reports, system descriptions, system specific data, system component list and previous EPU evaluations were reviewed to identify EPU-related vulnerabilities (LER, ESF actuation, Reactor scram, derate, OWA, Tech Spec entries, unexpected accelerated degradation) within the Main Steam System at Dresden and Quad Cities. In addition, site System Engineers, Operations and Maintenance personnel and Corporate Engineering personnel were interviewed as part of the system evaluation.

The increase in steam flow is the paramount change experienced by the Main Steam System as a result of EPU. The EPU evaluation projected an increase in steam flow from 9.81E6lb/hr (Dresden) and 9.76E6 lb/hr (Quad Cities) to 11.982E6 lb/hr. The increase in mass flow rate drives an increase in steam velocity, which manifests itself in increased system vibration. The post-EPU main steam velocities are the highest in the BWR fleet. Recently vibration data was collected at various power levels including the pre-EPU and post-EPU 100% power for QC1 and D3. In general, the vibration levels at Quad are substantially higher than at Dresden. For frequencies above approximately 50 Hz, Quad vibration levels are an order of magnitude greater than Dresden's. Quad has experienced fatigue related component failures; Dresden has not. Specifically, both of Quad's Steam Dryers (the steam dryers are addressed in the Reactor Internals evaluation) and two ERVs on Unit 1 experienced significant damage due to vibration. Similar vibration impacts have not occurred at Dresden, nor have other EPU plants reported vibration issues of the magnitude seen at Quad.

Exelon chartered a separate Vibration Task Force (VTF) to address Quad's vibration issues. The Main Steam EPU-EOC evaluation reviewed the VTF results and recommendations. At the time of the EPU-EOC evaluation, the VTF had completed the Quad Cities Main Steam Vibration evaluation and the data for the Dresden vibration evaluation had been collected and reduced. The VTF evaluation was comprehensive. It is expected that implementation of the VTF recommendations will address the main steam vibration issues. The recommendations were originally designed to address Exelon's intent to run QC Unit 1 at 100% CLTP for 3500 hours, with an intermediate limitation of 1600 hours. The VTF is currently refining their recommendations. The VTF recommendations are included, as an attachment to the Main Steam Report

Some preliminary recommendations were provided in support of the QC2 recently completed refuel outage (Q2R17). The results from implementation of the Q2R17 recommendations should be reviewed and used to refine the recommendations going forward. The Q2R17 recommendations are provided as an attachment to this report.

Following the list of recommendations are some observations. These are vulnerabilities related to EPU, reviewed during the system evaluation and have corrective actions currently in progress.

The Main Steam Sytems at Dresden and Quad were evaluated together. Below are the major differences between the two plants that were considered in the evaluation:

- At Quad Cities, branch connections off the MSL supply steam to the HPCI and RCIC systems. Dresden has an Isolation Condenser system that performs the function of the Quad Cities RCIC system. The Dresden Isolation Condenser and HPCI systems are supplied steam through separate RPV nozzles versus a MSL branch connection as at Quad.
- Quad Cities Unit 2 has PORVs instead of ERVs. At this writing, the QC2 PORVs are being replaced with ERVs.
- At Quad Cities sweepolets are used for the ERV and PORVs MSL connections. Dresden ERVs are connected to MSLs via pipe stubs.
- Dresden Unit 2's MSL venturi is smaller than the other three units.
- Currently, Dresden's Tech Specs require only 8 of 9 Main Steam Safety Valves to be operable. Note: the Target Rock Safety – Relief Valve is considered the ninth safety valve. 9 of 9 MSSVs are required to be operable at Quad Cities.

2. Recommendations and Bases (Project Instruction Attachment 5)

Below are the recommendations for the Main Steam System Recommendations from this evaluation.

Vibration Task Force Recommendations:

Exelon chartered a separate Vibration Task Force (VTF) to address Main Steam System vibration issues. At the time of the EPU-EOC evaluation, the VTF had completed the Quad Cities Main Steam Vibration evaluation and the data for the Dresden vibration evaluation had been collected and reduced. The VTF evaluation was comprehensive. It is expected that implementation of the VTF recommendations will address the main steam vibration issues. The VTF recommendations are included as an attachment. The Task Force's recommendations are under review. The recommendations were originally designed to address Exelon's intent to run QC Unit 1 at 100% CLTP for 3500 hours, with an intermediate limitation of 1600 hours.

Q2R17 Recommendations:

Some preliminary recommendations were provided in support of the QC2 recently completed refuel outage (Q2R17) (attached). The results from implementation of the Q2R17 recommendations should be reviewed and used to refine the recommendations going forward.

Observations:

Following the recommendations are some observations. These are vulnerabilities related to EPU, reviewed during the system evaluation and have corrective actions currently in progress.

Main Steam

Recommendations	Basis	Risk
Replace MSL flow ?p switches with a pressure transmitter and digital trip unit. As an option: Replace one flow switches in each steam line and perform an examination of the internals to assess the impact of pressure pulsation on the switches.	As part of the EPU project, the Main Steam line flow switches were upgraded to Barton 288 micro-switches. This increased the calibration range and the margin between normal system pressure and the switch setpoint. However, the increased MSL flow, at EPU conditions, has resulted in the new switches experiencing the same level of pressure pulsations as the old switches. There is a concern with accelerated degradation due to wear. Presently, this modification is planned for Quad Cities, but not Dresden.	Possible Group 1 isolation. Vulnerability Prioritization: (Effect*Probability*Detectability) 10X5X1 = 50 M
Provide operator guidance on 'pressure set' adjustments during a load drop. Assess the feasibility of increasing the 30 psid pressure controller setpoint.	During a load drop the operator must adjust 'Pressure Set' rapidly to avoid MSL pressure dropping below the Group 1 isolation setpoint. EPU exacerbates this issue. The MSL pressure drop increased from ~ 70 psi to ~ 95 psi. A load drop requires a greater response by the operator, at EPU conditions.. Operators must rely on judgment and experience to know how far to adjust 'Pressure Set'. Increasing the 30 psid pressure controller setpoint would result in less manual action, by the operator, in response to a load drop.	Possible Group 1 isolation. Vulnerability Prioritization: (Effect*Probability*Detectability) 10X1X5 = 50 M
Review EPU impacts on Flexibility Options in place at Dresden and Quad.	Post D/Q EPU implementation, GE Nuclear performed a process evaluation (GE NE-000-0015-5642-04, October 2003) that reviewed the impact of Flexibility Options on EPU Task Evaluations. The results of this review should be applied to the Flexibility Options in place at Quad and Dresden Note: D/Q's UFSARs no longer allow operation with a MSIV out of service; however, operator training information for the Main Steam System, states that the Main Steam System is designed to "permit high power operation with one line [MSL] isolated." The EPU high steam flows do not permit this operational flexibility.	Operating under the provisions of 'Flexibility Options that may not have been evaluated for EPU conditions. Vulnerability Prioritization: (Effect*Probability*Detectability) 10X1X5 = 50 M

Main Steam

Recommendations	Basis	Risk
Investigate the calibration accuracy of the Main Steam Flow instrumentation.	Dresden's and Quad's indicated main steam flow rates are above the analytical (calculated) limit determined from the PDLB heat balance (Task Report 0100). The heat balance used to calibrate power range instrumentation (APRMs) uses Feedwater flow. In addition, since the core thermal power is measured by the actual FW flow instrumentation, independent of the analysis performed by GE, there is assurance that thermal limits are maintained. In the analysis, steam flow is assumed to equal feedwater flow plus CRD cooling water flow. The heat balance analysis shows that reactor power is within the Licensed Thermal Power. Steam flow is used for indication and as an input to the Reactor Water Level Control system. Although, steam flow is not factored into the calibration of reactor power, the calibration of the Main Steam Flow Element is suspect.	The high steam flow indication should be investigated, as a check to verify Group I isolation is not received prematurely. Vulnerability Prioritization: (Effect*Probability*Detectability) 10X1X1 = 10 L
Reevaluate the model for Reactor Heat balance and ThermoKit for Dresden and Quad Cites focusing on: • Plant Specific Parameters • Consistency between Reactor and Turbine heat balances • Acceptability of process computer heat balance input parameters: o Feedwater Flow o Feedwater Temperature	The Thermokit is an analytical tool which forms the basis for expected core thermal power, for a given set of operating conditions such as MWe, condenser back pressure, FW temperature, FW flow, etc. While the actual Thermal Power is provided by the plant instrumentation, this tool is used to assess the differences between the actual Thermal Power and calculated value. The difference between the two values would then point to the thermal losses that can be investigated and if appropriate and cost effective, recovered. The current analytical model is based on the PDLB process (not plant specific) • Turbine heat balance (552HB1992, Rev 2), which provides 912 Mwe at 11.319 Mlb/hr steam flow. • The T0100 reactor heat balance provides 11.713 Mlb/hr at 2957 MWt Given that Dresden RWCU flow is greater than Quad Cities and results in less steam flow for the same thermal power and the fact the two analytical models do not match, it is recommended that plant specific models be developed to better mimic the plant performance.	The two analytical models for heat balances (Turbine and Reactor) are not consistent and does not account for higher RWCU flow for Dresden. Vulnerability Prioritization: (Effect*Probability*Detectability) 10 X 1 X 5 = 50 M

Main Steam Observations

	Observation	Discussion
1	Implementation of the Main Steam Vibration Task Force recommendations.	Exelon chartered a separate Vibration Task Force (VTF) to address Main Steam System vibration issues. At the time of the EPU-EOC evaluation, the VTF had completed the Quad Cities Main Steam Vibration evaluation and the data for the Dresden vibration evaluation had been collected and reduced. The VTF evaluation was comprehensive. It is expected that implementation of the VTF recommendations will address the main steam vibration issues. The VTF recommendations are provided in a handout.
2	Implementation of Technical Specification change to lower the MSL Low Pressure Setpoint.	During Turbine control valve testing, group 1 isolation isolation signals (1/2) have been received. Signal initiation is due to inadequate setpoint margin for the Main Steam Low Pressure relay. Although the issue was identified and addressed in SIL 130, the EPU evaluation did not identify the need for a setpoint change to maintain adequate margin. Reference: CRs No: 110938 & 99083.
3	Analysis to justify increasing the Main Steam Safety Valve setpoint tolerance.	Historically, drift has often resulted in MSSV setpoints outside the Tech Spec tolerance of +/- 1%. Test data indicates that a tolerance of +/- 2% is needed for a 95% confidence that draft will be within tolerance. MSSV setpoint drift out of tolerance is a pre-existing issue; however, pre EPU the ASME Overpressure Analysis could be met with a +/-3% MSSV tolerance. This was used as the standard LER response in regard to out of tolerance drift. Exelon Licensing, NFM and GE Nuclear (GNF) are involved in this analysis, currently in progress. Note: The NRC is linking this issue to the Dresden tech spec submittal to change the required number of operable MSSVs.

4	Dresden Technical Specification submittal to require 9 of 9 MSSVs to be operable.	Quad Cities Tech Specs require 9 of 9 MSSVs (includes TR S/RV) to be operable. Currently, Dresden's Tech Specs requires only 8 of 9 MSSVS to be operable. The EPU over pressurization transient analysis was performed assuming one Safety Valve out of service. Subsequently, a reload analysis for Dresden required 9 of 9 valves to be operable for overpressure protection. The transition core for Dresden was more limiting than the equilibrium core assumed in the EPU evaluation (TR 0900). Dresden has submitted a tech spec change that for 9 of 9 valves to be operable. NRC approval of the tech spec change is pending resolution of the chronic MSS setpoint out-of- tolerance issue. Both Dresden and Quad's Tech Specs require 5 of 5 Relief Valves to be operable. There is no operational flexibility for an out of service Safety or Relief valve. **Recommendation:** Perform an analysis to justify plant operation at reduced power in the event of an inoperable Main Steam Safety Valve or a Relief Valve and implement an associated Tech Spec change.

3. Known Vulnerabilities Checklist Review

System Name: Main Steam

Instructions: Place a check mark for each unit for which the document was reviewed.

Reviewed Documents	DRE U2	DRE U3	QDC U1	QDC U2
Pinch Points	☒	☒	☒	☒
Margin List	☒	☒	☒	☒
Focus Review	☒	☒	☒	☒
CHIP	☒	☒	☒	☒
SHIP	☒	☒	☒	☒
Single Point Vulnerability	☒	☒	☒	☒
Scram De-Rate Challenge	☒	☒	☒	☒
Industry Feedback	☒	☒	☒	☒
GE Task Reports	☒	☒	☒	☒
Common Cause Analysis	☒	☒	☒	☒
EPU Evaluation Changes	☒	☒	☒	☒

4. Summary of System Parameter Changes

The Main Steam System: The Main Steam System is affected primary by the increase in steam flow and the resulting increase in system vibration. Dresden's and Quad's indicated main steam flow rates are above the analytical limit determined from the PDLB heat balance (Task Report 0100). This condition existed pre-EPU, but is exacerbated at EPU Conditions. Reactor pressure remained constant; however turbine throttle pressure decreased slightly due to the increase in frictional losses associated with the increase in velocity. RPV temperature remained constant. There is a slight and negligible increase in the temperature gradient from the RPV nozzles to the turbine throttle corresponding to the pressure decrease at the throttle. The increase in N16 production is offset by the dilution from the increased flow rate. The concentration of N-16 in the steam remains essentially constant; however, the transient time from the core to the steam lines is reduced slightly providing less decay time. The overall affect is a slight increase in MSL radiation. Main steam line radiation measurements swing widely. Historically, the alarm setpoints are set at a multiple of normal background radiation at 100% power.

The table below summarizes the Main Steam parameter changes.

	Process Computer Point	Description	Unit	Summer Maximum		Winter Maximun		Analytical	Analytical	Reference
				Pre-EPU	Post-EPU	Pre-EPU	Post-EPU	Pre-EPU	Post-EPU	
D2	C222	* Steam Flow	M#/HR	9.68	12.18	10.11	12.18	9.81	11.713	TR 0100
D3	C322	* Steam Flow	M#/HR	10.02	11.91	10.03	11.89	9.81	11.713	TR 0100
Q1	QDC01V_D161	* Steam Flow	M#/HR	9.77	11.90	9.78	11.78	9.76	11.713	TR 0100
Q2	QDC02V_D261	* Steam Flow	M#/HR	9.81	11.68	9.73	11.62	9.76	11.713	TR 0100
D2	R212	MS Line Raditiation	MR/HR	3823.31	1557.72	1.07	1614.34	3 x Normal	3 x Normal	TR 0802
D3	R312	MS Line Raditiation	MR/HR	4497.94	1123.15	4354.63	959.80	3 x Normal	3 x Normal	TR 0802
Q1	QDC01V_R115	MS Line Raditiation	MR/HR	269.10	265.43	273.75	279.71	3 x Normal	3 x Normal	TR 0802
Q2	QDC02V_R215	MS Line Raditiation	MR/HR	205.85	209.68	265.97	230.34	3 x Normal	3 x Normal	TR 0802
D2	C200	Rx Pressure	psig	1005.73	1005.68	1007.97	1001.98	1020 psia	1020 psia	TR 0300
D3	C300	Rx Pressure	psig	1003.01	1000.66	1005.07	1000.41	1020 psia	1020 psia	TR 0300
Q1	QDC01V_D613	Rx Pressure	psig	1004.13	1003.14	1004.44	1003.82	1020 psia	1020 psia	TR 0300
Q2	QDC02V_D713	Rx Pressure	psig	1003.97	1004.40	1001.78	1003.50	1020 psia	1020 psia	TR 0300
D2	T200	Turb Throttle Stm Press	psig	933.96	924.88	931.85	914.76	965 psia	925 psia	TR 0700
D3	T300	Turb Throttle Stm Press	psig	935.10	905.41	935.09	899.27	965 psia	925 psia	TR 0700
Q1	QDC01V_T100	Turb Throttle Stm Press	psig	952.75	920.21	956.22	924.79	965 psia	925 psia	TR 0700
Q2	QDC02V_T200	Turb Throttle Stm Press	psig	943.13	919.48	938.20	918.34	965 psia	925 psia	TR 0700
D2	Derived Value	MSL Press Drop	psid	71.77	80.80	76.12	87.22	55 psia	95 psia	TR 0300 and TR C
D3	Derived Value	MSL Press Drop	psid	67.91	95.24	69.98	101.14	55 psia	95 psia	TR 0300 and TR C
Q1	Derived Value	MSL Press Drop	psid	51.37	82.93	48.22	79.03	55 psia	95 psia	TR 0300 and TR C
Q2	Derived Value	MSL Press Drop	psid	60.84	84.92	63.58	85.16	55 psia	95 psia	TR 0300 and TR C
D2	From Psat	Rx Temperature	Degrees	545 +/- 0.5	545 +/- 0.5	545 +/- 0.5	545 +/- 0.5	547	547	TR 0300
D3	From Psat	Rx Temperature	Degrees	545 +/- 0.5	545 +/- 0.5	545 +/- 0.5	545 +/- 0.5	547	547	TR 0300
Q1	From Psat	Rx Temperature	Degrees	545 +/- 0.5	545 +/- 0.5	545 +/- 0.5	545 +/- 0.5	547	547	TR 0300
Q2	From Psat	Rx Temperature	Degrees	545 +/- 0.5	545 +/- 0.5	545 +/- 0.5	545 +/- 0.5	547	547	TR 0300

5. Main Steam System Boundaries:

For the purposes of this evaluation, the main steam system is considered to start at the Reactor Pressure Vessel (RPV) nozzles and extend up to the Main Turbine Stop Valves. The Main Steam Line (MSL) drain lines to the Reactor Building Equipment Drain Sump and the Main condenser are also included. The MSL 'D Ring' is included; however, the Turbine Bypass Valves are scoped with the Turbine, not the Main Steam System. **For Quad Cities** the branch connections to the High Pressure Coolant Injection and Reactor Core Isolation Cooling Systems are included up to the inboard steam supply isolation valves. The Offgas steam supply lines are included up to the first isolation valves. This boundary definition is consistent with the system boundary used by the Dresden and Quad Cities System Engineers.

Major components within the system boundary include the Main Steam Isolation Valves, the Electromatic Relief Valves, the Power Operated Relief Valves **(Quad Cities 2 only)**, the Main Steam Safety Valves, the Main Steam Flow Elements, the Target Rock Safety/Relief Valves and the MSL pressure and flow instrumentation.

The Main Steam System is shown on P&ID M-60 for Quad Cities and M-12 for Dresden. The associated 'Horsenotes' are attached for convenient reference.

6. Vulnerability Review (Project Instruction Attachment 4)

Note: The information contained in this section is overlaps the information contained in the Section 2, Recommendations and Bases and Section 8, Data Base Report.

6a. System Vulnerability Review

Extent of Condition - Evaluation Criteria

Vulnerability Review

System: Main Steam
Parameter(s): Main Steam Flow
Component(s): N/A
Subcomponent(s): N/A

Instructions: The purpose of this form is to assess potential vulnerabilities related to a changed system parameter or parameters. For each attribute, the associated questions should be evaluated relative to the effect of the changed parameter on a system component or components. For the system level evaluation, if "Yes" is indicated in the "Action?" column, a component level evaluation is required for that changed parameter. For the component level evaluation, if "Yes" is indicated, the potential vulnerability must be evaluated for potential failure mechanisms and a risk ranking must be assigned.

Attribute	Evaluation	Action?
Component ratings: Does the changed parameter exceed a component rating or design value, e.g., with regard to capacity, duty, pressure, flow, temperature, fluid state, fluid chemistry or voltage?	Quad and Dresden have the highest steam velocities in the industry. This is due to a combination of relatively small Main Steam Lines (20 inch diameter) and the large mass flow required at EPU conditions. The estimated steam velocity, at EPU conditions, is 226 ft/second, which is a 23% increase from pre-EPU conditions. The increased steam velocity increases the system vibration levels. Above a frequency of 50Hz, Quad vibration levels are an order of magnitude greater than Dresden's. There are several efforts in progress to understand the cause of the high vibrations at Quad Cities. These efforts include Dryer Scale Model testing, Computational Fluid Dynamic Analysis and Acoustical Circuit Analysis. Quad steam dryers and ERVs have experienced significant degradation at EPU conditions. The Dryers have been modified. At this writing, the QC2 modified dryer has been inspection has been completed after 9 months of operations. The dryer has sustained several cracks. Initially a ERV tie back support mods was been implemented to address the vibration issues. A detail vibration analysis and modeling of the ERVs indicates that a mod is needed for the pilot valve actuating mechanism. A vibration analysis of the Main Steam System (EC # 346515) indicates that piping stresses are within code limits (ASME O&M Standard Part 3) and that the performance of Main Steam components at EPU conditions is acceptable with the exception of the ERV actuators and the HPCI 4 valve limit switch. These components, as currently configured, will experience excessive wear and fatigue degradation.	Yes

Attribute	Evaluation*	Action?
Operating margin: Does the changed parameter reduce the operating margin for a component to an unacceptable level? Does it require that standby equipment now operate during normal conditions such that the margin is reduced to an unacceptable level?	Operating margin is impacted in two areas. 1. During Turbine control valve testing, 1/2 initiation signals were received for a group 1 isolation. Signal initiation was due to inadequate setpoint margin for the Main Steam Low Pressure relay. Although the issue was identified and addressed in SIL 130, the EPU evaluation did not identify the need for a setpoint change. Reference: CRs No: 110938 & 99083. 2. Dresden's Tech Specs currently requires 8 of 9 valves to be operable. Dresden has submitted a tech spec change that will require 9 of 9 valves to be operable, similar to QC. The EPU Evaluation indicated that the Reactor over pressurization event would not require all 9 valves. Subsequently, a relo ad analysis for Dresden required 9 of 9 valves to be operable for Overpressure protection during a MSIV closure followed by a reactor scram.	Yes
Material condition: Does the changed parameter have an adverse effect on the material condition of a component such that the performance of the component is significantly affected or the margin for degradation is reduced to an unacceptable level? Consider whether the material condition was degraded pre-EPU and whether EPU could aggravate this degradation unacceptably.	The vibration level present at EPU conditions does adversely affect component material condition.	Yes
Load and load combinations: Does the changed parameter result in loads on a component (e.g., thermal or dynamic) such that the margin to failure is reduced to an unacceptable level?	Loads are increased to an unacceptable level for the ERVs. EPU Task Reports T0312 and T0316 identified several areas where the projected increases in piping stresses, due to increased flow, would exceed the allowable values. A Piping Problem Resolution Team (PPRT) was established under S&W's direction to resolve all piping issues. Piping stress limits is a function of flow, temperature, pressure and piping geometry. The specific limits, evaluations and associated recommendations are contained in Stone and Webster report Number SW-DRQC-EPU-ENG01.	Yes
Environmental conditions and environmental qualification: Does the changed parameter have an adverse effect on a component relative to EQ, for example, due to changes in pressure, temperature, humidity, corrosiveness, EMI or nuclear radiation (including duration exposure)?	No impact.	No
Interface considerations: Does the changed parameter have a potentially adverse effect on an interfacing system (including functional and physical interfaces)? [If "Yes," this information must be communicated to the EPU review team reviewing the affected system.]	The Flow induced vibration has an impact on the HPCI 4 Valve limit switch (to be included in the HPCI system evaluation).	Yes
Mechanical/structural considerations: Does the changed parameter have an adverse effect on component vibration, shock, reaction forces, equipment foundations or pipe supports?	Component vibration is impacted (see above).	Yes
Hydraulic considerations: Does the changed parameter have an adverse effect on pump net positive suction head, allowable pressure drop, allowable fluid velocity, fluid state, margin to flashing or steam quality?	Higher fluid velocity is a result of increased steam flow. Steam velocities are the highest in the industry.	Yes
Electrical considerations: Does the changed parameter have an adverse effect on component power, voltage, raceway requirements, fault current, electrical insulation or motor requirements? Does the changed parameter adversely affect interfacing electrical systems (e.g., motor control center)?	No impact.	No
Instrumentation and control considerations: Does the changed parameter affect component setpoints, instrument ranges or calibrations?	The setpoint was changed for the MSL low pressure instrument. The MSL flow instrument was changed to a model with a larger range.	Yes
Reliability and service life considerations: Does the changed parameter result in additional component cycles, changes in maintenance requirements (e.g., online maintenance changed to outage maintenance) or increased wear?	The service life and maintenance requirements for the ERVs and possibly the MSIVs are impacted.	Yes

- This evaluation should consider previous evaluations such that vulnerabilit ies that have already been identified and dispositioned (as verified by the review team) are not identified as vulnerabilities on this worksheet.

Extent of Condition - Evaluation Criteria

Vulnerability Review

System: Main Steam
Parameter(s): Main Steam Pressure: With reactor dome pressure held constant, the increase in steam flow results in a larger frictional pressure drop between the RPV nozzles and the Turbine throttle valves.
Component(s): N/A
Subcomponent(s): N/A

Instructions: The purpose of this form is to assess potential vulnerabilities related to a changed system parameter or parameters. For each attribute, the associated questions should be evaluated relative to the effect of the changed parameter on a system component or components. For the system level evaluation, if "Yes" is indicated in the "Action?" column, a component level evaluation is required for that changed parameter. For the component level evaluation, if "Yes" is indicated, the potential vulnerability must be evaluated for potential failure mechanisms and a risk ranking must be assigned.

Attribute	Evaluation*	Action?
Component ratings: Does the changed parameter exceed a component rating or design value, e.g., with regard to capacity, duty, pressure, flow, temperature, fluid state, fluid chemistry or voltage?	No impact.	No
Operating margin: Does the changed parameter reduce the operating margin for a component to an unacceptable level? Does it require that standby equipment now operate during normal conditions such that the margin is reduced to an unacceptable level?	No impact.	No
Material condition: Does the changed parameter have an adverse effect on the material condition of a component such that the performance of the component is significantly affected or the margin for degradation is reduced to an unacceptable level? Consider whether the material condition was degraded pre-EPU and whether EPU could aggravate this degradation unacceptably.	No impact.	No
Load and load combinations: Does the changed parameter result in loads on a component (e.g., thermal or dynamic) such that the margin to failure is reduced to an unacceptable level?	No impact.	No
Environmental conditions and environmental qualification: Does the changed parameter have an adverse effect on a component relative to EQ, for example, due to changes in pressure, temperature, humidity, corrosiveness, EMI or nuclear radiation (including duration exposure)?	No impact.	No
Interface considerations: Does the changed parameter have a potentially adverse effect on an interfacing system (including functional and physical interfaces)? [If "Yes," this information must be communicated to the EPU review team reviewing the affected system.]	No impact.	No
Mechanical/structural considerations: Does the changed parameter have an adverse effect on component vibration, shock, reaction forces, equipment foundations or pipe supports?	No impact.	No
Hydraulic considerations: Does the changed parameter have an adverse effect on pump net positive suction head, allowable pressure drop, allowable fluid velocity, fluid state, margin to flashing or steam quality?	No impact.	No
Electrical considerations: Does the changed parameter have an adverse effect on component power, voltage, raceway requirements, fault current, electrical insulation or motor requirements? Does the changed parameter adversely affect interfacing electrical systems (e.g., motor control center)?	No impact.	No
Instrumentation and control considerations: Does the changed parameter affect component setpoints, instrument ranges or calibrations?	Although reactor pressure did not change, turbine throttle pressure is lower due to the increase in head loss as a result of high steam flow. The Low MSL Pressure instrument is impacted by the lower turbine throttle pressure.	Yes

Attribute	Evaluation	Action?
Reliability and service life considerations: Does the changed parameter result in additional component cycles, changes in maintenance requirements (e.g., online maintenance changed to outage maintenance) or increased wear?	No impact.	No

- This evaluation should consider previous evaluations such that vulnerabilities that have already been identified and dispositioned (as verified by the review team) are not identified as vulnerabilities on this worksheet.

Extent of Condition - Evaluation Criteria

Vulnerability Review

System: Main Steam

Parameter(s): Main Steam Radiation: Main Steam Line Radiation is primarily due to N-16. There is an increase in N-16 production with an increase in core power. The increase in N-16 production is offset by dilution from an increased in steam flow rate. The concentration of N-16 in the steam remains essentially constant; however, the transient time from the core to the steam lines is reduced slightly providing less decay time. The overall affect is a slight increase in MSL radiation. Main steam line radiation measurements swing widely. Historically, alarm setpoints are three times the normal radiation level.

Component(s): N/A

Subcomponent(s): N/A

Instructions: The purpose of this form is to assess potential vulnerabilities related to a changed system parameter or parameters. For each attribute, the associated questions should be evaluated relative to the effect of the changed parameter on a system component or components. For the system level evaluation, if "Yes" is indicated in the "Action?" column, a component level evaluation is required for that changed parameter. For the component level evaluation, if "Yes" is indicated, the potential vulnerability must be evaluated for potential failure mechanisms and a risk ranking must be assigned.

Attribute	Evaluation*	Action?
Component ratings: Does the changed parameter exceed a component rating or design value, e.g., with regard to capacity, duty, pressure, flow, temperature, fluid state, fluid chemistry or voltage?	No impact.	No
Operating margin: Does the changed parameter reduce the operating margin for a component to an unacceptable level? Does it require that standby equipment now operate during normal conditions such that the margin is reduced to an unacceptable level?	No impact.	No
Material condition: Does the changed parameter have an adverse effect on the material condition of a component such that the performance of the component is significantly affected or the margin for degradation is reduced to an unacceptable level? Consider whether the material condition was degraded pre-EPU and whether EPU could aggravate this degradation unacceptably.	No impact.	No
Load and load combinations: Does the changed parameter result in loads on a component (e.g., thermal or dynamic) such that the margin to failure is reduced to an unacceptable level?	No impact.	No
Environmental conditions and environmental qualification: Does the changed parameter have an adverse effect on a component relative to EQ, for example, due to changes in pressure, temperature, humidity, corrosiveness, EMI or nuclear radiation (including duration exposure)?	No impact.	No
Interface considerations: Does the changed parameter have a potentially adverse effect on an interfacing system (including fun ctional and physical interfaces)? [If "Yes," this information must be communicated to the EPU review team reviewing the affected system.]	No impact.	No
Mechanical/structural considerations: Does the changed parameter have an adverse effect on component vibration, shock, reaction forces, equipment foundations or pipe supports?	No impact.	No
Hydraulic considerations: Does the changed parameter have an adverse effect on pump net positive suction head, allowable pressure drop, allowable fluid velocity, fluid state, margin to flashing or steam quality?	No impact.	No
Electrical considerations: Does the changed parameter have an adverse effect on component power, voltage, raceway requirements, fault current, electrical insulation or motor requirements? Does the changed parameter adversely affect interfacing electrical systems (e.g., motor control center)?	No impact.	No
Instrumentation and control considerations: Does the changed parameter affect component setpoints, instrument ranges or calibrations?	No impact.	No

Attribute	Evaluation*	Action?
Reliability and service life considerations; Does the changed parameter result in additional component cycles, changes in maintenance requirements (e.g., online maintenance changed to outage maintenance) or increased wear?	No impact.	No

- This evaluation should consider previous evaluations such that vulnerabilities that have already been identified and dispositioned (as verified by the review team) are not identified as vulnerabilities on this worksheet.

Extent of Condition - Evaluation Criteria

Vulnerability Review

System: **Main Steam**
Parameter(s): **Main Steam Temperature:** RPV temperature remained constant. There is a slight and negligible increase in the temperature gradient from the RPV nozzles to the turbine throttle corresponding to the pressure decrease at the throttle.

Component(s): **N/A**
Subcomponent(s): **N/A**

Instructions: The purpose of this form is to assess potential vulnerabilities related to a changed system parameter or parameters. For each attribute, the associated questions should be evaluated relative to the effect of the changed parameter on a system component or components. For the system level evaluation, if "Yes" is indicated in the "Action?" column, a component level evaluation is required for that changed parameter. For the component level evaluation, if "Yes" is indicated, the potential vulnerability must be evaluated for potential failure mechanisms and a risk ranking must be assigned.

Attribute	Evaluation*	Action?
Component ratings: Does the changed parameter exceed a component rating or design value, e.g., with regard to capacity, duty, pressure, flow, temperature, fluid state, fluid chemistry or voltage?	N/A	No
Operating margin: Does the changed parameter reduce the operating margin for a component to an unacceptable level? Does it require that standby equipment now operate during normal conditions such that the margin is reduced to an unacceptable level?	N/A	No
Material condition: Does the changed parameter have an adverse effect on the material condition of a component such that the performance of the component is significantly affected or the margin for degradation is reduced to an unacceptable level? Consider whether the material condition was degraded pre-EPU and whether EPU could aggravate this degradation unacceptably.	N/A	No
Load and load combinations: Does the changed parameter result in loads on a component (e.g., thermal or dynamic) such that the margin to failure is reduced to an unacceptable level?	N/A	No
Environmental conditions and environmental qualification: Does the changed parameter have an adverse effect on a component relative to EQ, for example, due to changes in pressure, temperature, humidity, corrosiveness, EMI or nuclear radiation (including duration exposure)?	N/A	No
Interface considerations: Does the changed parameter have a potentially adverse effect on an interfacing system (including functional and physical interfaces)? [If "Yes," this information must be communicated to the EPU review team reviewing the affected system.]	N/A	No
Mechanical/structural considerations: Does the changed parameter have an adverse effect on component vibration, shock, reaction forces, equipment foundations or pipe supports?	N/A	No
Hydraulic considerations: Does the changed parameter have an adverse effect on pump net positive suction head, allowable pressure drop, allowable fluid velocity, fluid state, margin to flashing or steam quality?	N/A	No
Electrical considerations: Does the changed parameter have an adverse effect on component power, voltage, raceway requirements, fault current, electrical insulation or motor requirements? Does the changed parameter adversely affect interfacing electrical systems (e.g., motor control center)?	N/A	No
Instrumentation and control considerations: Does the changed parameter affect component setpoints, instrument ranges or calibrations?	N/A	No

Attribute	Evaluation*	Action?
Reliability and service life considerations: Does the changed parameter result in additional component cycles, changes in maintenance requirements (e.g., online maintenance changed to outage maintenance) or increased wear?	N/A	No

- This evaluation should consider previous evaluations such that vulnerabilities that have already been identified and dispositioned (as verified by the review team) are not identified as vulnerabilities on this worksheet.

6b. Component Vulnerability Review

Extent of Condition - Evaluation Criteria

Vulnerability Review

System: Main Steam System
Parameter(s): Main Steam Flow
Component(s): Electromatic Relief Valves
Subcomponent(s): Valve body, Actuator, Pilot Solenoid

Instructions: The purpose of this form is to assess potential vulnerabilities related to a changed system parameter or parameters. For each attribute, the associated questions should be evaluated relative to the effect of the changed parameter on a system component or components. For the system level evaluation, if "Yes" is indicated in the "Action?" column, a component level evaluation is required for that changed parameter. For the component level evaluation, if "Yes" is indicated, the potential vulnerability must be evaluated for potential failure mechanisms and a risk ranking must be assigned.

Attribute	Evaluation	Action?
Component ratings: Does the changed parameter exceed a component rating or design value, e.g., with regard to capacity, duty, pressure, flow, temperature, fluid state, fluid chemistry or voltage?	During Q1F51, the ERV 3B actuator was discovered with significant damage and ERV 3E was discovered with excessive actuator wear. Pre-EPU vibration levels were acceptable for ERV operability and are considered the vibration limit for the initial ERV configuration. In the short term, higher vibration levels at EPU conditions will remain; therefore, the ERVs will need to be made more robust in order to withstand the higher vibration levels experienced at EPU conditions. To help understand the ERV response to vibration a shake table test was performed on an ERV assembly. The vibration testing was performed using IEEE 383 as a guide. The vibration limit for an ERV is a function the combination of vibration amplitude, frequency and mode. Testing revealed that the QC1 ERV damage occurred due to excitation of the pilot plunger (actuator internal) at 157 Hz. The highest amplitude measured for the 3B and 3E values was 0.39 g rms, at the pre-EPU power level (2488 MWt). Post-EPU, the highest amplitude measured for the 3B and 3E values is 1.02 g rms.	Yes
Operating margin: Does the changed parameter reduce the operating margin for a component to an unacceptable level? Does it require that standby equipment now operate during normal conditions such that the margin is reduced to an unacceptable level?	No	No
Material condition: Does the changed parameter have an adverse effect on the material condition of a component such that the performance of the component is significantly affected or the margin for degradation is reduced to an unacceptable level? Consider whether the material condition was degraded pre-EPU and whether EPU could aggravate this degradation unacceptably.	No	No
Load and load combinations: Does the changed parameter result in loads on a component (e.g., thermal or dynamic) such that the margin to failure is reduced to an unacceptable level?	No	No
Environmental conditions and environmental qualification: Does the changed parameter have an adverse effect on a component relative to EQ, for example, due to changes in pressure, temperature, humidity, corrosiveness, EMI or nuclear radiation (including duration exposure)?	No	No
Interface considerations: Does the changed parameter have a potentially adverse effect on an interfacing system (including functional and physical interfaces)? [If "Yes," this information must be communicated to the EPU review team reviewing the affected system.]	No	No
Mechanical/structural considerations: Does the changed parameter have an adverse effect on component vibration, shock, reaction forces, equipment foundations or pipe supports?	Yes, component vibration increases.	Yes

Attribute	Evaluation*	Action?
Hydraulic considerations: Does the changed parameter have an adverse effect on pump net positive suction head, allowable pressure drop, allowable fluid velocity, fluid state, margin to flashing or steam quality?	No	No
Electrical considerations: Does the changed parameter have an adverse effect on component power, voltage, raceway requirements, fault current, electrical insulation or motor requirements? Does the changed parameter adversely affect interfacing electrical systems (e.g., motor control center)?	No	No
Instrumentation and control considerations: Does the changed parameter affect component setpoints, instrument ranges or calibrations?	No	No
Reliability and service life considerations: Does the changed parameter result in additional component cycles, changes in maintenance requirements (e.g., online maintenance changed to outage maintenance) or increased wear? Is the current PM frequency adequate?	No	No

* This evaluation should consider previous evaluations such that vulnerabilities that have already been identified and dispositioned (as verified by the review team) are not identified as vulnerabilities on this worksheet.

Extent of Condition - Evaluation Criteria

Vulnerability Review

System: Main Steam
Parameter(s): Main Steam System Vibration
Component(s): MSIVs: Inboard and Outboard Valve Assemblies
Subcomponent(s): Valve body, Actuator, Limit Switches, Solenoid Valves

Instructions: The purpose of this form is to assess potential vulnerabilities related to a changed system parameter or parameters. For each attribute, the associated questions should be evaluated relative to the effect of the changed parameter on a system component or components. For the system level evaluation, if "Yes" is indicated in the "Action?" column, a component level evaluation is required for that changed parameter. For the component level evaluation, if "Yes" is indicated, the potential vulnerability must be evaluated for potential failure mechanisms and a risk ranking must be assigned.

Attribute	Evaluation	Action?
Component ratings: Does the changed parameter exceed a component rating or design value, e.g., with regard to capacity, duty, pressure, flow, temperature, fluid state, fluid chemistry or voltage?	Plant data indicates that the MSIV actuator vibration is below the qualification testing value with the exception of the 1C MSIV at Quad Cities. Vibration projections for the 1C MSIV were based on the 3C ERV inlet flange vibration data. No evidence of increased wear was observed during equipment walkdown. However at this vibration level, there can be long term degradation of the valve internals. The higher steam flow rate can cause agitation of the internal parts causing accelerated fretting corrosion wear in comparison with pre-EPU conditions (refer to SIL 568).	Yes
Operating margin: Does the changed parameter reduce the operating margin for a component to an unacceptable level? Does it require that standby equipment now operate during normal conditions such that the margin is reduced to an unacceptable level?	No	No
Material condition: Does the changed parameter have an adverse effect on the material condition of a component such that the performance of the component is significantly affected or the margin for degradation is reduced to an unacceptable level? Consider whether the material condition was degraded pre-EPU and whether EPU could aggravate this degradation unacceptably.	Yes. See response to Component Rating.	Yes
Load and load combinations: Does the changed parameter result in loads on a component (e.g., thermal or dynamic) such that the margin to failure is reduced to an unacceptable level?	No	No
Environmental conditions and environmental qualification: Does the changed parameter have an adverse effect on a component relative to EQ, for example, due to changes in pressure, temperature, humidity, corrosiveness, EMI or nuclear radiation (including duration exposure)?	No	No
Interface considerations: Does the changed parameter have a potentially adverse effect on an interfacing system (including functional and physical interfaces)? [If "Yes," this information must be communicated to the EPU review team reviewing the affected system.]	No, component vibration does not impact any interfacing system.	No
Mechanical/structural considerations: Does the changed parameter have an adverse effect on component vibration, shock, reaction forces, equipment foundations or pipe supports?	Yes, high component vibration.	Yes
Hydraulic considerations: Does the changed parameter have an adverse effect on pump net positive suction head, allowable pressure drop, allowable fluid velocity, fluid state, margin to flashing or steam quality?	No	No
Electrical considerations: Does the changed parameter have an adverse effect on component power, voltage, raceway requirements, fault current, electrical insulation or motor requirements? Does the changed parameter adversely affect interfacing electrical systems (e.g., motor control center)?	No	No
Instrumentation and control considerations: Does the changed parameter affect component setpoints, instrument ranges or calibrations?	No. MSIV limit switches see a higher vibration level, but are seismically qualified to a much higher level.	No

Attribute	Evaluation*	Action?
Reliability and service life considerations: Does the changed parameter result in additional component cycles, changes in maintenance requirements (e.g., online maintenance changed to outage maintenance) or increased wear? Is the current PM frequency adequate?	No. The MSIVs are not operated at a greater frequency.	No

* This evaluation should consider previous evaluations such that vulnerabilities that have already been identified and dispositioned (as verified by the review team) are not identified as vulnerabilities on this worksheet.

Extent of Condition - Evaluation Criteria

Vulnerability Review

System: Main Steam
Parameter(s): System Vibration
Component(s): Target Rock Safety/Relief Valve
Subcomponent(s): N/A

Instructions: The purpose of this form is to assess potential vulnerabilities related to a changed system parameter or parameters. For each attribute, the associated questions should be evaluated relative to the effect of the changed parameter on a system component or components. For the system level evaluation, if "Yes" is indicated in the "Action?" column, a component level evaluation is required for that changed parameter. For the component level evaluation, if "Yes" is indicated, the potential vulnerability must be evaluated for potential failure mechanisms and a risk ranking must be assigned.

Attribute	Evaluation	Action?
Component ratings: Does the changed parameter exceed a component rating or design value, e.g., with regard to capacity, duty, pressure, flow, temperature, fluid state, fluid chemistry or voltage?	The Target Rook Valves are a 3 stage design. No qualification data exist for these valves because they were qualified through the Seismic Qualification Users Group (SQUG) methodologies that did not address wear aging. Qualification aging of 2-stage TR valves has been successful with wear aging at 0.75 g acceleration. Testing was performed IAW Standard IEEE 382-1980 test value. The 3-stage TR valve is lighter and should display higher natural frequencies. Engineering judgment concludes that there is only a low to medium risk of degradation at EPU conditions. (General Electric Report No. DRF-0000-0023-4260, Dated January 9, 2004). There is some potential for pilot leakage and thread wear on the main piston/stem joint. Pilot valve leakage would be detectable by increased tailpipe temperatures.	Yes
Operating margin: Does the changed parameter reduce the operating margin for a component to an unacceptable level? Does it require that standby equipment now operate during normal conditions such that the margin is reduced to an unacceptable level?	No	No
Material condition: Does the changed parameter have an adverse effect on the material condit ion of a component such that the performance of the component is significantly affected or the margin for degradation is reduced to an unacceptable level? Consider whether the material condition was degraded pre-EPU and whether EPU could aggravate this degradation unacceptably.	No	No
Load and load combinations: Does the changed parameter result in loads on a component (e.g., thermal or dynamic) such that the margin to failure is reduced to an unacceptable level?	No	No
Environmental conditions and environmental qualification: Does the changed parameter have an adverse effect on a component relative to EQ, for example, due to changes in pressure, temperature, humidity, corrosiveness, EMI or nuclear radiation (including duration exposure)?	No	No
Interface considerations: Does the changed parameter have a potentially adverse effect on an interfacing system (including functional and physical interfaces)? [If "Yes," this information must be communicated to the EPU review team reviewing the affected system.]	No	No
Mechanical/structural considerations: Does the changed parameter have an adverse effect on component vibration, shock, reaction forces, equipment foundations or pipe supports?	Yes, vibration levels at Quad Cities are higher	Yes
Hydraulic considerations: Does the changed parameter have an adverse effect on pump net positive suction head, allowable pressure drop, allowable fluid velocity, fluid state, margin to flashing or steam quality?	No.	No
Electrical considerations: Does the changed parameter have an adverse effect on component power, voltage, raceway requirements, fault current, electrical insulation or motor requirements? Does the changed parameter adversely affect interfacing electrical systems (e.g., motor control center)?	No	No

Attribute	Evaluation*	Action?
Instrumentation and control considerations: Does the changed parameter affect component setpoints, instrument ranges or calibrations?	No	No
Reliability and service life considerations: Does the changed parameter result in additional component cycles, changes in maintenance requirements (e.g., online maintenance changed to outage maintenance) or increased wear? Is the current PM frequency adequate?	No	No

* This evaluation should consider previous evaluations such that vulnerabilities that have already been identified and dispositioned (as verified by the review team) are not identified as vulnerabilities on this worksheet.

Extent of Condition - Evaluation Criteria

Vulnerability Review

System: Main Steam System
Parameter(s): Vibration
Component(s): Snubbers
Subcomponent(s): N/A

Instructions: The purpose of this form is to assess potential vulnerabilities related to a changed system parameter or parameters. For each attribute, the associated questions should be evaluated relative to the effect of the changed parameter on a system component or components. For the system level evaluation, if "Yes" is indicated in the "Action?" column, a component level evaluation is required for that changed parameter. For the component level evaluation, if "Yes" is indicated, the potential vulnerability must be evaluated for potential failure mechanisms and a risk ranking must be assigned.

Attribute	Evaluation*	Action?
Component ratings: Does the changed parameter exceed a component rating or design value, e.g., with regard to capacity, duty, pressure, flow, temperature, fluid state, fluid chemistry or voltage?	No	No
Operating margin: Does the changed parameter reduce the operating margin for a component to an unacceptable level? Does it require that standby equipment now operate during normal conditions such that the margin is reduced to an unacceptable level?	No	No
Material condition: Does the changed parameter have an adverse effect on the material condition of a component such that the performance of the component is significantly affected or the margin for degradation is reduced to an unacceptable level? Consider whether the material condition was degraded pre-EPU and whether EPU could aggravate this degradation unacceptably.	No	No
Load and load combinations: Does the changed parameter result in loads on a component (e.g., thermal or dynamic) such that the margin to failure is reduced to an unacceptable level?	No	No
Environmental conditions and environmental qualification: Does the changed parameter have an adverse effect on a component relative to EQ, for example, due to changes in pressure, temperature, humidity, corrosiveness, EMI or nuclear radiation (including duration exposure)?	No	No
Interface considerations: Does the changed parameter have a potentially adverse effect on an interfacing system (including functional and physical interfaces)? [If "Yes," this information must be communicated to the EPU review team reviewing the affected system.]	No	No
Mechanical/structural considerations: Does the changed parameter have an adverse effect on component vibration, shock, reaction forces, equipment foundations or pipe supports?	The Piping Problem Resolution Team (PPRT), lead by Stone and Webster, performed a complete piping analysis for EPU conditions. Several modifications were recommended and implemented. Some mods involved snubbers. No impact to snubber performance from vibration was identified in the PPRT report (SW-DRQC-EPU-ENG01) or Task Report T0316; however, a snubber associated with an ERV and a snubber associated with a MSL had loose components discovered during a system walkdown. It is believed that these issues are related to workmanship versus vibration. Confirmation is needed that snubbers are not adversely impacted by the higher Main Steam vibration.	Yes
Hydraulic considerations: Does the changed parameter have an adverse effect on pump net positive suction head, allowable pressure drop, allowable fluid velocity, fluid state, margin to flashing or steam quality?	No	No

Attribute	Evaluation*	Action?
Electrical considerations: Does the changed parameter have an adverse effect on component power, voltage, raceway requirements, fault current, electrical insulation or motor requirements? Does the changed parameter adversely affect interfacing electrical systems (e.g., motor control center)?	No	No
Instrumentation and control considerations: Does the changed parameter affect component setpoints, instrument ranges or calibrations?	No	No
Reliability and service life considerations: Does the changed parameter result in additional component cycles, changes in maintenance requirements (e.g., online maintenance changed to outage maintenance) or increased wear? Is the current PM frequency adequate?	No	No

* This evaluation should consider previous evaluations such that vulnerabilities that have already been identified and dispositioned (as verified by the review team) are not identified as vulnerabilities on this worksheet.

Extent of Condition - Evaluation Criteria

Vulnerability Review

System: Main Steam System
Parameter(s): Pressure
Component(s): Main Steam Safety Valves
Subcomponent(s): N/A

Instructions: The purpose of this form is to assess potential vulnerabilities related to a changed system parameter or parameters. For each attribute, the associated questions should be evaluated relative to the effect of the changed parameter on a system component or components. For the system level evaluation, if "Yes" is indicated in the "Action?" column, a component level evaluation is required for that changed parameter. For the component level evaluation, if "Yes" is indicated, the potential vulnerability must be evaluated for potential failure mechanisms and a risk ranking must be assigned.

Attribute	Evaluation*	Action?
Component ratings: Does the changed parameter exceed a component rating or design value, e.g., with regard to capacity, duty, pressure, flow, temperature, fluid state, fluid chemistry or voltage?	No	No
Operating margin: Does the changed parameter reduce the operating margin for a component to an unacceptable level? Does it require that standby equipment now operate during normal conditions such that the margin is reduced to an unacceptable level?	Dresden's Tech Specs currently requires 8 of 9 valves to be operable. Dresden has submitted a tech spec change that will require 9 of 9 valves to be operable, similar to QC. The EPU Evaluation indicated that the Reactor over pressurization event would not require all 9 valves. Subsequently, a reload analysis for Dresden required 9 of 9 valves to be operable for overpressure protection durin g a MSIV closure followed by a reactor scram. NRC approval of the tech spec change is pending resolution of the chronic MSS setpoint out-of- tolerance issue, described below. Quad Cities Tech Specs require 9 of 9 valves (includes TR S/RV) to be operable. Setpoint drift often results in the setpoint drifting outside the Tech Spec tolerance of +/- 1%. Test data indicates that a two-sigma factor would require a tolerance of +/- 2%. MSSV setpoint drift out of tolerance is a pre-existing issue. There were some discussions on the MSSV tolerance issue during the EPU project. An analysis to revisit the setpoint tolerance was outside the EPU scope.	Yes
Material condition: Does the changed parameter have an adverse effect on the material condition of a component such that the performance of the component is significantly affected or the margin for degradation is reduced to an unacceptable level? Consider whether the material condition was degraded pre-EPU and whether EPU could aggravate this degradation unacceptably.	No	No
Load and load combinations: Does the changed parameter result in loads on a component (e.g., thermal or dynamic) such that the margin to failure is reduced to an unacceptable level?	No	No
Environmental conditions and environmental qualificat ion: Does the changed parameter have an adverse effect on a component relative to EQ, for example, due to changes in pressure, temperature, humidity, corrosiveness, EMI or nuclear radiation (including duration exposure)?	No	No
Interface considerations: Does the changed parameter have a potentially adverse effect on an interfacing system (including functional and physical interfaces)? [If "Yes," this information must be communicated to the EPU review team reviewing the affected system.]	No	No
Mechanical/structural considerations: Does the changed parameter have an adverse effect on component vibration, shock, reaction forces, equipment foundations or pipe supports?	No	Yes

Attribute	Evaluation*	Action?
Hydraulic considerations: Does the changed parameter have an adverse effect on pump net positive suction head, allowable pressure drop, allowable fluid velocity, fluid state, margin to flashing or steam quality?	No	No
Electrical considerations: Does the changed parameter have an adverse effect on component power, voltage, raceway requirements, fault current, electrical insulation or motor requirements? Does the changed parameter adversely affect interfacing electrical systems (e.g., motor control center)?	No	No
Instrumentation and control considerations: Does the changed parameter affect component setpoints, instrument ranges or calibrations?	No	No
Reliability and service life considerations: Does the changed parameter result in additional component cycles, changes in maintenance requirements (e.g., online maintenance changed to outage maintenance) or increased wear? Is the current PM frequency adequate?	No	No

* This evaluation should consider previous evaluations such that vulnerabilities that have already been identified and dispositioned (as verified by the review team) are not identified as vulnerabilities on this worksheet.

Extent of Condition - Evaluation Criteria

Vulnerability Review

System: Main Steam System
Parameter(s): Pressure
Component(s): Main Steam Line Drain Piping
Subcomponent(s): N/A

Instructions: The purpose of this form is to assess potential vulnerabilities related to a changed system parameter or parameters. For each attribute, the associated questions should be evaluated relative to the effect of the changed parameter on a system component or components. For the system level evaluation, if "Yes" is indicated in the "Action?" column, a component level evaluation is required for that changed parameter. For the component level evaluation, if "Yes" is indicated, the potential vulnerability must be evaluated for potential failure mechanisms and a risk ranking must be assigned.

Attribute	Evaluation	Action?
Component ratings: Does the changed parameter exceed a component rating or design value, e.g., with regard to capacity, duty, pressure, flow, temperature, fluid state, fluid chemistry or voltage?	No	No
Operating margin: Does the changed parameter reduce the operating margin for a component to an unacceptable level? Does it require that standby equipment now operate during normal conditions such that the margin is reduced to an unacceptable level?	No	No
Material condition: Does the changed parameter have an adverse effect on the material condition of a component such that the performance of the component is significantly affected or the margin for degradation is reduced to an unacceptable level? Consider whether the material condition was degraded pre-EPU and whether EPU could aggravate this degradation unacceptably.	No	No
Load and load combinations: Does the changed parameter result in loads on a component (e.g., thermal or dynamic) such that the margin to failure is reduced to an unacceptable level?	No	No
Environmental conditions and environmental qualification: Does the changed parameter have an adverse effect on a component relative to EQ, for example, due to changes in pressure, temperature, humidity, corrosiveness, EMI or nuclear radiation (including duration exposure)?	No	No
Interface considerations: Does the changed parameter have a potentially adverse effect on an interfacing system (including functional and physical interfaces)? [If "Yes," this information must be communicated to the EPU review team reviewing the affected system.]	No	No
Mechanical/structural considerations: Does the changed parameter have an adverse effect on component vibration, shock, reaction forces, equipment foundations or pipe supports?	The Main Steam Drain Line piping calculation results showed that the pipe stresses due to vibration are cut in half with tieback supports installed. Piping stresses are acceptable, provided tieback supports are properly installed.	Yes
Hydraulic considerations: Does the changed parameter have an adverse effect on pump net positive suction head, allowable pressure drop, allowable fluid velocity, fluid state, margin to flashing or steam quality?	No	No
Electrical considerations: Does the changed parameter have an adverse effect on component power, voltage, raceway requirements, fault current, electrical insulation or motor requirements? Does the changed parameter adversely affect interfacing electrical systems (e.g., motor control center)?	No	No
Instrumentation and control considerations: Does the changed parameter affect component setpoints, instrument ranges or calibrations?	No	No
Reliability and service life considerations: Does the changed parameter result in additional component cycles, changes in maintenance requirements (e.g., online maintenance changed to outage maintenance) or increased wear? Is the current PM frequency adequate?	No	No

* This evaluation should consider previous evaluatio ns such that vulnerabilities that have already been identified and dispositioned (as verified by the review team) are not identified as vulnerabilities on this worksheet.

7. Interview Results (Project Instruction Attachments 2 & 3)

EPU EOC Interviews – Main Steam Quad Cities

Date: 2/9/04

Interviewer(s): Bob Geier, Paul Doverspike, John Freeman, Terry McIntyre, Gary Sozzi

Participants	Title
Rick Swart	QC Main Steam System Engineer
Jim Trettin	QC System Engineering Lead
Bill Poppe	Dresden Main Steam System Engineer
John Boseman	GE Relief Valve Expert
Clyde Nieh	GE MSIV Expert
Joe Welch (on 2/17/04)	Dresden Operations
Dale Eaman	Dresden I&C Engineer
Joe Taft	Quad Cities I&C Engineer

Topic(s): Dresden and Quad Cities Main Steam System

The purpose of this interview is to identify potential EPU-related vulnerabilities that could result in one or more of the following:

- License Event Report
- Engineered Safety Features Actuation
- Reactor Scram
- Plant power derate
- Unplanned entry into Technical Specification
- Operator "Work Around" that increases the risk to one of the above events.
- Unexpected accelerated equipment that increases the risk to one of the above events.

EPU EOC Interviews – Main Steam Quad Cities

Systems:		
	Question	**Response**
1	How has system operation changed following the EPU? What parameters have changed (pressure, temperature, flow, fluid state, moisture content)?	**Quad:** Steam flow rate has increased to approx 220 ft/sec. The amplitude of existing vibrations appears to have increased based on equipment inspections. Both high frequency (steam dryer) and low frequency (steam dryer, HPCI LLRT Tap) vibrations have either been created, or have grown in amplitude based on equipment inspections. Pressures have increased slightly at the reactor and main steam lines near the reactor, and pressures have decreased slightly at the turbine end of the main steam lines. The amount of steam produced has increased, resulting in higher steam amounts to be relieved during certain accident scenarios supported by main steam ADS valves; this results in all ADS valves being required operable, where one valve could previously be inoperable. **Dresden:** Steam velocity has increased. Head loss has increased. MSL pressure gradient has increased (i.e., lower pressure at the turbine control valves). No change in system temperature. Moisture content decreased. No indication of an increase in system vibration. **Welch:** U2 is at BOC; U3 is at midcycle. No vib issues noted for Main Steam system. Moisture Carryover in general decreased with power uprate (result of dryer perforated plate modification). Moisture carryover spikes (increases from ~ 0.03 to ~ 0.06 following load drops or control rod pattern shifts. Flow/Steam velocity increased with EPU. Throttle pressure decreased. Potential for a group 1 isolation: During a rapid decrease in power, unless the pressure set setpoint is adjusted, a Group 1 isolation may occur on low MSL pressure. The time delay for this signal along with the higher setpoint sever to increase setpoint margin. Digital EHC would further sepoint margin. Quad is the lead plant for the EHC mod. Currently, the mod is on hold for budget reasons.
2	Is system operation since the EPU as expected? Have there been any unexpected problems, disappointments, concerns or issues?	**Quad:** Operation has not been as expected. Unexpected Main Steam problems include, but may not be limited to: Relay chatter associated with MSL Lo Lo Pressure Instruments, damage to MSL Drain lines, accelerated wear to ERV Solenoid actuators, HPCI LLRT tap vibrations, floor vibrations in area of HPCI Fluid Head and pipe hanger and associated equipment vibrations, MSSV's no longer analyzed for 3% tolerance allowed by OM Code (note: this was as expected), drain line tie back degradation. Other issues believed to be associated with vibrations include limit switch degradation on MSIVs and HPCI valves, and loosened fasteners on pipe hangers.

EPU EOC Interviews – Main Steam Quad Cities

Systems:		
	Question	**Response**
		Dresden: No evidence of the vibration related equipment degradation experienced at Quad. **MSIVs:** D/Q MSIVs were manufactured by Crane. Only two other plants (Millstone and Tsuruga) have Crane valves. The initial valve design resulted in Belleville Springs in the flow stream. The Belleville springs wear away in this location. A mod is planned to replace these springs with wave springs. This is a pre-EPU condition that was present at both Dresden and Quad. The wear rate may have increased with EPU. MSIV limit switches are hard to set (not EPU related). Limit switch closest to valve body is affected by excessive heat (not EPU related). Namco is the only qualified limit switch. Dresden's PM performs a surveillance ever RO and replaces switch, as necessary based on surveillance results. Quad replaces switch every other RO. Quad PM frequency may need to be revisited. There is an industry effort to qualify a more robust switch. Dresden MSIV LLRT failure rate appears to be higher post-EPU. MSIVs are cycled less often for surveillance testing. **MSS Valves:** Quad has not inspected MSS Valve internals since EPU implementation. No external evidence of degradation. Reduced margin: All 8 MSS plus the Target Rock SV are required to be operable and lift setpoint tolerance is 1% vs. 3%. (This was a known result of EPU) Exelon has expressed an interest in a TRAC G analysis to justify a 1% tolerance No vibrations issues evident for the Q2 PORVs. Q2 PORVs will be replaced next month with an ERV designed with a single tie-back. Lab testing is currently in progress to assess the ERV's response to vibratory loads experienced at QC. Q1 ERV failure occurred after 11 months of operation. Dresden has tested the lift setpoint for four MSS valves post EPU. All valves tested within +/- 1%. Vib issues with Dresden's ERVs are not evident. Over pressure report indicates a 1 psi margin to 1500 psi ATWS limit. (Freeman please clarify this statement) Dresden has seen a higher noise level on Main Steam Flow signal. The frequency content of this noise may provide some insight into steam line pressure acoustics and the relationship between driving and response frequencies. **Welch:** There is a 30 psi buzz on the sensing line for the MSL Low Pressure Switch. The sensing line is not designed well.
3	Was system operation post-EPU affected by	**Quad:** To be determined. Vibrations, which differ significantly between

Systems:		
	Question	**Response**
	changes in the performance of other systems?	Quad and Dresden, may be associated with the EPU steam flows at the reactor end or the D-Ring end of the main steam system. Steam dryer flow changes may be a contributor, or the tuning of the main steam lines themselves. Turbine control valves operating position has changed. Valve dp may have changed.
4	What components were affected by the EPU—either directly by changes in system parameters or indirectly by changes in interfacing systems?	**Quad:** MSL Vibrations, MSL supports, MSL Drains, ERV Solenoid Actuators, MSL Lo Lo setpoints, MSSV lift set point tolerance, and quantity of Relief Valves required Operable. MSIVs, ERVs, MSS, Flow Element.
5	What modifications were made to the system to support operation at the EPU conditions? • What failure modes or vulnerabilities were addressed by the modifications? • Did the modifications (as implemented) create any new failure modes or vulnerabilities? • Were you satisfied with the modification as implemented? • Did the modifications (as implemented) adequately address the existing material condition issues? • Has system performance post-modification been satisfactory	**Quad:** MSL Support changes, ADS requirements for accident response (no valves can be OOS), MSL drain tiebacks later installed. New failures and vulnerabilities appear to be due to increases in amplitude of vibrations. Quad appears to have had high vibrations pre-EPU. EPU has exasperated the problem Not satisfied with mod as implemented. The EPU evaluation did not maintain the 3% setpoint tolerance for the MSS values. This could lead to LERs if valves lift at greater than 1%, which is expected on about 30% of valves tested. There were existing material condition issues with the ERVs that have been amplified by the mod. It is not clear that the original mod review took into consideration the vibration limits of the ERVs. System performance has been degraded post-maintenance, including Functional Failures (ERVs), TMODS (HPCI LLRT Tap), Load Drops and shutdowns (DC Grounds, Steam Dryers, Steam Leaks, etc.) and various other failures on MSL's on both units. MSL flow switches span widened. Main Steam Line low pressure instrument oscillations not anticipated. Resulted in Group 1 half isolations. **Eaman:** The MSL High Flow switches were changed out with a Barton that has a wider

Systems:		
	Question	Response
		range (0 –400 psi vs 0- 200 psi). The replacement PM for these switches was every other RO (3 years). Dresden is currently has 2 year cycles. Three switches have been replaced after initial installation: One D3 switch was removed after one cycle (24 months). The switch will be evaluated for wear to determine the appropriate PM frequency. One D3 switch would not accept a setpoint adjustment and was replaced after 1 year of operation. One D2 switch after 2 years of operation no longer tracked with power changes, was declared inoperable and was replaced. All three switches were sent to Power Labs for evaluation. **Taft:** Quad Cities moved the calibration frequency from Quarterly to Monthly to address drift issues. This change was made 11/2003. QC has experienced no switch failures. Current plans include replace of the switches with Rosemount DP transmitters and electronic trip units. Eaman: The MSL Low Pressure Switch Setpoint did not have sufficient margin because of the increased head loss and resulting lower turbine throttle pressure. SIL 130 discusses this issue. Chatter resulted and ½ group 1 isolation signals were received during turbine valve testing.

EPU EOC Interviews – Main Steam Quad Cities

Procedures:		
	Question	**Response**
1	Were operating procedures changed to support operation at EPU conditions? • What failure modes or vulnerabilities were addressed by the procedure changes? • Did the procedure changes create any new failure modes, vulnerabilities or operator challenges? • Were you satisfied with the procedure changes as implemented? • Has system operation with the revised procedures been satisfactory?	**Quad:** QCOA 0250 series and QCOS 0250 series procedures were changed. No issues identified. Pressure setpoint change. MSL flow instrument cal procedure change. Welch: EHC and MSL Low pressure setpoint changes.
2	Were PM procedures for systems, components and subcomponents revised to reflect, with regards to frequency and scope, accelerated wear and degradation? • Which PMs were revised? • What was the basis for the changes? • Are there additional PM changes necessary to ensure system and component reliability at the EPU conditions?	**Quad:** PMs specific to EPU have not been reviewed. No PM issues exist at this point. The PMs driven by Tech Specs and Code are very conservative as compared to PCM templates or EPU. Currently, many PMs are performed as often as possible (once per cycle at refuel outage). PM for ERV currently replace all 4 actuators and pilots and 2 value bodies per RO.

EPU EOC Interviews – Main Steam Quad Cities

Material Condition:		
	Question	**Response**
1	What are the material condition issues associated with the components in the system?	**Quad:** Electromatic Relief Valve Solenoid Actuator wear is a known issue. Maintenance history indicates that wear is prevalent, and the actuators are therefore fully refurbished once per fuel cycle. This appears to be a Quad Cities issue. Across the industry, there are other examples of extreme vibrations (Nine Mile Point, have taken pilot off of ERV and mounted it to a stanchion to reduce vibration impacts) and minimal vibrations (Dresden, 690 days at EPU and actuators gave no signs of wear). Overall Main Steam System vibrations have been an issue historically.
2	Does operation at EPU conditions exacerbate any of these material condition issues?	**Quad:** On the ERVs, the standard amount of solenoid actuator wear seem to have been substantially accelerated by EPU level operation. Overall Main Steam System vibrations seem to have increased in amplitude or shifted in frequency, such that damage has been seen at drain lines, LLRT taps, tie backs, and miscellaneous attached equipment. Welch: No vibration issues. "The plant is holding together."
3	Was accelerated degradation of components with material condition issues adequately addressed by the EPU Project?	**Quad:** Unknown. The EPU modification was installed prior to my assignment as system manager, and only specific aspects of the project have been reviewed.

OPEX: Operating information for the fleet was provided by Boseman and Nieh comments.

System Boundary: The system boundary includes all Main Steam piping and components from the vessel nozzle up to the Main Turbine Stop Valves. The Main Steam Stop Valves and the Bypass Valves are not included. Major components included are the MSIVs, MSSVs, MSRVs, flow venturis, drain lines to the Main Condenser and RBEDT; and safety and relief valve discharge lines.

8. Database Report (Project Instruction Attachment 1)

Attached are two database reports. The first report contains the Master Components with recommendations and comments. The second report list the Master Components and the Grouped Components.

9. Miscellaneous Documents

Two tables are attached:

- The first table contains recommendations that were provided to Quad Cities for the current Refueling outage (Q2R17). These recommendations were developed at a mid-point in the system evaluation process. These recommendations should be revisited at the completion of Q2R17.

- The set of Vibration Task Force recommendation currently on record, are included in the second table. The Task Force's recommendations are under review and are expected to change. The recommendations were originally designed to address Exelon's intent to run QC Unit 1 at 100% CLTP for 3500 hours, with an intermediate limitation of 1600 hours.

Attachment 1: Main Steam System Recommendations for Q2R17

Main Steam System Recommendations for Q2R17			
	Recommendations for Q2R17	Basis	Risk
1	Develop a contingency plan for replacement of the PORVs.	QC1 ERV experienced excessive wear due to vibration. QC2 PORVs are scheduled to be replaced with ERVs this outage. ERV qualification testing is in progress and due to be completed by 2/13/04.	In the event that the ERVs don't pass qualification testing, PORVs will need to be used for the next cycle. At present, 2 PORVs are tested and ready for installation. Tech Specs require steam testing OR online testing of all 4 valves. As the station does not want to perform on line testing, steam testing will be required. 2 additional valves will need to be refurbished, or 2 valves will need to be pulled, steam tested, and reinstalled. The QC Main Steam System Engineer is presently working on a contingency plan.
2	Develop contingency plan for MSIV refurbishments.	During the October '03 D2 refuel outage, 6 of 8 MSIV unexpectedly failed their LLRT. This was the first MSIV failure in recent history. An EPU connection is unknown at this point.	MSIV refurbishment is a major scope of work that could impact outage resources and schedule. MSIV Contingency plans are in place for Q2R17.
3	Perform an analysis of the MSSV setpoint tolerance. The analysis should consider the setpoint draft data and expand the current +/- 1% tolerance, if possible.	MSSV setpoint drift is a NRC concern documented in CRs 200174 (Dresden) and 200772 (Quad).	An Operational Evaluation is in progress to address the question with MSSV setpoint drift. The recommended analysis is currently in progress.
4	Perform a Main Steam System walkdown and inspect piping and system components for excessive wear or vibration related degradation. Instrument QC2 for vibration monitoring	Measurements on QC1 indicate vibration levels are an order of magnitude greater than D3. The QC dryers and QC1 ERVs have experienced significant degradation due to vibratory loads.	Unexpected main steam component and subcomponent degradation. Currently, a system walk down and installation of vib instrumentation is included in the outage workscope.

Main Steam System Recommendations for Q2R17			
	Recommendations for Q2R17	Basis	Risk
	similar to QC1.		
5	Ensure that the MSSV, ERV (including actuators) and PORV testing and refurbish procedure adequately documents 'as found' wear conditions.	Understanding 'as found' conditions provides a bases for the estimated PM frequency.	Valuable information needed to understand valve degradation may be lost.
6	Ensure that the EPRI 2:1 weld repair techniques are implemented for the Main Steam System small bore piping.	The implementation of the EPRI 2:1 weld repair techniques significantly reduces the susceptibility of small bore piping to high cycle fatigue failure.	Repeat high cycle fatigue failures. (Quad Cities incorporated the EPRI guidelines following small bore piping failures in 2002.)
7	Remove and inspect an EHC pressure switch. mounted on the Turbine Control Valves. The inspection results will determine the need for follow-on actions. Additionally, a walk down of the EHC piping and tubing is recommended.	These switches are mounted on the Turbine Control Valves and may be subject to excessive vibratory loads. EHC piping and tubing has experienced past vibration related failures.	The wear rate under EPU conditions is unknown for these switches. A failure could result in a plant shutdown.
8	Remove and inspect one of the MSL Low Pressure switches.	Although the MSL Low Pressure Switches are located on an instrument rack several feet from the MSLs and it is expected that they are not significantly influenced by MSL vibration, they have experienced 'chatter'. The 'chatter' was the result of insufficient setpoint margin. QC had only one occurrence of 'chatter'. Dresden experienced more extensive 'chatter'.	A switch failure could result in a Group 1 isolation followed by a scram.
9	Remove one of the MSL High Flow switched and inspect for wear related issues.	Quad is scheduled to replace its MSL High Flow Switches with DP transmitters and electronic trip units in the next refueling outage (Q1R18 and Q2R18). Currently, the MSL High Flow Switches are	A switch failure could result in a Group 1 isolation followed by a scram.

Main Steam System Recommendations for Q2R17			
	Recommendations for Q2R17	Basis	Risk
		calibrated monthly. The calibration data indicates that there is no operational issues. Also, Quad has not experienced any switch failures. Nonetheless, it would be prudent to inspect one High Flow Switch as confirmation that there are no unknown risks.	
10	Proactively replace the MSIV Belleville springs with wave springs.	The MSIV Belleville springs are located in the main steam flow stream. In the past, the Belleville springs have deteriorated, broken apart and have been swept downstream. A failed Belleville spring was located at the turbine inlet strainer in the investigation following the Q2 dryer failure. It is possible for a Belleville spring to pass through the strainer and enter the turbine. A modification exists to replace the Belleville springs with a wave spring at an alternate location.	If a Belleville washer breaks free, turbine damage could result.

Attachment 2: Recommendations From Evaluation of Quad Cities Main Steam Line Vibrations At EPU Power Levels

Recommendations From Evaluation of Quad Cities Main Steam Line Vibrations at EPU Power Levels (EC # 346515)	
Prior to Removal of 1600 Hour Limitation	
1	Perform detailed testing on the ERV configuration, with Wyle Labs, to confirm analytical results, project wear rate, and to confirm proposed additional modification adequacy. Testing to be completed prior to installing ERVs in Unit 2. (Assignee: A8064MW-DR, due date: 2/23/04)
2	Complete analytical evaluation of the 3E ERV and comparison of results of all four ERVs to confirm Engineering Judgment determination of acceptable operational period of 3500 hours. (Assignee: A8064MW-DR, due date 2/11/04)
Prior/During Planned Outage After 3500 Hours of Operation	
3	Obtain a vibration data set monthly and after return to full power operation after any down power of greater than 10% and assess for variation/deviation from the analyzed data and any negative impacts. (Assignee: gather data A8426CMO, due date 5/30/04, Analysis assignee: A8064MW-DR, due date 5/30/04)
4	Install accelerometers on the 1C MSIV during the planned shutdown following 3500 hours of operation. Evaluation of data obtained will be used to define the need for the inspection required in Action 16 below. (Assignee: A8426CMO, due date 5/28/04)
5	Install accelerometers on the 3C ERV pilot valve to confirm model accuracy. (Assignee: A8426CMO, due date 5/28/04)
6	Install modification to the ERVs to reduce pilot valve response to vibration. (Assignee: A8452DEM, Due Date 5/30/04)
7	Inspect the components identified during Q1F51 as degraded. These include the ERV actuators, HPCI 4 valve operator and snubber mounting brackets. (Assignee: A8451NESPR, due date 5/30/04)
8	Monitor weekly individual MSL flows to provide any anomalies that indicate MSIV degradation. (A8452RRT, due date Q1R18, 3/22/05)
Prior to Q1R18 (Spring '05)	
9	Obtain a full set of vibration data when the unit is moved to full thermal power of 2957 MWth for the first time. This data will be used to confirm the assumptions made on expected maximum vibration levels. (Assignee: A8426CMO, due date 5/28/04; Analysis action assignee: A8064MW-DR, due date 6/11/04) (This is also an unverified assumption in Reference 4 calculation, ATI 194877-08)
10	Evaluate the current PM scope and frequency for all components included in this evaluation. The expectation is that some components

Recommendations From Evaluation of Quad Cities Main Steam Line Vibrations at EPU Power Levels (EC # 346515)	
	will need increased PM frequencies for rebuilds, refurbishments and/or replacements. (Assignee: A8430NESSC, due date 5/21/04) Consideration should be given to perform a post-EPU baseline inspection during the Q1R18 outage.
11	Perform testing on the NAMCO limit switches using the Quad specific vibration predominate frequencies and amplitudes. This testing to be performed prior to approval of full cycle operation. (Assignee: A8064MW-DR, due date 5/20/04)
12	Perform testing on the Limitorque actuator type and size SMB-2-80 or equivalent using the Quad specific vibration predominate frequencies and amplitudes. This testing to be performed prior to approval of full cycle operation (Assignee: A8064MW-DR, due date 5/20/04)
13	On the HPCI 4 valve, create and utilize an analytical model to evaluate the failure analysis results to determine the vulnerable sub-components and to design an effective modification. Install the modification of the 4-rotor design to remove the degradation vulnerability during Q1R18. Closely trend normal IST test data until that outage to ensure that any deleterious degradation is identified prior to challenging valve functionality. (Assignee: three separate tasks will be assigned, due date 6/1/04)
14	Evaluate the snubber inspection plan to ensure that sufficient MSL snubbers are functionally tested during Q1R18 to ensure that degradation is not occurring. Inspections should include the previously degraded snubbers, 1-66 and 1-71. Adjust inspection plan as necessary. (Assignee: A8451NESPR, due date 6/1/04)
15	Complete comparison of Quad Cities Unit 1 to Dresden Unit 3. This comparison may provide further insights into changes that can be made for Quad Cities to minimize the measured vibration level responses such that accelerated component degradation does not occur. This evaluation includes the following elements: (Assignee: A8064MW-DR, due date 6/25/04) Expansion of the ongoing MSL circuit analyses and/or scale model testing to include the frequency range up to 180 Hz. This may provide insight into the source of the measured 139 and 157 Hz predominate frequencies. Detailed configuration differences in all MS line branch line connections between Dresden and Quad. Detailed evaluation of the ERV configuration installed at Dresden. A comparison of steam flows between Dresden and Quad on MSL and all branch connections during EPU power level operation.
Prior to Q1R18 (Spring '05)	
16	Inspect the 1C MSIV; during the next refuel outage (Q1R18) to confirm that no degradation has occurred. If degradation is found, evaluate the need for expanding inspection to other MSIVs. (Q1R18 – Assignee: A8451NESPR, due date 3/22/05)
17	Perform any additional inspections as determined by evaluations performed in actions 7, 8, 9 & 11. (Assignee: A8450EM, due 3/22/005)
18	Install modification to the HPCI –4 valve rotor. (Assignee: A8452DEE, due date Q1R18, 3/22/05)

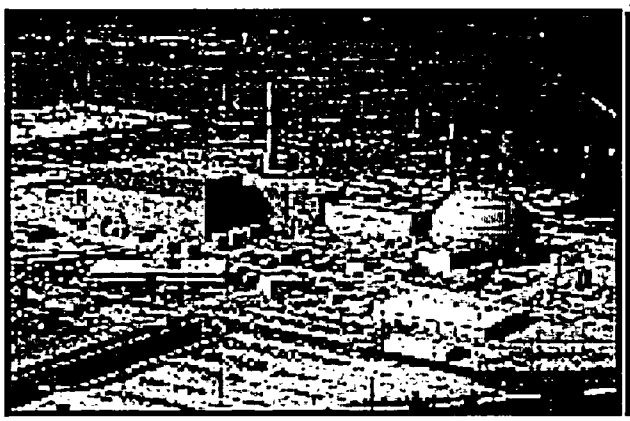
ATTACHMENT 2C

Feedwater System Report

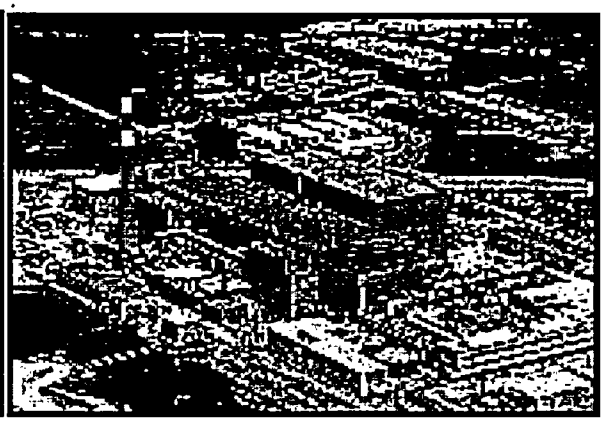
Exelon Corporation
&
GE Nuclear

Extended Power Uprate

Extent of Condition
System & Component Evaluation



Dresden



Quad Cities

Feedwater (FW) System

SYSTEM NAME:	Feedwater (FW) System
STATION(S):	☒ Dresden ☒ Quad Cities
SYSTEM REVIEWERS:	George Paptzun

Table of Contents

1. Executive Summary

Process

The review of the feedwater system was conducted using the guidance in Extended Power Uprate Extent of Condition Review Project Instruction Revision 0.

A total of 426 components within the Dresden Units 2 and 3 feedwater system and 1274 components within the Quad Cities Units 1 and 2 were included in this review. All feedwater system components from the Passport database except those designated, as "run-to-failure" were included. A listing of all components included in the review is provided in Section 9 of this report.

Various documents, including EPU Task Reports, system descriptions, system component lists, previous EPU evaluations, P&IDs and plant data were reviewed to identify potential EPU-related vulnerabilities. Interviews were conducted with the System Engineers at both Dresden and Quad Cities, as documented in Section 7 of this report. Input was also obtained from subject experts in areas such as chemistry control, AOVs, MOVs, pumps, instrumentation, flow accelerated corrosion and vibration.

Changes in System Parameters Resulting from EPU

The changes in system parameters resulting from EPU are summarized in the table in Section 4 of this report.

The most important change in system parameters affecting the feedwater systems at both Dresden and Quad Cities is the 17% increase in feedwater flow rate under normal operating conditions. This increase in feedwater flow results in increased flow induced vibrations, operations without a standby feedwater pump, and operations of the feedwater pumps at a higher discharge pressure than before EPU.

Operational strategy changes to the feedwater systems at both Dresden and Quad Cities were implemented to provide a controllable increase in the feedwater capacity. At both Dresden and Quad Cities the standby feedwater pumps are placed in service to increase to system flowrate by 17%. Use of the standby pumps not only presents a vulnerability in itself by taking away the standby capability; the use of the standby pumps results in a higher discharge pressure based on excess capability of the feedwater pumps.

EPU also resulted in small changes in process and ambient temperatures.

Vulnerabilities Identified

Potential vulnerabilities at both the system level and component level were identified and evaluated as documented in Section 6 of this report. Specific vulnerabilities are documented in

the databse report reproduced in Section 8 of this report and are summarized together with recommendations in Section 2 of this report.

The primary vulnerabilities centers around the following issues:

- Increased vibrations caused by the increased feedwater flow
- Loss of a standby feedwater pump as a backup
- Higher differential pressure around the "D" feedwater heaters resulting in higher differential pressures across the "D" feedwater heater isolation and bypass valves
- Operations of feedwater pumps at a higher pressure on the pump curves putting stress on pump seals and lower flow per pump.

The loss of a standby feedwater pump was recognized as a vulnerability going into EPU. However, other effects of the EPU may not have been considered to the level necessary to avoid future risks.

Recommendations

The most significant concern is the loss of a standby feedwater pump at both Dresden and Quad Cities. Recommendations are made to attempt to restore a standby feedwater pump through plant specific tests and analysis, and to attempt to improve the reliability of the continuously operating feedwater pumps to support the EPU feedwater demands. Other recommendations revolve around the premise that the affects of vibrations are not noticeable until equipment fails from EPU conditions. Therefore, even if vibration measurements are recorded, plotted, and extrapolated, the affects of vibrations on components and sub-components are unknown unless inspections and trending recommendation of components and sub-components are implemented by the stations. Only then will vulnerabilities be reduced to an extent where there will be a high confidence for operation at EPU conditions.

Additionally, recommendations are made to replace components in the piping that may break due to higher feedwater flows. The feedwater sample probes and thermowells addressed in the recommendations to prevent failure of equipment from becoming loose parts in the feedwater system. Potential corrective options are discussed in the Recommendations Table in Section 2.

Additional specific recommendations are summarized in Section 2 of this report, addressing each of the vulnerabilities identified.

References

1. Dresden Station RFP Pump Curves, Byron Jackson Pumps, T-29203-2, T-29227-1, T29226-1, T29447-1, T-29448-1 and 29446-1
2. Quad Cities RFP curves 33069/A, 33069/B, 33069/C, 33070/A, 33070/B, and 33070/C
3. Dresden Spec, GENERAL WORK SPECIFICATION, K-4080
4. Quad Cities Spec R-4411
5. Dresden UFSAR Current Version as of March 8, 2004
6. Quad Cities UFSAR Current Version as of March 8, 2004
7. GE Nuclear EPU Task Report, Dresden/Quad Cities EPU, Task 900, Transient Analysis

8. GE Nuclear EPU Task Report, Dresden/Quad Cities EPU, Task 701, Balance of Plant Power Cycle Thermal Performance
9. GE Nuclear EPU Task Report, Dresden/Quad Cities EPU, Supporting Reference for Task 900, OPL-3, Operating Plant Values for Transient Analysis
10. Dresden Drawing, 469601-1 Rev H DRAG VALVE 18 X18 GLOBE 900 ANSI FEEDWATER REGULATOR
11. GE-NE DRF A61-00052
12. Dresden Drawing, M-1146 Rev G PNEUMATIC DIAGRAM OF REACTOR FEED PUMP MOTOR VENTILATION SYSTEM
13. Dresden Drawing, M-14 Rev LO DIAGRAM OF REACTOR FEED PIPING (CRITICAL CONTROL ROOM DRAWING)
14. Dresden Drawing, M-174 Sheet 3 Rev G REACTOR FEED PUMP OIL PIPING
15. Dresden Drawing, M-292 Rev D REACTOR FEED WATER PUMPS VENTILATION UNIT 2
16. Dresden Drawing, M-310 Sheet 132 Rev 000 SAMPLE PROBE INSTALLATION DETAILS UNIT 2
17. Dresden Drawing, M-347 Rev BU DIAGRAM OF REACTOR FEED PUMP (CRITICAL CONTROL ROOM DRAWING)
18. Dresden Drawing, M-377 Rev P REACTOR FEED PIPING-PLAN
19. Dresden P&ID,
20. Quad Cities Drawing, QC D-11720-0-0040-1-N Rev B MECHANICAL OUTLINE DRAWING SPLIT UNIT ACTUATOR 642A COPES VULCAN VALVE FEEDWATER LEVEL CONTROL
21. Quad Cities Drawing, QC D-11720-0-0690-1-N Rev B MECHANICAL OUTLINE DRAWING HYDRAULIC CYLINDER 5 BORE4 STROKE FEEDWATER LEVEL CONTROL
22. Quad Cities Drawing, M-15 Sheet 1 Rev BK DIAGRAM OF REACTOR FEED PIPING (CRITICAL CONTROL ROOM DRAWING)
23. Quad Cities Drawing, M-15 Sheet 2 Rev C DIAGRAM OF REACTOR FEED PUMP LUBE OIL & SEAL COOLER PIPING (CRITICAL CONTROL ROOM DRAWING)
24. Quad Cities Drawing, M-15 Sheet 3 Rev F DIAGRAM OF FEEDWATER REGULATING VALVE ACTUATOR PIPING (CRITICAL CONTROL ROOM DRAWING)
25. Quad Cities Drawing, M-62 Sheet 1 Rev BL DIAGRAM OF REACTOR FEED PIPING (CRITICAL CONTROL ROOM DRAWING)
26. Quad Cities Drawing, M-62 Sheet 2 Rev D DIAGRAM OF REACTOR FEED PUMP LUBE OIL AND SEAL COOLER PIPING (CRITICAL CONTROL ROOM DRAWING)
27. Quad Cities Drawing, M-62 Sheet 3 Rev F DIAGRAM OF FEEDWATER REGULATING VALVE ACTUATOR PIPING (CRITICAL CONTROL ROOM DRAWING)
28. Memo, Vibes Measurements, From Mroz, Steven D. to Kruthoff, John D.; Schabilion, Steven; Ullrich, John R., Thursday, December 11, 2003
29. Pi data – Average of the periods listed in the table:

Unit	Pre-EPU Summer	Post-EPU Summer	Pre-EPU Winter	Post-EPU Winter
Dresden Unit 2	7/22/99 - 7/28/99	7/22/03 - 7/28/03	12/16/99 - 12/22/99	1/05/04 - 1/11/04
Dresden Unit 3	7/22/00 - 7/28/00	7/22/03 - 7/28/03	1/15/00 - 1/21/00	1/05/04 - 1/11/04
Quad Cities Unit 1	7/17/02 - 7/24/02	8/15/03 - 8/22/03	3/2/02 - 3/9/02	1/15/03 - 1/22/03
Quad Cities Unit 2	7/20/01 - 7/27/01	7/27/02 - 8/3/02	12/30/01 - 1/6/02	3/2/03 - 3/9/03

30. Pre EPU VWO HB, 349HB394
31. Pre EPU Rated HB, 349HB395
32. EPU Rated HB, 552HB1990 R2
33. EPU VWO HB, 552HB1989 R2
34. EPU Part Load HB, 552HB1991 R2

2. Recommendations and Bases (Project Instruction Attachment 5)

Recommendation	Basis	Risk
1. Evaluate if there is some (probably small) potential that full-range testing of pump capability might support operation at 100% EPU power with one of the pumps in standby or might support identification of cost-effective modifications (e.g., change in impeller design) that would allow restoration of the pre-EPU stand-by pump configuration. Perform testing to determine the optimum operating conditions at which to start and stop feedwater pumps for providing optimal power production and running conditions of these pumps. This would be a plant specific analysis..	Per Task Report T0701, when operating with the modified high pressure turbine with only two feed and three condensate / condensate booster pumps on line, the flow passing capability of the feedwater flow control valves will become limiting for operation near 100% of 2527/2511 MWth (pre-EPU 100% power level). While at 110 % of 2527/2511 MWth, the condensate / condensate booster pump motors will exceed their full load amp ratings. In order to achieve maximum operational flexibility a third feed pump should be placed on-line prior to exceeding 100% of 2527/2511 MWth (85.5% LPU), and the fourth condensate/condensate booster pump placed on-line prior to reaching 110% of 2527/2511 MWth (94 % LPU). These above values for placing additional pumps in service at 100% and 110% Pre-EPU power levels are also true if a feedwater or condensate / condensate booster pump is lost or removed from service which is not significantly dependent upon time of season which is based upon theoretically pump crves. Maintenance or loss of condensate, condensate booster or feedwater pumps/motors will result in unit derates during non-outage periods or to avoid derates will have to occur during outage periods. No testing has been performed to verify achievable power with less than full complement of pumps. Per discussion with system engineers, there has been no full-range testing to confirm/identify achievable power with less than the full complement of pumps. There are no spare gear boxes available that would be needed in the event of a failure.	At the minimum the plant would derate to 100% of pre-EPU conditions if a feedwater pump was placed out of service. Maintenance will result in unit derates during non-outage periods or to avoid derates will have to occur during outage periods. This require outage resources. Vulnerability Prioritization: 5 x 10 x 5 = 250 (High)
2. Measure the vibrations of FW pump components.	System engineers and Exelon Vibration experts recognize noticably increased vibration (possibly flow-change or equipment line-up induced) in small-bore pipe and tubing associated with the Feedwater System	At the minimum, the plant would derate to 100% of pre-EPU conditions if a feedwater pump was placed out of service. Vulnerability Prioritization: 5 x 1 x 5 = 25 (Medium)
3. Recommend that the seal PM be increased and a review of three-pump operation be performed in order to prevent a de-rate.	The Feedwater Pumps seal performance has been reviewed in the past due to many failures. The maintenance plan specifies a replacement of the seals every eight years. CMO suggests that the failure of the seals has increased due to the increased discharge pressure due to three pump EPU operation.	At the minimum the plant would derate to 100% of pre-EPU conditions if a feedwater pump was placed out of service. Vulnerability Prioritization:

Recommendation	Basis	Risk
	pump EPU operation.	5 x 5 x 5 = 125 (Medium)
4. Perform external valve/internal operator/motor inspection for wear on one of the 3201 MOVs during next refuel outage.	The effects of vibration and wear at increase EPU flows cannot be easily evaluated on the sub-components of these valves. The likely cause of failure would be due to wear or loosening of fasteners since operation at EPU condition of one cycle has not resulted in any fatigue failures of these valves .The most susceptible sub-components to wear would be the valves body to bonnet fasteners, valve stem or the actuator limit switches, torque switch or rotor. Body to bonnet fasteners have been found to be loose, in the past, due to vibrations. If sub-components are now excited at EPU operating frequencies or vibration amplitudes increased, then potential wear patterns on these sub-components could be detected after one operating cycle and if significant, replaced prior to failure. Visual inspection of the valve stem external surfaces should also be performed If significant wear is detected, replaced worn components prior to failure and adjustment PM cycle to control inspection/replacement frequencies of identified susceptible sub-components.	The Feedwater discharge valves are designed to provide isolation of the feedwater pump for maintenance during plant operations. In the event of an unisolable leak on the discharge check valve the unit may have to, at the extreme conditions, shutdown to repair a leak. The risk associated with a failure of the valve to operate would at least require an operator work around to manually close the MOV in the event the Feedwater Pump needs to be isolated during operations Vulnerability Prioritization: 1 x 5 x 5 = 25 (Medium)
5. Perform valve, actuator and positioner inspection for wear on the AO 3201 during the next refuel outage. Recommend overall of the valve and operator taking particular care to document as found conditions and recommend vibration analysis of the valve, particularly of the Air Operator, attached air piping, and the solenoid valve.	There may be additional vibrations imposed on the valve based on the operating experience gained from the system engineers. It was stated that the vibrations have increased due to three feedwater pumps in operations. Note that the consequences of a failure of the valve should be minimal because the minimum flow valves fail open on a loss of air. This would provide pump protection. The Feedwater System has the capability to makeup for an inadvertent opening of a single minimum flow control valve based on the additional flow available with three feedwater pumps in operation. The CMO identified that these valves are trouble-prone to leakage past their seats. The effects of vibration and wear at EPU conditions cannot be easily evaluated on the sub-components of this valve. The likely cause of failure would be wear since operation at EPU condition for one cycle has not resulted in any fatigue failures of these valves. Note, however, that the minimum flow valve (AO 3201) is normally closed at full power operation; therefore, the number of cycles resulting in fatigue may not have occurred at EPU operating conditions. However, if sub-component are now excited at EPU operating frequencies or vibration amplitudes increased, then potential wear	During a scram or recirculation runback with closure of FWRV's, the feedwater system is near or at deadhead conditions and over pressurizes without the minimum flow valve opening. Pump damage or component degradation may occur if this valve does not open. Vulnerability Prioritization: 1 x 1 x 5 = 5 (Low)

Recommendation	Basis	Risk
	patterns on the valves sub-components could be detected after one operating cycle and if significant replaced prior to failure.· Visual inspection of the valve stem external surfaces should also be performed. If significant wear is detected, replaced worn components prior to failure and adjustment PM cycle to control inspection/replacement frequencies of identified susceptible sub-components.	
6. Review sizing calculation of isolation valves MO 3202 A, B, C and MO 3204 A, B, C at the EPU operating differential pressure conditions.	The feedwater system MOVs consist of heater string isolation MO 3202A, B, C and MO 3204 A, B, C for the inlet and outlet isolation of the HP heaters strings. In addition, a bypass MOV, MO 3203, exists around the LP heater string. In the event of high heater level due to tube rupture, the inlet and outlet isolation valves close and the bypass valve opens to prevent turbine water induction. A higher flow results in a higher differential pressure for valve closure, which may impose additional loads on the motor, operated valves. It is understood that the margins for the motor operator motors are in the range of 200%. With an increased differential pressure across the valves, the primary potential is to affect the stroke time of the valve. The are no assumptions made of these valves in the transient analysis of the plants, however an assumption is made that the feedwater temperature change due to a feedwater heater isolation will take a minute or more. Therefore no affect on the affect on the loss of feedwater heating event and there is no foreseen loss of margin. However, in the event of a heater tube rupture, a failure of the valves to close could result in water backing up into the main turbine. In the event of high heater level due to tube rupture, the inlet and outlet isolation valves close and the bypass valve opens to prevent turbine water induction. ·Typically, motor operated valves are sized based upon design conditions (shutoff conditions that provide the maximum differential pressure at which the valves must open), however; actual operating conditions could have been used in the actuator sizing of MO 3202A, B, C and MO 3204 A, B, C.· EPU has increased the operating differential pressure across the D Feedwater Heaters by as much as 30 psid. Adjustment of the torque switch setting may be required based upon this sizing review..	The Feedwater heater system is designed to close the inlet and outlet isolation valves and open the bypass valve based upon high heater level due to tube rupture to prevent turbine water induction. Water induction into the turbine could cause turbine failure in which a scram would likely occur prior to the water induction if the valves do not operate properly. Vulnerability Prioritization: 10 x 1 x 5 = 50 (Medium)
7. Perform actuator/motor inspection for wear on one	The effects of vibration and wear at increase EPU flows cannot be easily evaluated on the	The Feedwater heater system is designed to close the inlet and

Recommendation	Basis	Risk
of the 3202 MOVs during next refuel outage	sub-components of these valves. The likely cause of failure would be due to wear since operation at EPU condition of one cycle has not resulted in any fatigue failures of these valves. The most susceptible sub-components to wear would be the valves stem or the actuator limit switches, torque switch or rotor. Quad Cities FW system engineer has noted loose components on hangers supporting the condensate booster piping that feeds the LP heaters on both units; perhaps typical of what could be expected from the higher flows through the 3202 valves. For the actuator/motor, the current PM inspects the actuator every 6 refuel cycles. However, if sub-components are now excited at EPU operating frequencies or vibration amplitudes increased, then potential wear patterns on these sub-components could be detected after one operating cycle and if significant, replaced prior to failure. Visual inspection of the valve stem external surfaces should also be performed. If significant wear is detected, replaced worn components prior to failure and adjustment PM cycle to control inspection/replacement frequencies of identified susceptible sub-components.	outlet isolation valves and open the bypass valve based upon high heater level due to tube rupture to prevent turbine water induction. Water induction into the turbine could cause turbine failure in which a scram would likely occur prior to the water induction if the valves do not operate properly Vulnerability Prioritization: 1 x 1 x 5 = 5 (Low)
8. Evaluate the capability of the motor operator to operate at the peak expected differential pressure. Review sizing calculation of bypass valve MO 3203 to ensure adequate margin exists at the EPU operating differential pressure conditions.	A higher flow results in a higher differential pressure for valve closure, which may impose additional loads on the motor, operated valves. It is understood that the margins for the motor operator motors are in the range of 200%. With an increased differential pressure across the MO-3203, D' HEATERS BYP VLV, the primary potential is to affect the stroke time of the valve. The assumptions made of this valve, in the transient analysis of the plants, is that the performance of the valve will be no better than a stroke time of one minute. Therefore no affect on the affect on the loss of feedwater heating event and there is no foreseen loss of margin. However, in the event of a heater tube rupture, a failure of the valve to open, along with failure of the D heater isolation valves to close due the increased differential pressure, could result in water backing up into the main turbine. In the event of high heater level due to tube rupture, the inlet and outlet isolation valves close and the bypass valve opens to prevent turbine water induction. Typically, motor operated valves are sized based upon design conditions (shutoff conditions that provide the maximum differential pressure at which the valves must open), however; actual operating conditions	The Feedwater heater system is designed to close the inlet and outlet isolation valves and open the bypass valve based upon high heater level due to tube rupture to prevent turbine water induction. Water induction into the turbine could cause turbine failure in which a scram would likely occur prior to the water induction if the valves do not operate properly. Vulnerability Prioritization: 10 x 1 x 5 = 50 (Medium)

Recommendation	Basis	Risk
	could have been used in the actuator sizing of MO 3203.· EPU has increased the operating differential pressure across MO 3203 by up to 30 psid. Adjustment of the torque switch setting may be required based upon this sizing review.	
9. Perform actuator/motor inspection for wear on one of the 3206 MOVs during next refuel outage.	The effects of vibration and wear at increase EPU flows cannot be easily evaluated on the sub-components of these valves. The likely cause of failure would be due to wear or loosening of fasteners since operation at EPU condition of one cycle has not resulted in any fatigue failures of these valves .The most susceptible sub-components to wear would be the valves body to bonnet fasteners, valve stem or the actuator limit switches, torque switch or rotor. Body to bonnet fasteners have been found to be loose, in the past, due to vibrations. If sub-components are now excited at EPU operating frequencies or vibration amplitudes increased, then potential wear patterns on these sub-components could be detected after one operating cycle and if significant, replaced prior to failure. Visual inspection of the valve stem external surfaces should also be performed If significant wear is detected, replaced worn components prior to failure and adjustment PM cycle to control inspection/replacement frequencies of identified susceptible sub-components.	The Feedwater Regulation isolation valves are designed to provide isolation of the control valves for out of service conditions. In the event of an unisolable leak of valve, the unit may have to, at the extreme conditions, shutdown to repair a leak Vulnerability Prioritization: 1 x 1 x 5 = 5 (Low)
10. Inspect one discharge check valve at the next outage and perform check valve monitoring or increase the frequency of the PM for the discharge check valve to manage the potential for accelerated wear on the hinge bushings.	Due to the lower flow rate per pump there is potential for accelerated wear on the feedwater pump discharge check valve flapper hinge bushings; due to potential fluctuations in the position of the valve flapper. If this condition persists, then the PM for the discharge check valve should be increased to manage the potential for accelerated wear on the hinge bushings	Avoid the potential of reverse flowthrough the feedwater pumps, when a pump is in standby, which would require operator work around to manage feedwater flow to the reactor. Vulnerability Prioritization: 1 x 1 x 5 = 5 (Low)
11. Revise Specification K-4080 o reflect a new design temperature of 356 F between the D Feedwater heaters and the Reactor.	The original design temperatures of 350 F downstream of the "D" FW heaters have been exceeded at EPU conditions.·The EPU analytic temperature is 352 F in which the actual operating temperatures are just below the analytic limit (356 F).	The non-conservative temperatures in the specification can result in a design error or a design control issue. Vulnerability Prioritization: 10 x 1 x 1 = 10 (Low)
12. Revise Specification K-4080to reflect a new design temperature of 318 F between the Feedwater	The original design temperatures of 300 F downstream of the Feedwater Pump minimum flow valves have been exceeded at EPU conditions	The non-conservative temperatures in the specification can result in a design error or a design control issue

Recommendation	Basis	Risk
Pump minimum flow line and the condenser.	conditions.	design control issue. Vulnerability Prioritization: 10 x 1 x 1 = 10 (Low)
13. For any remaining original probes, replace the feedwater sample probes	In response to several concerns, one of which would be the damage to the Dresden feedwater sparger. The shorter design should be less subject to failure. Dresden had sample probe failures.Three FW probes and one condensate sample probe failed due to high cycle fatigue induced by flow-induced vibration. GE-NE-000-0024-2731 determined ratio of Vortex Frequency for various sample probes but did not include thermowells. GE-NE-000-0024-2731 determined frequency ratios of 1.026, 0.87, 0.86, 0.3, 0.9 and 0.71 for various sample probe cases. Per ASME Appendix N - Below 0.76 acceptable however other documents more restrictive. GE-NE-000-0024-2731 has determined that for certain sample probe cases, the vortex shedding frequency is sufficiently close to a sample probe first natural frequency during various operating conditions that lock-on could occur (i.e. resonant condition). GE-NE-000-0024-2731 did not include thermowells ·Continued successful operation of existing probes or thermowells for an entire operating cycle cannot be assured. If it is determined by analysis that the vortex shedding frequency is sufficiently close to a thermowell's first natural frequency during various operating conditions that lock-on could occur, then the thermowell shall be re-designed.	Probe and thermowell failure could occur and either enter a feedwater heater and become lodged or enter a feedwater pump and possibly cause pump damage. A failure of a thermowell will also likely cause a breach in the system. In both cased, a unit derate or shutdown may be required. Vulnerability Prioritization: 5 x 10 x 5 = 250 (High)
14. Evaluate Feedwater system temperature element thermowells to eliminate concern with vortex shedding failure vulnerability.	Dresden has had sample probe failures. Three FW probes and one condensate sample probe failed due to high cycle fatigue induced by flow-induced vibration.·GE-NE-000-0024-2731 determined ratio of Vortex Frequency for various sample probes but did not include thermowells. GE-NE-000-0024-2731 determined frequency ratios of 1.026, 0.87, 0.86, 0.3, 0.9 and 0.71 for various sample probe cases. Per ASME Appendix N - Below 0.76 acceptable however other documents more restrictive. GE-NE-000-0024-2731 has determined that for certain sample probe cases, the vortex shedding frequency is sufficiently close to a sample probe first natural frequency during various operating conditions that lock-on could occur (i.e. resonant condition). GE-NE-000-0024-2731 did not include	Probe and thermowell failure could occur and either enter a feedwater heater and enter a feedwater sparger and possibly cause damage. A failure of a thermowell will also likely cause a breach in the system. In both cased, a unit derate or shutdown may be required. Vulnerability Prioritization: 5 x 10 x 5 = 250 (High)

Recommendation	Basis	Risk
	thermowells . Continued successful operation of existing probes or thermowells for an entire operating cycle cannot be assured. If it is determined by analysis that the vortex shedding frequency is sufficiently close to a thermowell's first natural frequency during various operating conditions that lock-on could occur, then the thermowell shall be re-designed.	
15. Perform actuator/valve inspection for wear on both of the Feedwater Reg valves during next refuel outage.	The effects of vibration and wear at increase EPU flows cannot be easily evaluated on the sub-components of these valves. The likely cause of failure would be due to wear since operation at EPU condition of one cycle has not resulted in any fatigue failures of these valves. The most susceptible sub-components to wear would be the valves stem or the actuator parts. Quad Cities FW system engineer has noted lan increase in vibrations. The current PM inspects the valve and actuator, one each outage. However, if sub-components are now excited at EPU operating frequencies or vibration amplitudes increased, then potential wear patterns on these sub-components could be detected after one operating cycle and if significant, replaced prior to failure. Visual inspection should also be performed and documeneted to determine significant wear is detected. The goal is to replace worn components prior to failure and adjust PM cycles to control inspection/replacement frequencies of identified susceptible sub-components.	At the minimum, failure of a regulating valve would require a de-rate to repair Vulnerability Prioritization: 5 x 5 x 5 = 125 (Medium)

3. Known Vulnerabilities Checklist Review

System Name: Feedwater (FW) System *Instructions: Place a check mark for each unit for which the document was reviewed.*				
Reviewed Documents	**DRE U2**	**DRE U3**	**QDC U1**	**QDC U2**
Pinch Points	☒	☒	☒	☒
Margin List	☒	☒	☒	☒
Focus Review	☒	☒	☒	☒
CHIP	☒	☒	☒	☒
SHIP	☒	☒	☒	☒
Single Point Vulnerability	☒	☒	☒	☒
Scram De-Rate Challenge	☒	☒	☒	☒
Industry Feedback	☐	☐	☐	☐
GE Task Reports	☒	☒	☒	☒
Common Cause Analysis	☒	☒	☒	☒
EPU Evaluation Changes	☒	☒	☒	☒

4. Summary of System Parameter Changes

System parameters with notes are attached for the FW Systems at Dresden and Quad Cities.

EPU Extent of Condition Review

System Parameter with Notes

Dresden

System 32 Feedwater (FW)

PARAMETER Flow - Total FW System (Mlbs/hr)

	Value	Notes/References
Analytical Pre-EPU Level:	9.72	Pre EPU VWO HB, 349HB394
Analytical Post-EPU Level:	11.71	VWO HB Figure 1-2, TR 0701, BOP Power Cycle Thermal Performance
System Design Basis Value	11.71	T0900 OPL-3, Equivalent to Steam Flow
Limiting Pre-EPU Plant Data	9.8	Sum of winter avg (Dres A3085 and A3086) Pi data Average of the periods listed in the FW Reference table
Limiting Post-EPU Plant Data	11.32	M#/HR sum of avg (Dres A3085, A3086, and A3087) Pi data Average of the periods listed in the FW Reference table

PARAMETER Pressure - FW Pmp Discharge (PSIG)

	Value	Notes/References
Analytical Pre-EPU Level:	N/A	N/A
Analytical Post-EPU Level:	1338	1353 psia, TPU HB Figure 1-1, TR 0701, BOP Power Cycle Thermal Performance
System Design Basis Value	1850	Spec. No K-4080
Limiting Pre-EPU Plant Data	1501	Summer avg (Dres A2042) Pi data Average of the periods listed in the FW reference table
Limiting Post-EPU Plant Data	1506	Winter avg (Dres A2042) Pi data Average of the periods listed in the FW reference table

PARAMETER Pressure - FW Pmp Suction (PSIG) Min

	Value	Notes/References
Analytical Pre-EPU Level:	N/A	N/A
Analytical Post-EPU Level:	166.7	181.4 psia VWO HB Figure 1-2 (166.7 psig) ,TR 0701, BOP Power Cycle Thermal Performance
System Design Basis Value	N/A	N/A
Limiting Pre-EPU Plant Data	193	Summer avg (Dres A3041) Pi data Average of the periods listed in the FW reference table.
Limiting Post-EPU Plant Data	175	Summer avg (Dres A2041) 175 psgi summer avg (Dres A3041) Pi data Average of the periods listed in the FW reference table.

PARAMETER Temperature - FW Heater D Discharge (Max/Min) (F)

	Value	Notes/References
Analytical Pre-EPU Level:	340/N/A	Pre EPU VWO HB, 349HB394
Analytical Post-EPU Level:	352/152	Task Report 701, VWO HB, TR 0701, BOP Power Cycle Thermal Performance,
System Design Basis Value	356/152 Transient Analysis 350 Pipe Design	356F LPU Basis from T0900; 355.6F at RLA from OPL-3. CHANGE IN FEEDWATER TEMPERATURE (MAXIMUM) (WORST SINGLE FAILURE OF FW HEATERS) basis from T0900 RLA from OPL-3. 350F from Spec K-4080
Limiting Pre-EPU Plant Data	343/322	Max - 343F, (DR A2170) winterMin 322F, (DR A2170) summer Pi data, Average of the periods listed in the FW Reference table
Limiting Post-EPU Plant Data	351/346	Max- 351F, (DR A2170) summer and winter Min - 346 F (DR A3170) winter Pi data Average of the periods listed in the FW Reference table

EPU Extent of Condition Review

System Parameter with Notes

Dresden

System 32 Feedwater (FW)

PARAMETER Temperature - FW Pmp Suction (F)

	Value	Notes/References
Analytical Pre-EPU Level:	300	Pre EPU Rated HB, 349HB395
Analytical Post-EPU Level:	318	Task Report 701, VWO HB, TR 0701, BOP Power Cycle Thermal Performance
System Design Basis Value	N/A	N/A
Limiting Pre-EPU Plant Data	307	DR-2) (A2192) Pi data Average of the periods listed in the FW reference table
Limiting Post-EPU Plant Data	315	315F(DR-2) (A2192) Pi data Average of the periods listed in the FW reference table

EPU Extent of Condition Review

System Parameter with Notes

Quad Cities

System FW Feed Water

PARAMETER Flow - Total FW System (Mlbs/hr)

	Value	Notes/References
Analytical Pre-EPU Level:	9.72	Pre EPU VWO HB, 349HB394
Analytical Post-EPU Level:	11.71	VWO HB Figure 1-2, TR 0701, BOP Power Cycle Thermal Performance
System Design Basis Value	11.71	T0900 OPL-3, Equivalent to Steam Flow
Limiting Pre-EPU Plant Data	9.74	Sum of winter avg (Quad A1086 and A1087) Pi data Average of the periods listed in the FW Reference table
Limiting Post-EPU Plant Data	11.57	M#/HR sum of summer avg (Quad A1085, A1086, and A3187) Pi data Average of the periods listed in the FW Reference table

PARAMETER Pressure - FW Pmp Discharge (PSIG)

	Value	Notes/References
Analytical Pre-EPU Level:	N/A	N/A
Analytical Post-EPU Level:	1338	1353 psia, TPU HB Figure 1-1, TR 0701, BOP Power Cycle Thermal Performance
System Design Basis Value	1850	Spec. No. R- 4411
Limiting Pre-EPU Plant Data	1336	Winter avg (Quad A2042) Pi data Average of the periods listed in the FW reference table
Limiting Post-EPU Plant Data	1410	Winter avg (Quad A1042) Pi data Average of the periods listed in the FW reference table

PARAMETER Pressure - FW Pmp Suction (PSIG) Min

	Value	Notes/References
Analytical Pre-EPU Level:	N/A	N/A
Analytical Post-EPU Level:	166.7	181.4 psia VWO HB Figure 1-2 (166.7 psig) ,TR 0701, BOP Power Cycle Thermal Performance
System Design Basis Value	N/A	N/A
Limiting Pre-EPU Plant Data	221	Summer avg (Quad A1041) Pi data Average of the periods listed in the FW reference table.
Limiting Post-EPU Plant Data	204	Summer avg (Quad A1041) Pi data Average of the periods listed in the FW reference table.

PARAMETER Temperature - FW Heater D Discharge (Max/Min) (F)

	Value	Notes/References
Analytical Pre-EPU Level:	340/N/A	Pre EPU VWO HB, 349HB394
Analytical Post-EPU Level:	352/152	Task Report 701, VWO HB, TR 0701, BOP Power Cycle Thermal Performance,
System Design Basis Value	356/152 Transient Analysis, 350 Pipe Design	356F LPU Basis from T0900; 355.6F at RLA from OPL-3. CHANGE IN FEEDWATER TEMPERATURE (MAXIMUM) (WORST SINGLE FAILURE OF FW HEATERS) basis from T0900 RLA from OPL-3. 350F from Spec K-4080
Limiting Pre-EPU Plant Data	339	PI data - average value of A2695
Limiting Post-EPU Plant Data	351	PI data - average value of A2695

EPU Extent of Condition Review

System Parameter with Notes

Quad Cities

System FW Feed Water

PARAMETER Temperature - FW Pmp Suction (F)

	Value	Notes/References
Analytical Pre-EPU Level:	300	Pre EPU Rated HB, 349HB395
Analytical Post-EPU Level:	318	Task Report 701, VWO HB, TR 0701, BOP Power Cycle Thermal Performance
System Design Basis Value	N/A	N/A
Limiting Pre-EPU Plant Data	304	Quad (A1192) Winter Pi data Average of the periods listed in the FW reference table
Limiting Post-EPU Plant Data	313	Quad (A2192) Summer Pi data Average of the periods listed in the FW reference table

5. System Boundaries

The boundaries for the FW systems are as shown on the applicable P&IDs (i.e., M-347 for Dresden and M-15 for Quad Cities).

6. Vulnerability Review (Project Instruction Attachment 4)

6a. System Vulnerability Review

System-level worksheets are attached for the FW systems.

6b. Component Vulnerability Review

System-level worksheets are attached for the FW systems

EXTENT OF CONDITION - EVALUATION CRITERIA VULNERABILITY REVIEW

System: Feedwater System (3200)
Parameter(s): Flow
Component(s): N/A
Subcomponent(s): N/A

Instructions: The purpose of this form is to assess potential vulnerabilities related to a changed system parameter or parameters. For each attribute, the associated questions should be evaluated relative to the effect of the changed parameter on a system component or components. For the system level evaluation, if "Yes" is indicated in the "Action?" column, a component level evaluation is required for that changed parameter. For the component level evaluation, if "Yes" is indicated, the potential vulnerability must be evaluated for potential failure mechanisms and a risk ranking must be assigned.

Attribute	Evaluation	Action?
Component ratings: Does the changed parameter exceed a component rating or design value, e.g., with regard to capacity, duty, pressure, flow, temperature, fluid state, fluid chemistry or voltage?	See Operating Margin Attribute (below)	N/A
Operating margin: Does the changed parameter reduce the operating margin for a component to an unacceptable level? Does it require that standby equipment now operate during normal conditions such that the margin is reduced to an unacceptable level?	Yes - The feedwater flows have increased from pre-EPU VWO conditions of 9,724,966 lbs/hr (3,241,655 lbs/hr per string/ 4,862,483 lbs/hr per pump and FCV), reference 349HB394 to 11,769,380 lbs/hr (3,923,126 lbs/hr per string and pump/ 5,884,690 lbs/hr per FCV) at post-EPU VWO conditions, reference 552HB1989 R2. This is a 21% increase that is outside the original design basis and is a significant change. In addition, four condensate booster pumps and three feedwater pumps are required to be in operation to achieve these EPU flow conditions compared to a 3/2 pump combination prior to EPU. Although the actual post-EPU operating flows are below the analyzed EPU VWO conditions per pump, with this increase in flow and different pump operation that can cause piping and component vibration changes and changes in system pressure losses, this items has to be evaluated on a component level. Therefore, this parameter change cannot be ruled out on a system basis in which known vulnerabilities exists (attached – "Known Feedwater System Vulnerabilities")	Yes, evaluate on component level
Material condition: Does the changed parameter have an adverse effect on the material condition of a component such that the performance of the component is significantly affected or the margin for degradation is reduced to an unacceptable level? Consider whether the material condition was degraded pre-EPU and whether EPU could aggravate this degradation unacceptably.	See Operating Margin Attribute (above)	N/A
Load and load combinations: Does the changed parameter result in loads on a component (e.g., thermal or dynamic) such that the margin to failure is reduced to an unacceptable level?	See Operating Margin Attribute (above)	N/A
Environmental conditions and environmental qualification: Does the changed parameter have an adverse effect on a component relative to EQ, for example, due to changes in pressure, temperature, humidity, corrosiveness, EMI or nuclear radiation (including duration exposure)?	See Operating Margin Attribute (above)	N/A
Interface considerations: Does the changed parameter have a potentially adverse effect on an interfacing system (including functional and physical interfaces)? [If "Yes," this information must be communicated to the EPU review team reviewing the affected system.]	See Operating Margin Attribute (above)	N/A
Mechanical/structural considerations: Does the changed parameter have an adverse effect on component vibration, shock, reaction forces, equipment foundations or pipe supports?	See Operating Margin Attribute (above)	N/A
Hydraulic considerations: Does the changed parameter have an adverse effect on pump net positive suction head, allowable pressure drop, allowable fluid velocity, fluid state, margin to flashing or steam quality?	See Operating Margin Attribute (above)	N/A

Attribute	Evaluation*	Action?
Electrical considerations: Does the changed parameter have an adverse effect on component power, voltage, raceway requirements, fault current, electrical insulation or motor requirements? Does the changed parameter adversely affect interfacing electrical systems (e.g., motor control center)?	See Operating Margin Attribute (above)	N/A
Instrumentation and control considerations: Does the changed parameter affect component setpoints, instrument ranges or calibrations?	See Operating Margin Attribute (above)	N/A
Reliability and service life considerations: Does the changed parameter result in additional component cycles, changes in maintenance requirements (e.g., online maintenance changed to outage maintenance) or increased wear?	See Operating Margin Attribute (above)	N/A

- This evaluation should consider previous evaluations such that vulnerabilities that have already been identified and dispositioned (as verified by the review team) are not identified as vulnerabilities on this worksheet.

EXTENT OF CONDITION - EVALUATION CRITERIA VULNERABILITY REVIEW

System: Feedwater System (3200)
Parameter(s): Pressure
Component(s): N/A
Subcomponent(s): N/A

Instructions: The purpose of this form is to assess potential vulnerabilities related to a changed system parameter or parameters. For each attribute, the associated questions should be evaluated relative to the effect of the changed parameter on a system component or components. For the system level evaluation, if "Yes" is indicated in the "Action?" column, a component level evaluation is required for that changed parameter. For the component level evaluation, if "Yes" is indicated, the potential vulnerability must be evaluated for potential failure mechanisms and a risk ranking must be assigned.

Attribute	Evaluation	Action?
Component ratings: Does the changed parameter exceed a component rating or design value, e.g., with regard to capacity, duty, pressure, flow, temperature, fluid state, fluid chemistry or voltage?	See Operating Margin Attribute (below)	N/A
Operating margin: Does the changed parameter reduce the operating margin for a component to an unacceptable level? Does it require that standby equipment now operate during normal conditions such that the margin is reduced to an unacceptable level?	Yes - The feedwater discharge pressure have increased from pre-EPU VWO conditions of 1283 psig (average of summer and winter pre-epu Pi-data) 1410 psig (average of summer and winter post-epu Pi-data). This is a 10% increase in discharge pressure that is within the original design basis, however this places additional demands on the Feedwater Pumps and Flow Control Valves and is a significant change. In addition, four condensate booster pumps and three feedwater pumps are required to be in operation to achieve these EPU flow conditions compared to a 3/2 pump combination prior to EPU. The actual post-EPU operating flows are below the analyzed EPU VWO conditions per pump resulting in higher head, with this increase in pressure (head driving the additional flow) and changes in system pressure losses, this items has to be evaluated on a component level. Therefore, this parameter change cannot be ruled out on a system basis in which known vulnerabilities exists (attached – "Known Feedwater System Vulnerabilities")	Yes, evaluate on component level
Material condition: Does the changed parameter have an adverse effect on the material condition of a component such that the performance of the component is significantly affected or the margin for degradation is reduced to an unacceptable level? Consider whether the material condition was degraded pre-EPU and whether EPU could aggravate this degradation unacceptably.	See Operating Margin Attribute (above)	N/A
Load and load combinations: Does the changed parameter result in loads on a component (e.g., thermal or dynamic) such that the margin to failure is reduced to an unacceptable level?	See Operating Margin Attribute (above)	N/A
Environmental conditions and environmental qualification: Does the changed parameter have an adverse effect on a component relative to EQ, for example, due to changes in pressure, temperature, humidity, corrosiveness, EMI or nuclear radiation (including duration exposure)?	See Operating Margin Attribute (above)	N/A
Interface considerations: Does the changed parameter have a potentially adverse effect on an interfacing system (including functional and physical interfaces)? [If "Yes," this information must be communicated to the EPU review team reviewing the affected system.]	See Operating Margin Attribute (above)	N/A
Mechanical/structural considerations: Does the changed parameter have an adverse effect on component vibration, shock, reaction forces, equipment foundations or pipe supports?	See Operating Margin Attribute (above)	N/A
Hydraulic considerations: Does the changed parameter have an adverse effect on pump net positive suction head, allowable pressure drop, allowable fluid velocity, fluid state, margin to flashing or steam quality?	See Operating Margin Attribute (above)	N/A

Attribute	Evaluation*	Action?
Electrical considerations: Does the changed parameter have an adverse effect on component power, voltage, raceway requirements, fault current, electrical insulation or motor requirements? Does the changed parameter adversely affect interfacing electrical systems (e.g., motor control center)?	See Operating Margin Attribute (above)	N/A
Instrumentation and control considerations: Does the changed parameter affect component setpoints, instrument ranges or calibrations?	See Operating Margin Attribute (above)	N/A
Reliability and service life considerations: Does the changed parameter result in additional component cycles, changes in maintenance requirements (e.g., online maintenance changed to outage maintenance) or increased wear?	See Operating Margin Attribute (above)	N/A

- This evaluation should consider previous evaluations such that vulnerabilities that have already been identified and dispositioned (as verified by the review team) are not identified as vulnerabilities on this worksheet.

EXTENT OF CONDITION - EVALUATION CRITERIA VULNERABILITY REVIEW

System: Feedwater System (3200)
Parameter(s): Temperature
Component(s): N/A
Subcomponent(s): N/A

Instructions: The purpose of this form is to assess potential vulnerabilities related to a changed system parameter or parameters. For each attribute, the associated questions should be evaluated relative to the effect of the changed parameter on a system component or components. For the system level evaluation, if "Yes" is indicated in the "Action?" column, a component level evaluation is required for that changed parameter. For the component level evaluation, if "Yes" is indicated, the potential vulnerability must be evaluated for potential failure mechanisms and a risk ranking must be assigned.

Attribute	Evaluation*	Action?
Component ratings: Does the changed parameter exceed a component rating or design value, e.g., with regard to capacity, duty, pressure, flow, temperature, fluid state, fluid chemistry or voltage?	Within exitting temperature design specifications	N/A
Operating margin: Does the changed parameter reduce the operating margin for a component to an unacceptable level? Does it require that standby equipment now operate during normal conditions such that the margin is reduced to an unacceptable level?	Minimal impact- The feedwater temperatures have increased from pre-EPU Rated conditions of 302F at the Feedwater Pump Discharge reference 349HB395 to 320F post-EPU VWO HB conditions reference 349HB394. Actual operating data indicates less of a change due to EPU of 8F increase.	N/A
Material condition: Does the changed parameter have an adverse effect on the material condition of a component such that the performance of the component is significantly affected or the margin for degradation is reduced to an unacceptable level? Consider whether the material condition was degraded pre-EPU and whether EPU could aggravate this degradation unacceptably.	No material conditions known that could be affected by the slight temperature increase.	N/A
Load and load combinations: Does the changed parameter result in loads on a component (e.g., thermal or dynamic) such that the margin to failure is reduced to an unacceptable level?	No	N/A
Environmental conditions and environmental qualification: Does the changed parameter have an adverse effect on a component relative to EQ, for example, due to changes in pressure, temperature, humidity, corrosiveness, EMI or nuclear radiation (including duration exposure)?	Minimal change in process temperature – no affect on EQ.	N/A
Interface considerations: Does the changed parameter have a potentially adverse effect on an interfacing system (including functional and physical interfaces)? [If "Yes," this information must be communicated to the EPU review team reviewing the affected system.]	Potentially adverse affect on the NPSHA to the Recir c Pumps due to higher reactor feedwater temperature.	N/A
Mechanical/structural considerations: Does the changed parameter have an adverse effect on component vibration, shock, reaction forces, equipment foundations or pipe supports?	No	N/A
Hydraulic considerations: Does the changed parameter have an adverse effect on pump net positive suction head, allowable pressure drop, allowable fluid velocity, fluid state, margin to flashing or steam quality?	Yes – NPSH for the FW Pumps affected but calculated as acceptable.	N/A
Electrical considerations: Does the changed parameter have an adverse effect on component power, voltage, raceway requirements, fault current, electrical insulation or motor requirements? Does the changed parameter adversely affect interfacing electrical systems (e.g., motor control center)?	No	N/A
Instrumentation and control considerations: Does the changed parameter affect component setpoints, instrument ranges or calibrations?	No	N/A
Reliability and service life considerations: Does the changed parameter result in additional component cycles, changes in maintenance requirements (e.g., online maintenance changed to outage maintenance) or increased wear?	Potential affects to life of pump seals due to an increase in temperature of the feedwater.	N/A

- This evaluation should consider previous evaluations such that vulnerabilities that have already been identified and dispositioned (as verified by the review team) are not identified as vulnerabilities on this worksheet.

EXTENT OF CONDITION - EVALUATION CRITERIA VULNERABILITY REVIEW

System: Feedwater System (3200)
Parameter(s): Flow/Temperature/Pressure
Component(s): Reactor Feed Pump Lube Oil and Seal Coolers Subsystem
Subcomponent(s): N/A

Instructions: The purpose of this form is to assess potential vulnerabilities related to a changed system parameter or parameters. For each attribute, the associated questions should be evaluated relative to the effect of the changed parameter on a system component or components. For the system level evaluation, if "Yes" is indicated in the "Action?" column, a component level evaluation is required for that changed parameter. For the component level evaluation, if "Yes" is indicated, the potential vulnerability must be evaluated for potential failure mechanisms and a risk ranking must be assigned.

Attribute	Evaluation*	Action?
Component ratings: Does the changed parameter exceed a component rating or design value, e.g., with regard to capacity, duty, pressure, flow, temperature, fluid state, fluid chemistry or voltage?	Performance demands should be less due to lower per pump feedwater flow.	N/A
Operating margin: Does the changed parameter reduce the operating margin for a component to an unacceptable level? Does it require that standby equipment now operate during normal conditions such that the margin is reduced to an unacceptable level?	No change in performance. Standby Feedwater Pump and respective Lube Oil System in service.. Should only affect PMs on this system with regards to actions based upon oil sampling results.	N/A
Material condition: Does the changed parameter have an adverse effect on the material condition of a component such that the performance of the component is significantly affected or the margin for degradation is reduced to an unacceptable level? Consider whether the material condition was degraded pre-EPU and whether EPU could aggravate this degradation unacceptably.	No change.	N/A
Load and load combinations: Does the changed parameter result in loads on a component (e.g., thermal or dynamic) such that the margin to failure is reduced to an unacceptable level?	No	N/A
Environmental conditions and environmental qualification: Does the changed parameter have an adverse effect on a component relative to EQ, for example, due to changes in pressure, temperature, humidity, corrosiveness, EMI or nuclear radiation (including duration exposure)?	No change.	N/A
Interface considerations: Does the changed parameter have a potentially adverse effect on an interfacing system (including functional and physical interfaces)? [If "Yes," this information must be communicated to the EPU review team reviewing the affected system.]	None.	N/A
Mechanical/structural considerations: Does the changed parameter have an adverse effect on component vibration, shock, reaction forces, equipment foundations or pipe supports?	No	N/A
Hydraulic considerations: Does the changed parameter have an adverse effect on pump net positive suction head, allowable pressure drop, allowable fluid velocity, fluid state, margin to flashing or steam quality?	No	N/A
Electrical considerations: Does the changed parameter have an adverse effect on component power, voltage, raceway requirements, fault current, electrical insulation or motor requirements? Does the changed parameter adversely affect interfacing electrical systems (e.g., motor control center)?	No	N/A
Instrumentation and control considerations: Does the changed parameter affect component setpoints, instrument ranges or calibrations?	No	N/A
Reliability and service life considerations: Does the changed parameter result in additional component cycles, changes in maintenance requirements (e.g., online maintenance changed to outage maintenance) or increased wear?	Increased wear based on a lack of a standby Feedwater Pump. However, oil sampling should be the key indicator to prevent unplanned system shutdowns..	N/A

- This evaluation should consider previous evaluations such that vulnerabilities that have already been identified and dispositioned (as verified by the review team) are not identified as vulnerabilities on this worksheet.

EXTENT OF CONDITION - EVALUATION CRITERIA VULNERABILITY REVIEW

System: Feedwater System (3200)
Parameter(s): Flow and Pressure
Component(s): AO-3201-A, '(ASSY) 2A RFP MIN FLOW VLV
Subcomponent(s): N/A

Instructions: The purpose of this form is to assess potential vulnerabilities related to a changed system parameter or parameters. For each attribute, the associated questions should be evaluated relative to the effect of the changed parameter on a system component or components. For the system level evaluation, if "Yes" is indicated in the "Action?" column, a component level evaluation is required for that changed parameter. For the component level evaluation, if "Yes" is indicated, the potential vulnerability must be evaluated for potential failure mechanisms and a risk ranking must be assigned.

Attribute	Evaluation	Action?
Component ratings: Does the changed parameter exceed a component rating or design value, e.g., with regard to capacity, duty, pressure, flow, temperature, fluid state, fluid chemistry or voltage?	No change with regards to the setpoint at which the minimum flow valve is required to operate.	N/A
Operating margin: Does the changed parameter reduce the operating margin for a component to an unacceptable level? Does it require that standby equipment now operate during normal conditions such that the margin is reduced to an unacceptable level?	No change with regards to the required flowrate. This valve is not in the process stream. The valve is exposed to additional pressure during operations, however the nature of the operations of this valve is that when the feedwater pump has low flow, which is coincident with a higher pressure (follows the pump curve). No foreseen affect to the operating margin. In addition, even though all feedwater pumps will be operating this does not affect the minimum flow valve, based on one valve in the standby condition, because these valves should only operate during unplanned power maneuvering or low power conditions.	N/A
Material condition: Does the changed parameter have an adverse effect on the material condition of a component such that the performance of the component is significantly affected or the margin for degradation is reduced to an unacceptable level? Consider whether the material condition was degraded pre-EPU and whether EPU could aggravate this degradation unacceptably.	See Operating Margin Attribute (above)	N/A
Load and load combinations: Does the changed parameter result in loads on a component (e.g., thermal or dynamic) such that the margin to failure is reduced to an unacceptable level?	There may be additional vibrations imposed on the valve based on the operating experience gained from the system engineers. It was stated that the vibrations have increased due to three feedwater pumps in operations. Recommend vibration analysis of the valve, particularly of the Air Operator, attached air piping, and the solenoid valve. Note that the consequences of a failure of the valve should be minimal because the minimum flow valves fail open on a loss of air. This would provide pump protection. The Feedwater System has the capability to makeup for an inadvertent opening of a single minimum flow control valve based on the additional flow available with three feedwater pumps in operation.	Yes
Environmental conditions and environmental qualification: Does the changed parameter have an adverse effect on a component relative to EQ, for example, due to changes in pressure, temperature, humidity, corrosiveness, EMI or nuclear radiation (including duration exposure)?	Projection is that 30 years life still remain. GE evaluation documented in DRF-A61-00052- Tab 6 concluded that the pumps will be acceptable for an additional 30 years at EPU conditions, with all three pumps running.	N/A
Interface considerations: Does the changed parameter have a potentially adverse effect on an interfacing system (including functional and physical interfaces)? [If "Yes," this information must be communicated to the EPU review team reviewing the affected system.]	Flow from minimum flow valve is piped to the main condenser. No change in interface because the flowrate is controlled by an installed orifice in the feedwater system downstream of the minimum flow valve.	N/A
Mechanical/structural considerations: Does the changed parameter have an adverse effect on component vibration, shock, reaction forces, equipment foundations or pipe supports?	See Load and load combinations attribute (above)	N/A
Hydraulic considerations: Does the changed parameter have an adverse effect on pump net positive suction head, allowable pressure drop, allowable fluid velocity, fluid state, margin to flashing or steam quality?	Flow from minimum flow valve is piped to the main condenser. No change in interface because the flowrate is controlled by an installed orifice in the feedwater system downstream of the minimum flow valve.	N/A

Attribute	Evaluation*	Action?
Electrical considerations: Does the changed parameter have an adverse effect on component power, voltage, raceway requirements, fault current, electrical insulation or motor requirements? Does the changed parameter adversely affect interfacing electrical systems (e.g., motor control center)?	No – This is an air operated valve with the air control ed by a solenoid valve.	N/A
Instrumentation and control considerations: Does the changed parameter affect component setpoints, instrument ranges or calibrations?	No change to setpoint; setpoint based on pump protections; not affected by EPU.	N/A
Reliability and service life considerations: Does the changed parameter result in additional component cycles, changes in maintenance requirements (e.g., online maintenance changed to outage maintenance) or increased wear?	See Operating Margin Attribute (above)	N/A

- This evaluation should consider previous evaluations such that vulnerabilities that have already been identified and dispositioned (as verified by the review team) are not identified as vulnerabilities on this worksheet.

EXTENT OF CONDITION - EVALUATION CRITERIA VULNERABILITY REVIEW

System: Feedwater System (3200)
Parameter(s): Flow
Component(s): MO-3201, RFP DISCH VLV
3208 - RFP DISCH CK VLV
Subcomponent(s): N/A

Instructions: The purpose of this form is to assess potential vulnerabilities related to a changed system parameter or parameters. For each attribute, the associated questions should be evaluated relative to the effect of the changed parameter on a system component or components. For the system level evaluation, if "Yes" is indicated in the "Action?" column, a component level evaluation is required for that changed parameter. For the component level evaluation, if "Yes" is indicated, the potential vulnerability must be evaluated for potential failure mechanisms and a risk ranking must be assigned.

Attribute	Evaluation	Action?
Component ratings: Does the changed parameter exceed a component rating or design value, e.g., with regard to capacity, duty, pressure, flow, temperature, fluid state, fluid chemistry or voltage?	No - Normal 100% flow has decreased following EPU.	N/A
Operating margin: Does the changed parameter reduce the operating margin for a component to an unacceptable level? Does it require that standby equipment now operate during normal conditions such that the margin is reduced to an unacceptable level?	No - Normal 100% flow has decreased following EPU.	N/A
Material condition: Does the changed parameter have an adverse effect on the material condition of a component such that the performance of the component is significantly affected or the margin for degradation is reduced to an unacceptable level? Consider whether the material condition was degraded pre-EPU and whether EPU could aggravate this degradation unacceptably.	Normal 100% flow has decreased following EPU, however, the swing in the suction pressure for Quad Unit 2 (as shown by the Pi data) may cause partial movement in the feedwater pump discharge check valve. The system engineer documented that the flows and pressures in the feedwater system oscillate with a period of about 10 seconds. The peak-to-peak amplitude of the pressure oscillations is about 20 psi in the winter and about 40 psi in the summer. The driver for this oscillation has not been identified and it is not known whether this oscillation also occurred pre-EPU. It may be possible to reduce this oscillation with further tuning of the feed reg valve but the need for any corrective measure has not been established. Potential concern is effect of cyclic temperature on motor winding. A 10-second period of oscillation was also noted pre-EPU and attributed to a loose thrust nut. Another potential may be advanced wear on the feedwater pump discharge check valve flapper hinge bushings; if the pressure oscillations result in flow oscillations that affect the position of the valve flapper. If this condition persists, then the PM for the discharge check valve should be increased to manage the potential for accelerated wear on the hinge bushings.	Yes
Load and load combinations: Does the changed parameter result in loads on a component (e.g., thermal or dynamic) such that the margin to failure is reduced to an unacceptable level?	No - Normal 100% flow has decreased following EPU.	N/A
Environmental conditions and environmental qualification: Does the changed parameter have an adverse effect on a component relative to EQ, for example, due to changes in pressure, temperature, humidity, corrosiveness, EMI or nuclear radiation (including duration exposure)?	Limited affect on Buna-N in this system.	N/A
Interface considerations: Does the changed parameter have a potentially adverse effect on an interfacing system (including functional and physical interfaces)? [If "Yes," this information must be communicated to the EPU review team reviewing the affected system.]	No interface with other systems.	N/A
Mechanical/structural considerations: Does the changed parameter have an adverse effect on component vibration, shock, reaction forces, equipment foundations or pipe supports?	There may be additional vibrations imposed on the discharge valve (3201) based on the operating experience gained from the system engineers. It was stated that the vibrations have increased due to three feedwater pumps in operations. Recommend vibration analysis of the valve, particularly of the Motor Operator.	Yes

Attribute	Evaluation*	Action?
Hydraulic considerations: Does the changed parameter have an adverse effect on pump net positive suction head, allowable pressure drop, allowable fluid velocity, fluid state, margin to flashing or steam quality?	No - Normal 100% flow has decreased following EPU.	N/A
Electrical considerations: Does the changed parameter have an adverse effect on component power, voltage, raceway requirements, fault current, electrical insulation or motor requirements? Does the changed parameter adversely affect interfacing electrical systems (e.g., motor control center)?	No – Each motor operated discharge valve is normally open with the their respective feedwater pumps in service.	N/A
Instrumentation and control considerations: Does the changed parameter affect component setpoints, instrument ranges or calibrations?	N/A	N/A
Reliability and service life considerations: Does the changed parameter result in additional component cycles, changes in maintenance requirements (e.g., online maintenance changed to outage maintenance) or increased wear?	See Material Condition Attribute (above)	N/A

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EXTENT OF CONDITION - EVALUATION CRITERIA VULNERABILITY REVIEW

System: Feedwater System (3200)
Parameter(s): Pressure
Component(s): MO-3201-A, RFP DISCH VLV
MO-3202 D1 HP HTR FW INLET VLV
MO-3203, D' HEATERS BYP VLV
MO-3204, D1 HP HTR FW OUTLET VLV
MO-3205, FW INLET ISOL VLV (MOV)
MO-3206, FW REG VLV INLET SV
3208 - RFP DISCH CK VLV
0220-59, Feedwater System to Reactor Check Valve
0220-62, Outboard Feedwater System to Reactor Check Valve
0220-58, Inboard Feedwater System to Reactor Check Valve
0220-57, Feedwater to Reactor Manual Isolation Valve
Subcomponent(s): N/A

Instructions: The purpose of this form is to assess potential vulnerabilities related to a changed system parameter or parameters. For each attribute, the associated questions should be evaluated relative to the effect of the changed parameter on a system component or components. For the system level evaluation, if "Yes" is indicated in the "Action?" column, a component level evaluation is required for that changed parameter. For the component level evaluation, if "Yes" is indicated, the potential vulnerability must be evaluated for potential failure mechanisms and a risk ranking must be assigned.

Attribute	Evaluation*	Action?
Component ratings: Does the changed parameter exceed a component rating or design value, e.g., with regard to capacity, duty, pressure, flow, temperature, fluid state, fluid chemistry or voltage?	No. The existing feedwater design pressure and temperature requirements are adequate , reference T0701. However, the actual pressure increase is greater than predicted. Dresden Unit 2 Feedwater Pump Discharge Pressure averages 1506 psig during the winter. Predicted value from the TPU Heat Balance was 1353 psia. The system design pressure is stated in Dresden Spec K-4080 and Quad Cities Spec R-4411 to be 1850 psig. This is based on the shut-off head of the feedwater pumps. Current feedwater pumps' shut-off head curves ares consistent with this value. References: Dresden Station RFP Pump Curves, Byron Jackon Pumps, T-29203-2, T-29227-1, T29226-1, T29447-1, T-29448-1 and 29446-1, Quad Cities RFP curves 33069/A, 33069/B, 33069/C, 33070/A, 33070/B, and 33070/C	N/A
Operating margin: Does the changed parameter reduce the operating margin for a component to an unacceptable level? Does it require that standby equipment now operate during normal conditions such that the margin is reduced to an unacceptable level?	System maximum pressure pressure is not changed by EPU. The feedwater pump shut-off head pressure is not changed.	N/A
Material condition: Does the changed parameter have an adverse effect on the material condition of a component such that the performance of the component is significantly affected or the margin for degradation is reduced to an unacceptable level? Consider whether the material condition was degraded pre-EPU and whether EPU could aggravate this degradation unacceptably.	No - Parameter does not change with EPU.	N/A
Load and load combinations: Does the changed parameter result in loads on a component (e.g., thermal or dynamic) such that the margin to failure is reduced to an unacceptable level?	No - Parameter does not change with EPU.	N/A
Environmental conditions and environmental qualification: Does the changed parameter have an adverse effect on a component relative to EQ, for example, due to changes in pressure, temperature, humidity, corrosiveness, EMI or nuclear radiation (including duration exposure)?	No - Parameter does not change with EPU.	N/A
Interface considerations: Does the changed parameter have a potentially adverse effect on an interfacing system (including functional and physical interfaces)? [If "Yes," this information must be communicated to the EPU review team reviewing the affected system.]	No - Parameter does not change with EPU.	N/A
Mechanical/structural considerations: Does the changed parameter have an adverse effect on component vibration, shock, reaction forces, equipment foundations or pipe supports?	No - Parameter does not change with EPU.	N/A

Attribute	Evaluation*	Action?
Hydraulic considerations: Does the changed parameter have an adverse effect on pump net positive suction head, allowable pressure drop, allowable fluid velocity, fluid state, margin to flashing or steam quality?	No - Parameter does not change with EPU.	N/A
Electrical considerations: Does the changed parameter have an adverse effect on component power, voltage, raceway requirements, fault current, electrical insulation or motor requirements? Does the changed parameter adversely affect interfacing electrical systems (e.g., motor control center)?	No - Parameter does not change with EPU.	N/A
Instrumentation and control considerations: Does the changed parameter affect component setpoints, instrument ranges or calibrations?	No - Parameter does not change with EPU.	N/A
Reliability and service life considerations: Does the changed parameter result in additional component cycles, changes in maintenance requirements (e.g., online maintenance changed to outage maintenance) or increased wear?	No - Parameter does not change with EPU.	N/A

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EXTENT OF CONDITION - EVALUATION CRITERIA VULNERABILITY REVIEW

System: Feedwater System (3200)
Parameter(s): Flow
Component(s): MO-3202, D1 HP HTR FW INLET VLV
MO-3203, D' HEATERS BYP VLV
MO-3204, D1 HP HTR FW OUTLET VLV
MO-3205, FW INLET ISOL VLV (MOV)
MO-3206, FW REG VLV INLET SV
0220-59, Feedwater System to Reactor Check Valve
0220-62, Outboard Feedwater System to Reactor Check Valve
0220-58, Inboard Feedwater System to Reactor Check Valve
0220-57, Feedwater to Reactor Manual Isolation Valve
Subcomponent(s): N/A

Instructions: The purpose of this form is to assess potential vulnerabilities related to a changed system parameter or parameters. For each attribute, the associated questions should be evaluated relative to the effect of the changed parameter on a system component or components. For the system level evaluation, if "Yes" is indicated in the "Action?" column, a component level evaluation is required for that changed parameter. For the component level evaluation, if "Yes" is indicated, the potential vulnerability must be evaluated for potential failure mechanisms and a risk ranking must be assigned.

Attribute	Evaluation*	Action?
Component ratings: Does the changed parameter exceed a component rating or design value, e.g., with regard to capacity, duty, pressure, flow, temperature, fluid state, fluid chemistry or voltage?	Within design pressure per K-4080 and R-4411.	N/A
Operating margin: Does the changed parameter reduce the operating margin for a component to an unacceptable level? Does it require that standby equipment now operate during normal conditions such that the margin is reduced to an unacceptable level?	The feedwater flows have increased from pre-EPU VWO conditions of 9,724,966 lbs/hr (3,241,655 lbs/hr per string/ 4,862,483 lbs/hr per injection line and FCV), reference 349HB394 to 11,769,380 lbs/hr (3,923,126 lbs/hr per string and pump/ 5,884,690 lbs/hr per FCV and injection line) at post-EPU VWO conditions, reference 552HB1989 R2. This is a 21% increase that is outside the original design basis and is a significant change. Although the actual post-EPU operating flows are below the analyzed EPU VWO conditions per pump, with this increase in flow and different pump operation that can cause piping and component vibration changes and changes in system pressure losses. Valves experiencing a potential increase of flow to 3,923,126 lbs/hr: MO-3202, D HP HTR FW INLET VLV MO-3203, D HEATERS BYP VLV MO-3204, D HP HTR FW OUTLET VLV Valves experiencing a potential increase of flow to 5,884,690 lbs/hr: MO-3205, FW INLET ISOL VLV (MOV) MO-3206, FW REG VLV INLET SV 0220-59, Feedwater System to Reactor Check Valve 0220-62, Outboard Feedwater System to Reactor Check Valve 0220-58, Inboard Feedwater System to Reactor Check Valve 0220-57, Feedwater to Reactor Manual Isolation Valve	N/A
Material condition: Does the changed parameter have an adverse effect on the material condition of a component such that the performance of the component is significantly affected or the margin for degradation is reduced to an unacceptable level? Consider whether the material condition was degraded pre-EPU and whether EPU could aggravate this degradation unacceptably.	There may be additional vibrations imposed on the valves, particularly Motor-Operated valves, based on the operating experience gained from the system engineers. It was stated that the vibrations have increased due to three feedwater pumps in operations. Recommend vibration analysis of the valve, particularly of the Motor Operator.	N/A

Attribute	Evaluation	Action?
Load and load combinations: Does the changed parameter result in loads on a component (e.g., thermal or dynamic) such that the margin to failure is reduced to an unacceptable level?	With an increased flow a sudden change of flow should not have an impact on the check valves because the flow has to go from a higher flow through the pre-EPU flow value. In addition, a higher flow results in a higher differential pressure for valve closure which may imposed additional loads on the motor operated valves. It is understood that the margjns for the motor operater motors are in the range of 200%. With an increased differential pressure across the MO-3203, D' HEATERS BYP VLV,the primary potential is to affect the stroke time of the valve. The assumptions made of this valve in the transient analysis of the plants is that the performance of the valve will be no better than a stroke time of one minute. Therefore no affect on the affect on the loss of feedwater heating event and there is no foreseen loss of margin. Reference T0900 FTR; OPL-3	N/A
Environmental conditions and environmental qualification: Does the changed parameter have an adverse effect on a component relative to EQ, for example, due to changes in pressure, temperature, humidity, corrosiveness, EMI or nuclear radiation (including duration exposure)?	No – EPU increased the temperature of the htr bay, RFP and turbine building area are less than 10F, not exceeding 140F This increased temperature and radiological conditions on the motor operators has a minor (negligible) effect on the thermal and radiation aging of the motor (operator) and its associated components.	N/A
Interface considerations: Does the changed parameter have a potentially adverse effect on an interfacing system (including functional and physical interfaces)? [If "Yes," this information must be communicated to the EPU review team reviewing the affected system.]	No – No issues identified	N/A
Mechanical/structural considerations: Does the changed parameter have an adverse effect on component vibration, shock, reaction forces, equipment foundations or pipe supports?	No - Increase flow could increase vibration levels of valve, actuator and motor, possible leading to failure due to wear or fatigue. However, no specific problems with MOVs have been reported, and there is no basis to expect future problems caused by EPU	N/A
Hydraulic considerations: Does the changed parameter have an adverse effect on pump net positive suction head, allowable pressure drop, allowable fluid velocity, fluid state, margin to flashing or steam quality?	No – The Feedwater System MOVs are used of isolation and not flow throttling. Therefore, hydraulic considerations due to increase EPU flows need not be further discussed	N/A
Electrical considerations: Does the changed parameter have an adverse effect on component power, voltage, raceway requirements, fault current, electrical insulation or motor requirements? Does the changed parameter adversely affect interfacing electrical systems (e.g., motor control center)?	No – The increased temperature and radiological conditions in the areas has a minor effect on the MOV operation. This minor effect is considered negligible on component operation and maintenance (see Reliability and service life considerations attribute evaluation)..	N/A
Instrumentation and control considerations: Does the changed parameter affect component setpoints, instrument ranges or calibrations?	No – No issues identified	N/A
Reliability and service life considerations: Does the changed parameter result in additional component cycles, changes in maintenance requirements (e.g., online maintenance changed to outage maintenance) or increased wear?	Per the MOV template per MA-AA-716-210-1001, a severe service condition is defined as being in an ambient condition of > 120 deg F or installed in a fluid medium of >400 deg F. With the increase in ambient temperature from 120 to 125 deg F, some of the Feedwater System MOVs are now considered in a severe service condition. However, this severe operating condition is during summer operation only.	N/A

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EXTENT OF CONDITION - EVALUATION CRITERIA VULNERABILITY REVIEW

System: Feedwater System (3200)
Parameter(s): Flow
Component(s): Feedwater Reg. Valve Drag Valve (DRESDEN)
Quad HO Reg. Valve
Subcomponent(s): N/A

Instructions: The purpose of this form is to assess potential vulnerabilities related to a changed system parameter or parameters. For each attribute, the associated questions should be evaluated relative to the effect of the changed parameter on a system component or components. For the system level evaluation, if "Yes" is indicated in the "Action?" column, a component level evaluation is required for that changed parameter. For the component level evaluation, if "Yes" is indicated, the potential vulnerability must be evaluated for potential failure mechanisms and a risk ranking must be assigned.

Attribute	Evaluation*	Action?
Component ratings: Does the changed parameter exceed a component rating or design value, e.g., with regard to capacity, duty, pressure, flow, temperature, fluid state, fluid chemistry or voltage?	No – Valve design is within the design valves. The operating conditions are listed up to 9.8M#/Hr, for each valve, however that is just one of two conditions calculated based on the design of the disk stack from which the Cv is produced by the vendor. Reference 469601-1	N/A
Operating margin: Does the changed parameter reduce the operating margin for a component to an unacceptable level? Does it require that standby equipment now operate during normal conditions such that the margin is reduced to an unacceptable level?	No – An additional (3rd) feedwater pump is placed in service prior to the plant reached 100% OLTP. This provides the additional head (and therefore flow) for the control valves to operate with margin at the higher flow rate needed to support EPU conditions. Reference T0701 FTR	N/A
Material condition: Does the changed parameter have an adverse effect on the material condition of a component such that the performance of the component is significantly affected or the margin for degradation is reduced to an unacceptable level? Consider whether the material condition was degraded pre-EPU and whether EPU could aggravate this degradation unacceptably.	Yes - The vibrations at the feedwater control stations existed prior to EPU conditions; however based on systems engineering interviews the vibrations have appeared to increase. Even though one of the feedwater regulating control valves are overhauled at each refueling it is recommended that both of the feedwater control valves be disassembled and inspected at the next opportunity to ensure that wear due to vibrations is monitored and tracked to prevent potential failures due to wear.	Yes
Load and load combinations: Does the changed parameter result in loads on a component (e.g., thermal or dynamic) such that the margin to failure is reduced to an unacceptable level?	No – None noted.	N/A
Environmental conditions and environmental qualification: Does the changed parameter have an adverse effect on a component relative to EQ, for example, due to changes in pressure, temperature, humidity, corrosiveness, EMI or nuclear radiation (including duration exposure)?	Not a significant change in environmental conditions. Reference input from system engineer and plant temperature profiles as produced for this project.	N/A
Interface considerations: Does the changed parameter have a potentially adverse effect on an interfacing system (including functional and physical interfaces)? [If "Yes," this information must be communicated to the EPU review team reviewing the affected system.]	None – The valve responds to control signals from the feedwater control system.	N/A
Mechanical/structural considerations: Does the changed parameter have an adverse effect on component vibration, shock, reaction forces, equipment foundations or pipe supports?	Yes – See Material Condition above.	Yes
Hydraulic considerations: Does the changed parameter have an adverse effect on pump net positive suction head, allowable pressure drop, allowable fluid velocity, fluid state, margin to flashing or steam quality?	No- Because a third feedwater pump has to be started to allow for proper positioning of the feedwater regulating valves; to provide additional flow, therefore, the pressure drop across the control valves increases and the discharge pressure of the feedwater pumps increases. These changes have a potential negative affect on the feedwater pumps (see the feedwater pumps vulnerability review) and potential additional wear and vibrations on the feedwater regulating valves due to the additional pressure drop and flow.	N/A
Electrical considerations: Does the changed parameter have an adverse effect on component power, voltage, raceway requirements, fault current, electrical insulation or motor requirements? Does the changed parameter adversely affect interfacing electrical systems (e.g., motor control center)?	No change – The same amount of motive power (air for Dresden, hydraulic oil for Quad) is necessary before EPU as there is after EPU.	N/A

Attribute	Evaluation*	Action?
Instrumentation and control considerations: Does the changed parameter affect component setpoints, instrument ranges or calibrations?	No affect directly on the controls for this valve as they are outside the scope of this system review.	N/A
Reliability and service life considerations: Does the changed parameter result in additional component cycles, changes in maintenance requirements (e.g., online maintenance changed to outage maintenance) or increased wear?	There is additional flow and potential for vibration affects on these valves. There is a possibility of increased wear due to EPU. Even though one of the feedwater regulating control valves are overhauled at each refueling it is recommended that both of the feedwater control valves be disassembled and inspected at the next opportunity to ensure that wear due to vibrations is monitored and tracked to prevent potential failures due to wear.	Yes

- This evaluation should consider previous evaluations such that vulnerabilities that have already been identified and dispositioned (as verified by the review team) are not identified as vulnerabilities on this worksheet.

EXTENT OF CONDITION - EVALUATION CRITERIA VULNERABILITY REVIEW

System: Feedwater System (3200)
Parameter(s): Flow
Component(s): Pumps
Subcomponent(s): N/A

Instructions: The purpose of this form is to assess potential vulnerabilities related to a changed system parameter or parameters. For each attribute, the associated questions should be evaluated relative to the effect of the changed parameter on a system component or components. For the system level evaluation, if "Yes" is indicated in the "Action?" column, a component level evaluation is required for that changed parameter. For the component level evaluation, if "Yes" is indicated, the potential vulnerability must be evaluated for potential failure mechanisms and a risk ranking must be assigned.

Attribute	Evaluation	Action?
Component ratings: Does the changed parameter exceed a component rating or design value, e.g., with regard to capacity, duty, pressure, flow, temperature, fluid state, fluid chemistry or voltage?	No. The changed parameters (flow, pressure & temperature) do not exceed ratings or design values associated with the feedwater pumps or motors. The pumps are no longer operating at their design condition with regards to the best efficiency point, however; a flowrate per pump down to 8000 gpm will be close to BEP as in pre-EPU conditions. The BEP centers between 10,000 to 10,500 gpm for the Feedwater Pumps. At EPU conditions the flow per pump is about 8770 gpm; therefore the feedwater pumps will be operating at about the same efficiency as before EPU. References: Dresden Station RFP Pump Curves, Byron Jackon Pumps, T-29203-2, T-29227-1, T29226-1, T29447-1, T-29448-1 and 29446-1, Quad Cities RFP curves 33069/A, 33069/B, 33069/C, 33070/A, 33070/B, and 33070/C.	N/A
Operating margin: Does the changed parameter reduce the operating margin for a component to an unacceptable level? Does it require that standby equipment now operate during normal conditions such that the margin is reduced to an unacceptable level?	Yes. The higher total flow requirement at EPU conditions requires that equipment (feed pumps) previously in standby at 100% pre-EPU power level be operated 24/7 at 100% EPU power levels. This is a previously identified vulnerability. Per Task Report T0701, when operating with the modified high pressure turbine with only two feed and three condensate / condensate booster pumps on line, the flow passing capability of the feedwater flow control valves will become limiting for operation near 100% of 2527/2511 MWth (pre-EPU 100% power level). In order to achieve maximum operational flexibility a third feed pump should be placed on-line prior to exceeding 100% of 2527/2511 MWth (85.5% LPU). Maintenance or loss of feedwater pumps/motors will result in unit derates during non-outage periods or to avoid derates will have to occur during outage periods. Per discussion with system engineers, there has been no full-range testing to confirm/identify achievable power with less than the full complement of pumps. There is some (probably small) potential that full-range testing of pump capability might support operation at 100% EPU power with one of the pumps in standby or might support identification of cost-effective modifications (e.g., change in impeller design) that would allow restoration of the pre-EPU stand-by pump configuration.	Yes

Attribute	Evaluation	Action?
Material condition: Does the changed parameter have an adverse effect on the material condition of a component such that the performance of the component is significantly affected or the margin for degradation is reduced to an unacceptable level? Consider whether the material condition was degraded pre-EPU and whether EPU could aggravate this degradation unacceptably.	The Feedwater Pumps seal performance has been reviewed in the past due to many failures. The maintenance plan specifies a replacement of the seals every eight years. CMO suggests that the failure of the seals has increased due to the increased discharge pressure due to three pump EPU operation. Recommend that the seal PM be increased and a review of three-pump operation be performed in order to prevent a down power. Reference Diary of Significant Events. Normal 100% flow has decreased following EPU, however, the swing in the discharge pressure for Quad Cities Unit 2 may cause unexpected wear in the Feedwater Pumps due to the ocillating forces, translating into movement, on the pump shaft that may cause extra wear on the thrust bearings and pump seals. The system engineer documented that the flows and pressures in the feedwater system oscillate with a period of about 10 seconds. The peak-to-peak amplitude of the pressure oscillations is about 20 psi in the winter and about 40 psi in the summer. The driver for this oscillation has not been identified and it is not known whether this oscillation also occurred pre-EPU. It may be possible to reduce this oscillation with further tuning of the feed reg valve but the need for any corrective measure has not been established. Potential concern is effect of cyclic temperature on motor winding. A 10-second period of oscillation was also noted pre-EPU and attributed to a loose thrust nut.	Yes
Load and load combinations: Does the changed parameter result in loads on a component (e.g., thermal or dynamic) such that the margin to failure is reduced to an unacceptable level?	None.	N/A
Environmental conditions and environmental qualification: Does the changed parameter have an adverse effect on a component relative to EQ, for example, due to changes in pressure, temperature, humidity, corrosiveness, EMI or nuclear radiation (including duration exposure)?	No - Feedwater Pumps operate in areas with less airflow to the motors than pre-EPU (with some exceptions at Quad). Motor discharge air temperatures are normally controlled at 125F. The worst summer case at Quad was identified with a motor discharge air temperature of 140F. An analysis has been performed to evaluate the life of the Feedwater Pump Motors at the EPU conditions. This evaluation concluded that the motors, assuming three feedwater pump operations, have an expected life of an additional 30 years based on traditional maintenance practices for large motors. The evaluation work scope included motor windings and mechanical components. The evaluation did take into consideration that the motors are tasked with a lower horsepower output or 7400 Hp compared with the original rated value of 9000 Hp. Reference GE-NE DRF A61-00052	N/A
Interface considerations: Does the changed parameter have a potentially adverse effect on an interfacing system (including functional and physical interfaces)? [If "Yes," this information must be communicated to the EPU review team reviewing the affected system.]	No identified issues.	N/A
Mechanical/structural considerations: Does the changed parameter have an adverse effect on component vibration, shock, reaction forces, equipment foundations or pipe supports?	Yes - System engineer reports noticably increased vibration (possibly flow-change or equipment line-up induced) in small-bore pipe and tubing associated with the Feedwater System. Recommend to measure the vibrations of FW components, especially in the FW Pump Rooms and in the vicinity of the FW Regulation Valves	Yes

Attribute	Evaluation*	Action?
Hydraulic considerations: Does the changed parameter have an adverse effect on pump net positive suction head, allowable pressure drop, allowable fluid velocity, fluid state, margin to flashing or steam quality?	Yes. Operation in a 3 FW pump configuration at 100% EPU power has reduced the individual pump flow and causes the individual FW pumps to operate lower on their pump curve, away from the BEP. This move away from the BEP is suspected to be a contributor to increased seal wear and has a potential to cause cavitation caused by pump internal recirculation. There is some (probably small) potential that full-range testing of pump capability might support operation at 100% EPU power with one of the pumps in standby or might support identification of cost-effective modifications (e.g., change in impeller design) that would allow restoration of the pre-EPU stand-by pump configuration	Yes
Electrical considerations: Does the changed parameter have an adverse effect on component power, voltage, raceway requirements, fault current, electrical insulation or motor requirements? Does the changed parameter adversely affect interfacing electrical systems (e.g., motor control center)?	No – The load per pump has decreased from approximately 8117 Hp to 7400 Hp. Reference GE-NE-DRF-A61-00052.	N/A
Instrumentation and control considerations: Does the changed parameter affect component setpoints, instrument ranges or calibrations?	No. Changes in Flow, Pressure and Temperature are within the range of existing instruments.	N/A
Reliability and service life considerations: Does the changed parameter result in additional component cycles, changes in maintenance requirements (e.g., online maintenance changed to outage maintenance) or increased wear?	Yes. Maintenance that could be performed on-line at 100% power now can be performed only at an outage or at derated power since there are no spare Feedwater Pumps at current 100% EPU power. This is a previously identified pinch point. Feedwater Pump on line work will be recoded to refuel outages to avoid derates. Feedwater SR's be handled under AR 91238 task 61 for PM outage work.	Yes

- This evaluation should consider previous evaluations such that vulnerabilities that have already been identified and dispositioned (as verified by the review team) are not identified as vulnerabilities on this worksheet.

7. Interview Results (Project Instruction Attachments 2 & 3)

Interview results are attached for increased heat loads for the FW systems.

Attachment 3

Questionnaire for Interviews of Dresden and Quad Cities Personnel

D/Q EPU EOC INTERVIEWS

Date: February 5, 2004, February 19, 2004
FEEDWATER SYSTEM

Interviewer(s):
Brian Stedman, Exelon
George Paptzun, GE
Bob Jackson, GE
Alex Zarechnak, MPR

PARTICIPANTS	TITLE
JAY ROLFES 309-227-3235	**SENIOR PLANT ENGINEER** Quad Cities Feedwater System
HARRY PALAS 630-657-3881	**CORPORATE ENGINEERING** Pump Specialist
NED ZEMAN 815-455-2974	**PUMP CONSULTANT** Zeman Sales and Engineering
TERRY HOFFMAN (BY PHONE) 815-416-2248	**CMO** Dresden Pump Specialist
TONY BEZOUSKA (FEB 19) 815-416-2972	**SENIOR PLANT ENGINEER** Dresden Feedwater System
JOHN KRUTHOFF (FEB 19) 309-227-2151	**VIBRATION AND THERMOGRAPHY SPECIALIST** Quad Cities
JOHN ULLRICH (FEB 19) 309-227-2190	**VIBRATION SPECIALIST** Quad Cities

Topic(s): Feedwater System; condensate pumps and condensate booster pumps were also discussed; see also separate interview notes for condensate systems.

The purpose of this interview is to identify potential EPU-related vulnerabilities that could result in one or more of the following:

- License Event Report
- Engineered Safety Features Actuation
- Reactor Scram
- Plant power derate
- Unplanned entry into Technical Specification
- Operator "Work Around" that increases the risk to one of the above events.
- Unexpected accelerated equipment that increases the risk to one of the above events.

Note: Responses to the interview questions should consider different operating modes/configurations: normal operation (summer and winter); components out of service (e.g., FWH string, FW pump); ECCS conditions.

Systems:		
	Question	Response
1	How has system operation changed following the EPU? What parameters have changed (pressure, temperature, flow, fluid state, moisture content)?	Feedwater flow has increased about 17% for both Dresden and Quad Cities.
2	Is system operation since the EPU as expected? Have there been any unexpected problems, disappointments, concerns or issues?	**Vibration due to higher flow** There have been several instances of post-EPU failures of small-bore piping attached to the feedwater system at Dresden and Quad Cities (the most susceptible being the small bore vents and drains, seal cooling lines, suction RVs and oil piping). These failures have been attributed to high cycle fatigue caused by higher post-EPU vibration levels. Since the endurance limit has been reached for these oscillations, additional such failures are not expected during normal operation. High cycle fatigue failure may still occur at off-normal operating conditions. Evaluation of vibration levels of small-bore piping has and is being conducted using the EPRI process and spreadsheet. This methodology indicates if the measured vibration levels for the current configuration are acceptable. Handwheels on many manual valves at Quad Cities have a buzzing vibration. The stem and attached handwheel on the manual isolation valve for the pressure gage (1-2141-141B) on the 1B feedwater pump discharge have broken off due to vibration. As a result this manual valve is currently not usable and will be repaired during the next outage. Vibration for the Quad Cities feedwater pumps are perceived to be higher than pre-EPU but have not increased to the action level, and thus there is no quantitative indication or analysis of increased vibration.

Systems:		
	Question	Response
		Dresden recently experienced vibration alerts for the 3C and 2C feed pumps. The 3C alert was caused by a motor problem during steady state plant operations, the alarm cleared within about seven minutes; observed vibrations were found to be acceptable. The 2C alert followed the pump overhaul with a new rotor; such alarms are not unusual during plant startups. Vibration measurements are planned for the next outage for the Dresden 2B feedpump suction relief valve. Jeff Drowley's reported impression was that vibration levels on the Feedwater piping at Dresden post-EPU are much higher than they were when he was last in the Dresden plant in 1996. Quad Cities had prepared a plan for additional small bore feedwater piping vibration measurements on Unit 1 but this plan was not implemented due to load restrictions. The Quad Cities vibration specialist agreed to provide a copy of this plan to the EPU review team. The majority of piping attached to the Feedwater system at Quad Cities has tie-back supports and 2:1 welds Quad Cities, unlike Dresden does not have vibration monitoring at their feed reg valves and feedpumps. Dresden U2 feedwater check valve 1A was recently inspected, for the first time in twelve years, and no issues with wear were observed. The Feedwater Regulating Valves have always experienced high vibrations; inspections performed every outage have not revealed an increase in wear.

Systems:		
	Question	Response
		Oscillating flows and pressures Flows and pressures in the feedwater system oscillate with a period of about 10 seconds. The peak-to-peak amplitude of the pressure oscillations is about 20 psi in the winter and about 40 psi in the summer. The driver for this oscillation has not been identified and it is not known whether this oscillation also occurred pre-EPU. It may be possible to reduce this oscillation with further tuning of the feed reg valve but the need for any corrective measure has not been established. Potential concern is effect of cyclic temperature on motor winding. A 10-second period of oscillation was also noted pre-EPU and attributed to a loose thrust nut. Feedwater system flow noise (peak to peak), as reflected in the pressure signal, is higher post-EPU. This higher noise level may impact the AMAG flow meter calibration at Dresden especially if much under 25 Hz. **Feedwater Pump Casing Drains:** There have been unexpected failures from feedwater pump casing drain isolation valves. One, in particular, started leaking (unisolable) during operations and was sealed shut with a Fermanite Product. It is thought that increased vibrations may have caused valve failure. Dresden feed pump discharge pressure has increased from about 1200 psi to about 1450 psi. Higher pressure increases the stress and potential for leakage by the pump casing gasket. The Feedwater Pumps are horizontal split casing pumps that have experienced erosion of the casing gasket before EPU. The feed pump casing is being modified to address this issue.

Systems:		
	Question	Response
3	Was system operation post-EPU affected by changes in the performance of other systems?	See discussion of interaction with condensate booster minimum flow.
4	What components were affected by the EPU—either directly by changes in system parameters or indirectly by changes in interfacing systems?	**Feedwater pumps** Running with all three feed pumps results in operation higher up on the head/flow curve, near 70% of best efficiency point. The vane-passing frequency is now more pronounced because we are farther from best efficiency point but pump vibrations are still within allowables. Typically problems with internal recirculation cavitation occur below 40%. Vibration levels are slightly higher than pre-EPU. ***(Rolfes to forward available vibration data).*** The SE expressed concern that these higher vibration levels may challenge the pump seals and bearings. However, no failures of seals or bearings have been attributed to EPU and generally seal performance has been improved post-EPU because we are operating with improved Bergman seals. The 2A QC feed pumps will be inspected this upcoming outage Q2R17 followed by 1A (Q1R18), 2B (Q2R18), 1B (Q1R19), 2C (Q2R18), 1C (Q1R19). The SE agreed that this planned inspection needs to include examination for impeller damage attributed to internal recirculation cavitation observed recently for a condensate pump at Dresden. The original EPU plan included scheduled mid-cycle plant derates to perform individual feed pump inspections and refurbishment. After the first such derate further derates were not implemented and all refurbishment work is now scheduled for the refueling outages. Overall reliability of the feed pumps is now a greater challenge per the Maintenance Rule. Recent refurbishments of the U2 feed pumps at Dresden included installing 16.5-inch impellers. Originally, 16" impellers were installed which is

Systems:		
	Question	Response
		reflected on the 'test' pump curves. Another 'calculated' pump curve exists for the 16.5" impellers. At least two of three feed pumps had 16-inch impellers prior to this refurbishment. Similar refurbishment is planned for U3 (3A and 3B feedwater pumps). The resulting improved performance could increase the maximum output for two-pump operation. **Operation with two feed pumps and three condensate pumps** The feedwater system engineer and the corporate rotating equipment engineer noted that the limits of power operation at Quad Cities with two feedwater pumps and three condensate/booster pumps have not been established. The EPU design and operation were based on Dresden which is more limiting. It may be desirable at Quad Cities to operate with fewer pumps if the resulting plant power is not limited. Further, even at Dresden it may be possible to operate with two feedpumps and three condensate pumps at full power with appropriate upgrades to these pumps or with the addition of variable speed drives. There are several undesirable issues associated with running with all three feed pumps. One of these is that with two pumps running the electric load could be shared equally between two independent buses. With three pump operation, two of the pumps are on one bus and the third is on another. Electrical transients that result in loss of a bus will be more severe with the three-pump combination. Another disadvantage is that about 4 MW of power are lost in this configuration since the pumps are operating off their best efficiency point and the feed reg valves are dissipating significant energy. *N OTE: this option could be considered as a potential EPU enhancement or as a potential solution to unresolved problems with continuous operation with three feed pumps and four condensate/booster pumps.*

Systems:		
	Question	Response
		Electric Relays in Adjacent Spaces The SE has observed that vibration levels in spaces adjacent to the feedwater pumps and expressed concern that these increased vibration levels may affect the relays located in racks (2252-9/2251/9) in these spaces. These relays include feedpump relays and High Reactor Water Level relays for Unit 1. The Unit 1 relay rack is located inside the RFP room while the Unit 2 relay rack is located outside the RFP room near the 4kV switchgear. The SE has observed grating vibrations in the CRD level next to the heater drain controls. It was identified that a piping support on the condensate booster pump discharge had a loose rod in Unit 2 (lock nut loose allowed turnbuckle to spin), and the same line in Unit 1 at a different location had a loose pipe clamp nut. For Q2R17 the SE will inspect all Unit 2 accessible FW piping and heater drain piping.
5	What modifications were made to the system to support operation at the EPU conditions? • What failure modes or vulnerabilities were addressed by the modifications? • Did the modifications (as implemented) create any new failure modes or vulnerabilities? • Were you satisfied with the modification as implemented? • Did the modifications (as implemented) adequately address the existing material condition issues? • Has system performance post-modification been satisfactory	Feedwater level control is being reviewed by a separate system team. Digital feedwater modification is outside the scope of the EPU. The range for the feedwater flow measurement was increased to 7000 gpm whereas the flow through each pump at full power decreased to about 4000 gpm. The important point to note here is that even though the normal 100% flow of feedwater per pump decreased for EPU; the span of the instrumentation was increased to support the digital feedwater modification.

Systems:		
	Question	Response
6	Provide additional information on condensate pumps	**Additional information regarding condensate system pumps:** The interviewees included the Exelon rotating equipment engineer, a consultant familiar with Dresden/Quad pumps, and the Dresden pump engineer. Information regarding not only the feedwater pumps but pumps in other systems was solicited. The following information pertains to the condensate and condensate booster pumps: The 3C condensate pump impeller at Dresden was recently found to have cavitation damage. The damage initiated from the back side of the impeller and propagated through the wall with several through wall holes. The location of the damage is consistent with internal recirculation cavitation rather than inadequate NPSH. The 3C impeller had been in service less than two years. There are several reported differences between the 3C pump and the other condensate pumps in Unit 3. Cavitation noise was reported to be louder for the 3C than for the other pumps. The other pump impellers were inspected earlier in their post-EPU life and those inspections were not as detailed. (The damage on the back side of the impeller would not be visible without a boroscope exam.) The 3C impeller material is ASTM 296A Grade -CF-8M The material of the other pump impellers is the same. To address concerns with potentially unstable operating conditions testing of the Dresden condensate booster pumps in various configurations was conducted. This testing was conducted with specially installed proximity probes and existing motor current meters to measure shaft axial motion. The testing established conditions of stable and unstable pump operation, resulting in the development of guidance for recommended operating zones for the booster pumps. Unstable operation, based on proximity probe

Systems:		
	Question	Response
		measurements was inferred in low flow regions that are susceptible for internal recirculation and associated cavitation. This testing allowed Dresden to reduce the minimum current used to ensure proper operation of the condensate booster pump from 180 amps to 160 amps. No similar testing has been conducted for the condensate pumps nor for the Quad Cities condensate booster pumps which therefore still have a 180-amp limit. Terry Hoffman has reported that the condensate booster pumps wear rings at Dresden are getting eaten up and look like volcanic rock, indicative of inadequate minimum flow. The plant used to be able to select appropriate combinations of CD/CB pumps "to keep the stronger pump(s) from running the weaker pump(s) back on their pump curve"; however, with all pumps in operation, this no longer can be done. This may suggest that some activity may be needed to balance the flow of the pumps more accurately.

Procedures:		
	Question	Response
1	Were operating procedures changed to support operation at EPU conditions? • What failure modes or vulnerabilities were addressed by the procedure changes? • Did the procedure changes create any new failure modes, vulnerabilities or operator challenges? • Were you satisfied with the procedure changes as implemented? • Has system operation with the revised procedures been satisfactory?	Operating procedures were modified to address three pump operation. The current flow point for starting the third feedwater pump is 9.8Mlb/hr, the same as it was prior to EPU. For a short time Immediately following implementation of EPU, this flow point was changed to 9Mlb/hr based on Dresden FWRVs becoming unstable due to a sensitive tuning issue which is being resolved. . The SE noted that it may be possible to operate on two pumps even above 9.8Mlb/hr, and noted that no guidance is currently provide to operations as to when to reduce the number of operating pumps following a drop in plant load. Operations at Quad Cities has not accepted use of the automatic control of the minimum flow line downstream of the condensate booster pumps, since operational testing of this control logic was never successfully tested. The minimum flow line is manually closed at full power. There have been several instances of lifting of the thermal relief valves downstream of the booster pump with resulting flooding of the adjacent spaces (LP heater bay area and down cable trays). (e.g. following a scram and trip of the feedwater pumps. The main impact to the Condensate pumps is operation at no-flow (dead head) conditions.
2	Were PM procedures for systems, components and subcomponents revised to reflect, with regards to frequency and scope, accelerated wear and degradation? • Which PMs were revised? • What was the basis for the changes?	PMs have been revised several times to respond to changing strategies relative of PM work on-line (with derate), during scheduled refueling outages and during planned partial load reductions. The PM frequencies of the feedwater system components have not been affected by EPU other than the feed

Procedures:		
	Question	Response
	• Are there additional PM changes necessary to ensure system and component reliability at the EPU conditions?	pumps are now on a 12-year overhaul period. The RFP motors did not originally have a PM cleaning frequency (i.e. not cleaned in 30 years). The current PM cleaning frequency is 10 years. 2C has now been refurbished. Q2R17 will refurbish 2A, then Q1R18, etc.)The feedwater heater inspection rate is 2 heaters per outage. Some feedwater heaters have never been opened and are scheduled next. The pressure test is performed every outage and no major leaks have been reported. The PM frequency on the seals is 8 years for the condensate system and 6 years for the feedwater system. The mechanical coupling between the gearbox and the pump has been changed out; the coupling between the motor and the gearbox will be changed out for the first time during motor refurbishment.

Material Condition:		
	Question	Response
1	What are the material condition issues associated with the components in the system?	Quad Cities U1 feed pumps: A pump is the strongest. In U2, all three feed pumps are approximately the same. Currently Quad Cities have two relief valves (setpoints 450 psi) in the unit 2 low pressure heaters bays (booster pumps area) that are leaking based on thermography. These relief valves will be replaced during the next outage. Quad Cities had a lift of the high pressure heater relief valve D heater (setpoint 1900 psi) last Monday – cause of lifting has not been established – this one had also showed as having leakage based on thermography. It will be sent out to be inspected. Vibes on high pressure heaters are not severe and are not noticeably affected by EPU. The Quad Cities FW pump casing vent and drain (manual) valves run to failure; however, three valves have been leaking with an increasing leakage rate QC drains differ from Dresden's which have 2" drain lines with single isolation. Cooling lines for the Dresden feed pumps have cracked several times both pre-EPU and post-EPU. These lines are being replaced with flexible hose connections. Unit 2 flex lines are installed; Unit 3 installation is planned for this outage. At Dresden two feed pump sway braces have failed – one prior to EPU and one post-EPU. At Dresden the feed pump suction line pressure gage has always been noisy (exhibiting fluctuating pressure)

Material Condition:		
	Question	Response
		At Dresden there may be a potential for additional failures of components due to a perceived increase in vibrations on the Feedwater Pump Lube Oil System. Recently, a temperature control valve failed to respond properly. There was no intermediate positioning of the valve; either full open or full closed (overactive). This may be a problem with a bad relay. The plan is to replace the temperature control valve. At Dresden there was a high cycle fatigue failure of 2B feed pump oil cooling line attributed to operating in a dead head condition for 24 hours. This is not EPU related. However, overall the feedwater oil piping and instruments (pressure, temperature switches) attached to the skid may be susceptible to increased vibrations.
2	Does operation at EPU conditions exacerbate any of these material condition issues?	The previously identified weak pump (1A) may be further stressed by the higher discharge pressure provide by the other feedwater pumps.
3	Was accelerated degradation of components with material condition issues adequately addressed by the EPU Project?	Potential for already weak individual pumps to be pushed further up their head/flow curve was not addressed by the EPU task reports.
4	What are critical EPU-related spare parts issues?	EPU operation requires all three feed pumps to be operating at full power. Spare parts must be available for replacement of pump parts that fail in order to minimize plant downtime or minimize derate associated with pump part failures. The system engineer identified two critical parts that have no spares: the gear box speed increasers and the pump motors. Neither QC nor Dresden has spares for these parts. This is a known vulnerability point. The CRD pumps are in a similar situation.

8. Database Report (Project Instruction Attachment 1)

A component database report is attached.

EPU Extent of Condition Review

Master/Sub Component

Dresden

System 32 Feedwater (FW)

Master Component Type	Master EPN/ECode	Equipment Name
Piping Specialty,Fittings (P20)	*EPU-DRE-32-0001 300000053*	*FEEDWATER PIPING*

Cmp EPN		Equipment Name	Sub Cmp Type
EPU-DRE-32-0001	Mstr Cmp	FEEDWATER PIPING	Piping Specialty,Fittings (P20)

Master Component Type	Master EPN/ECode	Equipment Name
Piping Specialty,Fittings (P20)	*EPU-DRE-32-0002 300000125*	*FEEDWATER MIN FLOW PIPING*

Cmp EPN		Equipment Name	Sub Cmp Type
EPU-DRE-32-0002	Mstr Cmp	FEEDWATER MIN FLOW PIPING	Piping Specialty,Fittings (P20)

Master Component Type	Master EPN/ECode	Equipment Name
Probs, Temp, Levels, Sampling (P27)	*EPU-DRE-32-0003 300000184*	*PROBE FEEDWATER INJECTION/SAMPLE*

Cmp EPN		Equipment Name	Sub Cmp Type
EPU-DRE-32-0003	Mstr Cmp	PROBE FEEDWATER INJECTION/SAMPLE	Probs, Temp, Levels, Sampling (P27)

Master Component Type	Master EPN/ECode	Equipment Name
Pump (P30)	*2-3201-A 0000801241*	*(PUMP) 2A REACTOR FEED PUMP*

Cmp EPN		Equipment Name	Sub Cmp Type
2-3201-A	Sub Cmp	(GEAR) 2A REACTOR FEED PUMP	Gear (G23)
2-3201-A	Sub Cmp	2A REACTOR FEED PUMP SPEED INCREASING GEAR	Gear Reducer (G34)
2-3201-A	Sub Cmp	(MOTOR) 2A REACTOR FEED PUMP	Motor (M10)
2-3201-A	Mstr Cmp	(PUMP) 2A REACTOR FEED PUMP	Pump (P30)
2-3201-A	Sub Cmp	2-3201A 2A REACTOR FEED PUMP	Pump Assembly (PMPA)
2-3201-B	Sub Cmp	GEAR REACTOR FEED PUMP #B	Gear (G23)
2-3201-B	Sub Cmp	2B REACTOR FEED PUMP SPEED INCREASING GEAR	Gear Reducer (G34)
2-3201-B	Sub Cmp	(MOTOR) 2B REACTOR FEED PUMP	Motor (M10)
2-3201-B	Sub Cmp	(PUMP) 2B REACTOR FEED PUMP	Pump (P30)
2-3201-B	Sub Cmp	2-3201B 2B REACTOR FEED PUMP	Pump Assembly (PMPA)
2-3201-C	Sub Cmp	GEAR REACTOR FEED PUMP #C	Gear (G23)
2-3201-C	Sub Cmp	2C REACTOR FEED PUMP SPEED INCREASING GEAR	Gear Reducer (G34)
2-3201-C	Sub Cmp	(MOTOR) 2C REACTOR FEED PUMP	Motor (M10)
2-3201-C	Sub Cmp	(PUMP) 2C REACTOR FEED PUMP	Pump (P30)
2-3201-C	Sub Cmp	2-3201C 2C REACTOR FEED PUMP	Pump Assembly (PMPA)
3-3201-A	Sub Cmp	GEAR REACTOR FEED PUMP #A	Gear (G23)
3-3201-A	Sub Cmp	3A REACTOR FEED PUMP SPEED INCREASING GEAR	Gear Reducer (G34)
3-3201-A	Sub Cmp	MOTOR 3-3201A 3A REACTOR FEED PUMP	Motor (M10)
3-3201-A	Sub Cmp	3-3201A 3A REACTOR FEED PUMP	Pump (P30)
3-3201-A	Sub Cmp	3-3201A 3A REACTOR FEED PUMP	Pump Assembly (PMPA)
3-3201-B	Sub Cmp	GEAR REACTOR FEED PUMP #B	Gear (G23)
3-3201-B	Sub Cmp	3B REACTOR FEED PUMP SPEED INCREASING GEAR	Gear Reducer (G34)
3-3201-B	Sub Cmp	MOTOR 3-3201B 3B REACTOR FEED PUMP	Motor (M10)
3-3201-B	Sub Cmp	3-3201B 3B REACTOR FEED PUMP	Pump (P30)
3-3201-B	Sub Cmp	3-3201B 3B REACTOR FEED PUMP	Pump Assembly (PMPA)
3-3201-C	Sub Cmp	GEAR REACTOR FEED PUMP #C	Gear (G23)
3-3201-C	Sub Cmp	3C REACTOR FEED PUMP SPEED INCREASING GEAR	Gear Reducer (G34)
3-3201-C	Sub Cmp	(MOTOR) 3C REACTOR FEED PUMP	Motor (M10)
3-3201-C	Sub Cmp	3-3201C 3C REACTOR FEED PUMP	Pump (P30)
3-3201-C	Sub Cmp	3-3201C 3C REACTOR FEED PUMP	Pump Assembly (PMPA)

Master Component Type	Master EPN/ECode	Equipment Name
Temp Thermowell (TW)	*EPU-DRE-32-0004 300000297*	*FEEDWATER TERMOWELLS*

Cmp EPN		Equipment Name	Sub Cmp Type
EPU-DRE-32-0004	Mstr Cmp	FEEDWATER TERMOWELLS	Temp Thermowell (TW)

Master Component Type	Master EPN/ECode	Equipment Name
Valves - Air Operated (V05)	*2-3201-A 0000800860*	*2A RFP MIN FLOW VLV*

Cmp EPN		Equipment Name	Sub Cmp Type

EPU Extent of Condition Review

Master/Sub Component

Dresden

System 32 Feedwater (FW)

Master Component Type	Master EPN/ECode	Equipment Name
Valves - Air Operated (V05)	*2-3201-A 0000800860*	*2A RFP MIN FLOW VLV*

Cmp EPN		Equipment Name	Sub Cmp Type
2-3201-A	Sub Cmp	(ACTUATOR) 2A RFP MIN FLOW VLV	Actuator,Air (Valves) (A12)
2-3201-A	Sub Cmp	(ASSY) 2A RFP MIN FLOW VLV	Air Operated Valve Assembly (AOVA)
2-3201-A	Sub Cmp	2A RFP MIN FLOW VLV SO VLV	Solenoids (S45)
2-3201-A	Sub Cmp	SWITCH LIMIT VLV CLOSED FOR PCV-2-3201-A	Special Switch Closed (ZSC)
2-3201-A	Sub Cmp	SWITCH LIMIT VLV OPEN FOR PCV-2-3201-A	Special Switch Open (ZSO)
2-3201-A	Mstr Cmp	2A RFP MIN FLOW VLV	Valves - Air Operated (V05)
2-3201-B	Sub Cmp	(ACTUATOR) B RFP MIN FLOW VLV	Actuator,Air (Valves) (A12)
2-3201-B	Sub Cmp	(ASSY) 2B RFP MIN FLOW VLV	Air Operated Valve Assembly (AOVA)
2-3201-B	Sub Cmp	2B RFP MIN FLOW VLV SO VLV	Solenoids (S45)
2-3201-B	Sub Cmp	SWITCH LIMIT VLV CLOSED FOR PCV-2-3201-B	Special Switch Closed (ZSC)
2-3201-B	Sub Cmp	SWITCH LIMIT VLV OPEN FOR PCV-2-3201-B	Special Switch Open (ZSO)
2-3201-B	Sub Cmp	2B RFP MIN FLOW VLV	Valves - Air Operated (V05)
2-3201-C	Sub Cmp	(ACTUATOR) 2C RFP MIN FLOW VLV	Actuator,Air (Valves) (A12)
2-3201-C	Sub Cmp	(ASSY) 2C RFP MIN FLOW VLV	Air Operated Valve Assembly (AOVA)
2-3201-C	Sub Cmp	2C RFP MIN FLOW VLV SO VLV	Solenoids (S45)
2-3201-C	Sub Cmp	SWITCH LIMIT VLV CLOSED FOR PCV-2-3201-C	Special Switch Closed (ZSC)
2-3201-C	Sub Cmp	SWITCH LIMIT VLV OPEN FOR PCV-2-3201-C	Special Switch Open (ZSO)
2-3201-C	Sub Cmp	2C RFP MIN FLOW VLV	Valves - Air Operated (V05)
3-3201-A	Sub Cmp	(ACTUATOR) A RFP MIN FLOW VLV	Actuator,Air (Valves) (A12)
3-3201-A	Sub Cmp	(ASSY) 3A RFP MIN FLOW VLV	Air Operated Valve Assembly (AOVA)
3-3201-A	Sub Cmp	3A RFP MIN FLOW SO VLV	Solenoids (S45)
3-3201-A	Sub Cmp	SWITCH LIMIT VLV CLSD FOR PCV 3-3201-A	Special Switch Closed (ZSC)
3-3201-A	Sub Cmp	SWITCH LIMIT VLV OPEN FOR PCV 3-3201-A	Special Switch Open (ZSO)
3-3201-A	Sub Cmp	3A RFP MIN FLOW VLV	Valves - Air Operated (V05)
3-3201-B	Sub Cmp	(ACTUATOR) 3B RFP MIN FLOW VLV	Actuator,Air (Valves) (A12)
3-3201-B	Sub Cmp	(ASSY) 3B RFP MIN FLOW VLV	Air Operated Valve Assembly (AOVA)
3-3201-B	Sub Cmp	3B RFP MIN FLOW SO VLV	Solenoids (S45)
3-3201-B	Sub Cmp	SWITCH LIMIT CLSD ON PCV-3-3201B	Special Switch Closed (ZSC)
3-3201-B	Sub Cmp	SWITCH LIMIT OPEN ON PCV-3-3201B	Special Switch Open (ZSO)
3-3201-B	Sub Cmp	3B RFP MIN FLOW VLV	Valves - Air Operated (V05)
3-3201-C	Sub Cmp	(ACTUATOR) 3C RFP MIN FLOW VLV	Actuator,Air (Valves) (A12)
3-3201-C	Sub Cmp	(ASSY) 3C RFP MIN FLOW VLV	Air Operated Valve Assembly (AOVA)
3-3201-C	Sub Cmp	3C RFP MIN FLOW SO VLV	Solenoids (S45)
3-3201-C	Sub Cmp	SWITCH LIMIT CLSD ON PCV-3-3201C	Special Switch Closed (ZSC)
3-3201-C	Sub Cmp	SWITCH LIMIT OPEN ON PCV-3-3201C	Special Switch Open (ZSO)
3-3201-C	Sub Cmp	3C RFP MIN FLOW VLV	Valves - Air Operated (V05)

Master Component Type	Master EPN/ECode	Equipment Name
Valves - Air Operated (V05)	*EPU-DRE-32-0005 300000418*	*FEEDWATER REG VALVE*

Cmp EPN		Equipment Name	Sub Cmp Type
EPU-DRE-32-0005	Mstr Cmp	FEEDWATER REG VALVE	Valves - Air Operated (V05)

Master Component Type	Master EPN/ECode	Equipment Name
Valves - Motor Operated (V20)	*2-3201-A 0000800875*	*2A RFP DISCH VLV*

Cmp EPN		Equipment Name	Sub Cmp Type
2-3201-A	Sub Cmp	2A RFP DISCH VLV	Motor (M10)
2-3201-A	Sub Cmp	ACTUATOR 2A RFP DISCH VLV	Mov Valve Operator (L05)
2-3201-A	Mstr Cmp	2A RFP DISCH VLV	Valves - Motor Operated (V20)
2-3201-B	Sub Cmp	MOTOR 2B RFP DISCH VLV	Motor (M10)
2-3201-B	Sub Cmp	2B RFP DISCH VLV	Motor Operated Valve Assembly (MOVA)
2-3201-B	Sub Cmp	ACTUATOR 2B RFP DISCH VLV	Mov Valve Operator (L05)
2-3201-B	Sub Cmp	2B RFP DISCH VLV	Valves - Motor Operated (V20)
2-3201-C	Sub Cmp	MOTOR 2C RFP DISCH VLV	Motor (M10)
2-3201-C	Sub Cmp	2C RFP DISCH VLV	Motor Operated Valve Assembly (MOVA)
2-3201-C	Sub Cmp	ACTUATOR 2C RFP DISCH VLV	Mov Valve Operator (L05)
2-3201-C	Sub Cmp	2C RFP DISCH VLV	Valves - Motor Operated (V20)
3-3201-A	Sub Cmp	MOTOR 3A RFP DISCH VLV	Motor (M10)
3-3201-A	Sub Cmp	3A RFP DISCH VLV	Motor Operated Valve Assembly (MOVA)

EPU Extent of Condition Review

Master/Sub Component

Dresden

System 32 Feedwater (FW)

Master Component Type	Master EPN/ECode	Equipment Name
Valves - Motor Operated (V20)	*2-3201-A 0000800875*	*2A RFP DISCH VLV*

Cmp EPN		Equipment Name	Sub Cmp Type
3-3201-A	Sub Cmp	(ACTUATOR) 3A RFP DISCH VLV	Mov Valve Operator (L05)
3-3201-A	Sub Cmp	3A RFP DISCH VLV	Valves - Motor Operated (V20)
3-3201-B	Sub Cmp	MOTOR 3B RFP DISCH VLV	Motor (M10)
3-3201-B	Sub Cmp	3B RFP DISCH VLV	Motor Operated Valve Assembly (MOVA)
3-3201-B	Sub Cmp	(ACTUATOR) 3B RFP DISCH VLV	Mov Valve Operator (L05)
3-3201-B	Sub Cmp	3B RFP DISCH VLV	Valves - Motor Operated (V20)
3-3201-C	Sub Cmp	MOTOR 3C RFP DISCH VLV	Motor (M10)
3-3201-C	Sub Cmp	3C RFP DISCH VLV	Motor Operated Valve Assembly (MOVA)
3-3201-C	Sub Cmp	(ACTUATOR) 3C RFP DISCH VLV	Mov Valve Operator (L05)
3-3201-C	Sub Cmp	3C RFP DISCH VLV	Valves - Motor Operated (V20)

Master Component Type	Master EPN/ECode	Equipment Name
Valves - Motor Operated (V20)	*2-3202-A 0000800889*	*2D1 HP HTR FW INLET VLV*

Cmp EPN		Equipment Name	Sub Cmp Type
2-3202-A	Sub Cmp	MOTOR 2D1 HP HTR FW INLET VLV	Motor (M10)
2-3202-A	Sub Cmp	2D1 HP HTR FW INLET VLV	Motor Operated Valve Assembly (MOVA)
2-3202-A	Sub Cmp	ACTUATOR 2D1 HP HTR FW INLET VLV	Mov Valve Operator (L05)
2-3202-A	Mstr Cmp	2D1 HP HTR FW INLET VLV	Valves - Motor Operated (V20)
2-3202-B	Sub Cmp	MOTOR 2D2 HP HTR FW INLET VLV	Motor (M10)
2-3202-B	Sub Cmp	2D2 HP HTR FW INLET VLV	Motor Operated Valve Assembly (MOVA)
2-3202-B	Sub Cmp	ACTUATOR 2D2 HP HTR FW INLET VLV	Mov Valve Operator (L05)
2-3202-B	Sub Cmp	2D2 HP HTR FW INLET VLV	Valves - Motor Operated (V20)
2-3202-C	Sub Cmp	MOTOR 2D3 HP HTR FW INLET VLV	Motor (M10)
2-3202-C	Sub Cmp	2D3 HP HTR FW INLET VLV	Motor Operated Valve Assembly (MOVA)
2-3202-C	Sub Cmp	ACTUATOR 2D3 HP HTR FW INLET VLV	Mov Valve Operator (L05)
2-3202-C	Sub Cmp	2D3 HP HTR FW INLET VLV	Valves - Motor Operated (V20)
2-3204-A	Sub Cmp	MOTOR 2D1 HP HTR FW OUTLET VLV	Motor (M10)
2-3204-A	Sub Cmp	2D1 HP HTR FW OUTLET VLV	Motor Operated Valve Assembly (MOVA)
2-3204-A	Sub Cmp	ACTUATOR 2D1 HP HTR FW OUTLET VLV	Mov Valve Operator (L05)
2-3204-A	Sub Cmp	2D1 HP HTR FW OUTLET VLV	Valves - Motor Operated (V20)
2-3204-B	Sub Cmp	MOTOR 2D2 HP HTR FW OUTLET VLV	Motor (M10)
2-3204-B	Sub Cmp	2D2 HP HTR FW OUTLET VLV	Motor Operated Valve Assembly (MOVA)
2-3204-B	Sub Cmp	ACTUATOR 2D2 HP HTR FW OUTLET VLV	Mov Valve Operator (L05)
2-3204-B	Sub Cmp	2D2 HP HTR FW OUTLET VLV	Valves - Motor Operated (V20)
2-3204-C	Sub Cmp	MOTOR 2D3 HP HTR FW OUTLET VLV	Motor (M10)
2-3204-C	Sub Cmp	2D3 HP HTR FW OUTLET VLV	Motor Operated Valve Assembly (MOVA)
2-3204-C	Sub Cmp	ACTUATOR 2D3 HP HTR FW OUTLET VLV	Mov Valve Operator (L05)
2-3204-C	Sub Cmp	2D3 HP HTR FW OUTLET VLV	Valves - Motor Operated (V20)
3-3202-A	Sub Cmp	MOTOR 3D1 HP HTR FW INLET VLV	Motor (M10)
3-3202-A	Sub Cmp	3D1 HP HTR FW INLET VLV	Motor Operated Valve Assembly (MOVA)
3-3202-A	Sub Cmp	ACTUATOR 3D1 HP HTR FW INLET VLV	Mov Valve Operator (L05)
3-3202-A	Sub Cmp	3D1 HP HTR FW INLET VLV	Valves - Motor Operated (V20)
3-3202-B	Sub Cmp	MOTOR 3D2 HP HTR FW INLET VLV	Motor (M10)
3-3202-B	Sub Cmp	3D2 HP HTR FW INLET VLV	Motor Operated Valve Assembly (MOVA)
3-3202-B	Sub Cmp	ACTUATOR 3D2 HP HTR FW INLET VLV	Mov Valve Operator (L05)
3-3202-B	Sub Cmp	3D2 HP HTR FW INLET VLV	Valves - Motor Operated (V20)
3-3202-C	Sub Cmp	MOTOR 3D3 HP HTR FW INLET VLV	Motor (M10)
3-3202-C	Sub Cmp	3D3 HP HTR FW INLET VLV	Motor Operated Valve Assembly (MOVA)
3-3202-C	Sub Cmp	ACTUATOR 3D3 HP HTR FW INLET VLV	Mov Valve Operator (L05)
3-3202-C	Sub Cmp	3D3 HP HTR FW INLET VLV	Valves - Motor Operated (V20)
3-3204-A	Sub Cmp	MOTOR 3D1 HP HTR FW OUTLET VLV	Motor (M10)
3-3204-A	Sub Cmp	3D1 HP HTR FW OUTLET VLV	Motor Operated Valve Assembly (MOVA)
3-3204-A	Sub Cmp	ACTUATOR 3D1 HP HTR FW OUTLET VLV	Mov Valve Operator (L05)
3-3204-A	Sub Cmp	3D1 HP HTR FW OUTLET VLV	Valves - Motor Operated (V20)
3-3204-B	Sub Cmp	MOTOR 3D2 HP HTR FW OUTLET VLV	Motor (M10)
3-3204-B	Sub Cmp	3D2 HP HTR FW OUTLET VLV	Motor Operated Valve Assembly (MOVA)
3-3204-B	Sub Cmp	ACTUATOR 3D2 HP HTR FW OUTLET VLV	Mov Valve Operator (L05)
3-3204-B	Sub Cmp	3D2 HP HTR FW OUTLET VLV	Valves - Motor Operated (V20)
3-3204-C	Sub Cmp	MOTOR 3D3 HP HTR FW OUTLET VLV	Motor (M10)

EPU Extent of Condition Review

Master/Sub Component

Dresden

System 32 Feedwater (FW)

Master Component Type	Master EPN/ECode	Equipment Name
Valves - Motor Operated (V20)	*2-3202-A 0000800889*	*2D1 HP HTR FW INLET VLV*

Cmp EPN		Equipment Name	Sub Cmp Type
3-3204-C	Sub Cmp	3D3 HP HTR FW OUTLET VLV	Motor Operated Valve Assembly (MOVA)
3-3204-C	Sub Cmp	ACTUATOR 3D3 HP HTR FW OUTLET VLV	Mov Valve Operator (L05)
3-3204-C	Sub Cmp	3D3 HP HTR FW OUTLET VLV	Valves - Motor Operated (V20)

Master Component Type	Master EPN/ECode	Equipment Name
Valves - Motor Operated (V20)	*2-3203 0000800902*	*U2 'D' HEATERS BYP VLV*

Cmp EPN		Equipment Name	Sub Cmp Type
2-3203	Sub Cmp	MOTOR U2 'D' HEATERS BYP VLV	Motor (M10)
2-3203	Sub Cmp	U2 'D' HEATERS BYP VLV	Motor Operated Valve Assembly (MOVA)
2-3203	Sub Cmp	ACTUATOR U2 'D' HEATERS BYP VLV	Mov Valve Operator (L05)
2-3203	Mstr Cmp	U2 'D' HEATERS BYP VLV	Valves - Motor Operated (V20)
3-3203	Sub Cmp	MOTOR 3D HP HTRS BYP VLV	Motor (M10)
3-3203	Sub Cmp	3D HP HTRS BYP VLV	Motor Operated Valve Assembly (MOVA)
3-3203	Sub Cmp	ACTUATOR 3D HP HTRS BYP VLV	Mov Valve Operator (L05)
3-3203	Sub Cmp	3D HP HTRS BYP VLV	Valves - Motor Operated (V20)

Master Component Type	Master EPN/ECode	Equipment Name
Valves - Motor Operated (V20)	*2-3205-A 0000800918*	*2A FW INLET ISOL VLV (MOV)*

Cmp EPN		Equipment Name	Sub Cmp Type
2-3205-A	Sub Cmp	(MOTOR) 2A FW INLET ISOL VLV (MOV)	Motor (M10)
2-3205-A	Sub Cmp	2A FW INLET ISOL VLV (MOV)	Motor Operated Valve Assembly (MOVA)
2-3205-A	Sub Cmp	(ACTUATOR) 2A FW INLET ISOL VLV (MOV)	Mov Valve Operator (L05)
2-3205-A	Mstr Cmp	2A FW INLET ISOL VLV (MOV)	Valves - Motor Operated (V20)
2-3205-B	Sub Cmp	(MOTOR) 2B FW INLET ISOL VLV (MOV)	Motor (M10)
2-3205-B	Sub Cmp	2B FW INLET ISOL VLV (MOV)	Motor Operated Valve Assembly (MOVA)
2-3205-B	Sub Cmp	(ACTUATOR) 2B FW INLET ISOL VLV (MOV)	Mov Valve Operator (L05)
2-3205-B	Sub Cmp	2B FW INLET ISOL VLV (MOV)	Valves - Motor Operated (V20)
2-3206-A	Sub Cmp	MOTOR 2A FW REG VLV INLET SV	Motor (M10)
2-3206-A	Sub Cmp	2A FW REG VLV INLET SV	Motor Operated Valve Assembly (MOVA)
2-3206-A	Sub Cmp	ACTUATOR 2A FW REG VLV INLET SV	Mov Valve Operator (L05)
2-3206-A	Sub Cmp	2A FW REG VLV INLET SV	Valves - Motor Operated (V20)
2-3206-B	Sub Cmp	MOTOR 2B FW REG VLV INLET SV	Motor (M10)
2-3206-B	Sub Cmp	2B FW REG VLV INLET SV	Motor Operated Valve Assembly (MOVA)
2-3206-B	Sub Cmp	(ACTUATOR) U2 B FW REG VLV INLET ISOL VLV	Mov Valve Operator (L05)
2-3206-B	Sub Cmp	U2 B FW REG VLV INLET ISOL VLV	Valves - Motor Operated (V20)
3-3205-A	Sub Cmp	MOTOR 3A FW INLET ISOL VLV	Motor (M10)
3-3205-A	Sub Cmp	3A FW INLET ISOL VLV	Motor Operated Valve Assembly (MOVA)
3-3205-A	Sub Cmp	(ACTUATOR) 3A FW INLET ISOL VLV	Mov Valve Operator (L05)
3-3205-A	Sub Cmp	3A FW INLET ISOL VLV	Valves - Motor Operated (V20)
3-3205-B	Sub Cmp	MOTOR 3B FW INLET ISOL VLV	Motor (M10)
3-3205-B	Sub Cmp	3B FW INLET ISOL VLV	Motor Operated Valve Assembly (MOVA)
3-3205-B	Sub Cmp	(ACTUATOR) 3B FW INLET ISOL VLV	Mov Valve Operator (L05)
3-3205-B	Sub Cmp	3B FW INLET ISOL VLV	Valves - Motor Operated (V20)
3-3206-A	Sub Cmp	MOTOR 3A FW REG VLV INLET SV	Motor (M10)
3-3206-A	Sub Cmp	3A FW REG VLV INLET SV	Motor Operated Valve Assembly (MOVA)
3-3206-A	Sub Cmp	(ACTUATOR) 3A FW REG VLV INLET SV	Mov Valve Operator (L05)
3-3206-A	Sub Cmp	3A FW REG VLV INLET SV	Valves - Motor Operated (V20)
3-3206-B	Sub Cmp	MOTOR 3B FW REG VLV INLET SV	Motor (M10)
3-3206-B	Sub Cmp	3B FW REG VLV INLET SV	Motor Operated Valve Assembly (MOVA)
3-3206-B	Sub Cmp	(ACTUATOR) 3B FW REG VLV INLET SV	Mov Valve Operator (L05)
3-3206-B	Sub Cmp	3B FW REG VLV INLET SV	Valves - Motor Operated (V20)

Master Component Type	Master EPN/ECode	Equipment Name
Valves - No Operators (V25)	*2-3208-A 0000800969*	*2A RFP DISCH CK VLV*

Cmp EPN		Equipment Name	Sub Cmp Type
2-3208-A	Mstr Cmp	2A RFP DISCH CK VLV	Valves - No Operators (V25)
2-3208-B	Sub Cmp	2B RFP DISCH CK VLV	Valves - No Operators (V25)

EPU Extent of Condition Review

Master/Sub Component

Dresden

System 32 Feedwater (FW)

Master Component Type	Master EPN/ECode	Equipment Name
Valves - No Operators (V25)	*2-3208-A* *0000800969*	*2A RFP DISCH CK VLV*

Cmp EPN		Equipment Name	Sub Cmp Type
2-3208-C	Sub Cmp	2C RFP DISCH CK VLV	Valves - No Operators (V25)
3-3208-A	Sub Cmp	3A RFP DISCH CK VLV	Valves - No Operators (V25)
3-3208-B	Sub Cmp	3B RFP DISCH CK VLV	Valves - No Operators (V25)
3-3208-C	Sub Cmp	3C RFP DISCH CK VLV	Valves - No Operators (V25)

EPU Extent of Condition Review

Vulnerability of Components/Subcomponents

Dresden
System 32 Feedwater (FW)

System Parameter

Parameter	Analytical Pre-EPU Level	Analytical Post-EPU Level	System Design Basis Value	Limiting Pre-EPU Plant Data	Limiting Post-EPU Plant Data
Flow - Total FW System (Mlbs/hr)	9.72	11.71	11.71	9.8	11.32

Related Vulnerabilities of Components (See Master/Sub Component Report for listing of all components evaluated)

Component Type	Failure Vulnerability:	Critical Characteristic	Expect Value for Critical Characteristic	Actual Value for Critical Characteristic	Limit	Failure Mechanism	GAP	Risk Priority
Probs, Temp, Levels, Sampling	De-rate	Vibration	N/A	N/A	N/A	Wear	Yes	H

Master EPN : EPU-DRE-32-0003 PROBE FEEDWATER INJECTION/SAMPLE

Rec: For any remaining original probes, replace the feedwater sample probes

Comments: BASIS OF RECOMMENDATION
In response to several concerns, one of which would be the damage to the Dresden feedwater sparger. The shorter design should be less subject to failure. Dresden had sample probe failures.Three FW probes and one condensate sample probe failed due to high cycle fatigue induced by flow-induced vibration. GE-NE-000-0024-2731 determined ratio of Vortex Frequency for various sample probes but did not include thermowells. GE-NE-000-0024-2731 determined frequency ratios of 1.026, 0.87, 0.86, 0.3, 0.9 and 0.71 for various sample probe cases. Per ASME Appendix N - Below 0.76 acceptable however other documents more restrictive. GE-NE-000-0024-2731 has determined that for certain sample probe cases, the vortex shedding frequency is sufficiently close to a sample probe first natural frequency during various operating conditions that lock-on could occur (i.e. resonant condition). GE-NE-000-0024-2731 did not include thermowells ·Continued successful operation of existing probes or thermowells for an entire operating cycle cannot be assured. If it is determined by analysis that the vortex shedding frequency is sufficiently close to a thermowell's first natural frequency during various operating conditions that lock-on could occur, then the thermowell shall be re-designed.
RISK
Probe and thermowell failure could occur and either enter a feedwater heater and become lodged or enter a feedwater pump and possibly cause pump damage. A failure of a thermowell will also likely cause a breach in the system. In both cased, a unit derate or shutdown may be required.

Pump	De-rate	Number of Pumps	3	3	3	Wear	Yes	H

Master EPN : 2-3201-A (PUMP) 2A REACTOR FEED PUMP

Rec: Evaluate if there is some (probably small) potential that full-range testing of pump capability might support operation at 100% EPU power with one of the pumps in standby or might support identification of cost-effective modifications (e.g., change in impeller design) that would allow restoration of the pre-EPU stand-by pump configuration. Perform testing to determine the optimum operating conditions at which to start and stop feedwater pumps for providing optimal power production and running conditions of these pumps. This would be a plant specific analysis.

Comments: BASIS OF RECOMENDATION
Per Task Report T0701, when operating with the modified high pressure turbine with only two feed and three condensate / condensate booster pumps on line, the flow passing capability of the feedwater flow control valves will become limiting for operation near 100% of 2527/2511 MWth (pre-EPU 100% power level). While at 110 % of 2527/2511 MWth, the condensate / condensate booster pump motors will exceed their full load amp ratings. In order to achieve maximum operational flexibility a third feed pump should be placed on-line prior to exceeding 100% of 2527/2511 MWth (85.5% LPU), and the fourth condensate/condensate booster pump placed on-line prior to reaching 110% of 2527/2511 MWth (94 % LPU).·These above values for placing additional pumps in service at 100% and 110% Pre-EPU power levels are also true if a feedwater or condensate / condensate booster pump is lost or removed from service which is not significantly dependent upon time of season which is based upon theoretically pump crves. Maintenance or loss of condensate, condensate booster or feedwater pumps/motors will result in unit derates during non-outage periods or to avoid derates will have to occur during outage periods. No testing has been performed to verify achievable power with less than full complement of pumps. Per discussion with system engineers, there has been no full-range testing to confirm/identify achievable power with less than the full complement of pumps. There are no spare gear boxes available that would be needed in the event of a failure.
RISK
At the minimum the plant would derate to 100% of pre-EPU conditions if a feedwater pump was placed out of service. Maintenance will result in unit derates during

non-outage periods or to avoid derates will have to occur during outage periods. This require outage resources.

Pump	De-rate	Vibration	N/A	N/A	N/A	Wear	Yes	M

Master EPN : 2-3201-A (PUMP) 2A REACTOR FEED PUMP

Rec: Measure the vibrations of FW pump components.

Comments: BASIS OF RECOMMENDATION
System engineers and Exelon Vibration experts recognize noticably increased vibration (possibly flow-change or equipment line-up induced) in small-bore pipe and tubing associated with the Feedwater System
RISK
At the minimum, the plant would derate to 100% of pre-EPU conditions if a feedwater pump was placed out of service.

Temp Thermowell	De-rate	Vibration	N/A	N/A	N/A	Wear	Yes	H

Master EPN : EPU-DRE-32-0004 FEEDWATER TERMOWELLS

Rec: Evaluate Feedwater system temperature element thermowells to eliminate concern with vortex shedding failure vulnerability.

Comments: BASIS OF RECOMMENDATION
Dresden has had sample probe failures. Three FW probes and one condensate sample probe failed due to high cycle fatigue induced by flow-induced vibration. GE-NE-000-0024-2731 determined ratio of Vortex Frequency for various sample probes but did not include thermowells. GE-NE-000-0024-2731 determined frequency ratios of 1.026, 0.87, 0.86, 0.3, 0.9 and 0.71 for various sample probe cases. Per ASME Appendix N - Below 0.76 acceptable however other documents more restrictive. GE-NE-000-0024-2731 has determined that for certain sample probe cases, the vortex shedding frequency is sufficiently close to a sample probe first natural frequency during various operating conditions that lock-on could occur (i.e. resonant condition). GE-NE-000-0024-2731 did not include thermowells . Continued successful operation of existing probes or thermowells for an entire operating cycle cannot be assured. If it is determined by analysis that the vortex shedding frequency is sufficiently close to a thermowell's first natural frequency during various operating conditions that lock-on could occur, then the thermowell shall be re-designed
RISK
Probe and thermowell failure could occur and either enter a feedwater heater and enter a feedwater sparger and possibly cause damage. A failure of a thermowell will also likely cause a breach in the system. In both cased, a unit derate or shutdown may be required.

Valves - Air Operated	De-rate	Vibration	N/A	N/A	N/A	Wear	Yes	M

Master EPN : EPU-DRE-32-0005 FEEDWATER REG VALVE

Rec: Perform actuator/valve inspection for wear on both of the Feedwater Reg valves during next refuel outage.

Comments: BASIS OF RECOMMENDATION
The effects of vibration and wear at increase EPU flows cannot be easily evaluated on the sub-components of these valves. The likely cause of failure would be due to wear since operation at EPU condition of one cycle has not resulted in any fatigue failures of these valves. The most susceptible sub-components to wear would be the valves stem or the actuator parts. Quad Cities FW system engineer has noted lan increase in vibrations. The current PM inspects the valve and actuator, one each outage. However, if sub-components are now excited at EPU operating frequencies or vibration amplitudes increased, then potential wear patterns on these sub-components could be detected after one operating cycle and if significant, replaced prior to failure. Visual inspection should also be performed and documeneted to determine significant wear is detected. The goal is to replace worn components prior to failure and adjust PM cycles to control inspection/replacement frequencies of identified susceptible sub-components.
RISK
At the minimum, failure of a regulating valve would require a de-rate to repair

Valves - Motor Operated	SCRAM	Stroke Time	greater than 1 minute	N/A	no faster than 1 minute	Binding	Yes	M

Master EPN : 2-3203 U2 'D' HEATERS BYP VLV

Rec: Evaluate the capability of the motor operator to operate at the peak expected differential pressure. Review sizing calculation of bypass valve MO 3203 to ensure

adequate margin exists at the EPU operating differential pressure conditions.

Comments: BASIS FOR RECOMMENDATION
A higher flow results in a higher differential pressure for valve closure, which may impose additional loads on the motor, operated valves. It is understood that the margins for the motor operator motors are in the range of 200%. With an increased differential pressure across the MO-3203, D' HEATERS BYP VLV, the primary potential is to affect the stroke time of the valve. The assumptions made of this valve, in the transient analysis of the plants, is that the performance of the valve will be no better than a stroke time of one minute. Therefore no affect on the affect on the loss of feedwater heating event and there is no foreseen loss of margin. However, in the event of a heater tube rupture, a failure of the valve to open, along with failure of the D heater isolation valves to close due the increased differential pressure, could result in water backing up into the main turbine. In the event of high heater level due to tube rupture, the inlet and outlet isolation valves close and the bypass valve opens to prevent turbine water induction. ·Typically, motor operated valves are sized based upon design conditions (shutoff conditions that provide the maximum differential pressure at which the valves must open), however; actual operating conditions could have been used in the actuator sizing of MO 3203.· EPU has increased the operating differential pressure across MO 3203 by up to 30 psid. Adjustment of the torque switch setting may be required based upon this sizing review.
RISK
The Feedwater heater system is designed to close the inlet and outlet isolation valves and open the bypass valve based upon high heater level due to tube rupture to prevent turbine water induction. Water induction into the turbine could cause turbine failure in which a scram would likely occur prior to the water induction if the valves do not operate properly.

Valves - Motor Operated	SCRAM	Stroke Time	N/A	N/A	N/A	Binding	Yes	M

Master EPN : 2-3202-A 2D1 HP HTR FW INLET VLV

Rec: Review sizing calculation of isolation valves MO 3202 A, B, C and MO 3204 A, B, C at the EPU operating differential pressure conditions.

Comments: BASIS FOR RECOMMENDATION
The feedwater system MOVs consist of heater string isolation MO 3202A, B, C and MO 3204 A, B, C for the inlet and outlet isolation of the HP heaters strings. In addition, a bypass MOV, MO 3203, exists around the LP heater string. In the event of high heater level due to tube rupture, the inlet and outlet isolation valves close and the bypass valve opens to prevent turbine water induction. A higher flow results in a higher differential pressure for valve closure, which may impose additional loads on the motor, operated valves. It is understood that the margins for the motor operator motors are in the range of 200%. With an increased differential pressure across the valves, the primary potential is to affect the stroke time of the valve. The are no assumptions made of these valves in the transient analysis of the plants, however an assumption is made that the feedwater temperature change due to a feedwater heater isolation will take a minute or more. Therefore no affect on the affect on the loss of feedwater heating event and there is no foreseen loss of margin. However, in the event of a heater tube rupture, a failure of the valves to close could result in water backing up into the main turbine. In the event of high heater level due to tube rupture, the inlet and outlet isolation valves close and the bypass valve opens to prevent turbine water induction. ·Typically, motor operated valves are sized based upon design conditions (shutoff conditions that provide the maximum differential pressure at which the valves must open), however; actual operating conditions could have been used in the actuator sizing of MO 3202A, B, C and MO 3204 A, B, C.· EPU has increased the operating differential pressure across the D Feedwater Heaters by as much as 30 psid. Adjustment of the torque switch setting may be required based upon this sizing review.
RISK
The Feedwater heater system is designed to close the inlet and outlet isolation valves and open the bypass valve based upon high heater level due to tube rupture to prevent turbine water induction. Water induction into the turbine could cause turbine failure in which a scram would likely occur prior to the water induction if the valves do not operate properly.

Valves - Motor Operated	Unexpected Acc Degradation	Vibration	N/A	N/A	N/A	Wear	Yes	L

Master EPN : 2-3205-A 2A FW INLET ISOL VLV (MOV)

Rec: Perform actuator/motor inspection for wear on one of the 3206 MOVs during next refuel outage.

Comments: BASIS OF RECOMMENDATION
The effects of vibration and wear at increase EPU flows cannot be easily evaluated on the sub-components of these valves. The likely cause of failure would be due to wear or loosening of fasteners since operation at EPU condition of one cycle has not resulted in any fatigue failures of these valves .The most susceptible sub-components to wear would be the valves body to bonnet fasteners, valve stem or the actuator limit switches, torque switch or rotor. Body to bonnet fasteners have been found to be loose, in the past, due to vibrations. If sub-components are now excited at EPU operating frequencies or vibration amplitudes increased, then

potential wear patterns on these sub-components could be detected after one operating cycle and if significant, replaced prior to failure. Visual inspection of the valve stem external surfaces should also be performed If significant wear is detected, replaced worn components prior to failure and adjustment PM cycle to control inspection/replacement frequencies of identified susceptible sub-components.
RISK
The Feedwater Regulation isolation valves are designed to provide isolation of the control valves for out of service conditions. In the event of an unisolable leak of valve, the unit may have to, at the extreme conditions, shutdown to repair a leak

Valves - Motor Operated	Orp Work Around	Vibration	N/A	N/A	N/A	Wear	Yes	L

Master EPN : 2-3202-A 2D1 HP HTR FW INLET VLV

Rec: Perform actuator/motor inspection for wear on one of the 3202 MOVs during next refuel outage

Comments: BASIS FOR RECOMMENDATION
The effects of vibration and wear at increase EPU flows cannot be easily evaluated on the sub-components of these valves. The likely cause of failure would be due to wear since operation at EPU condition of one cycle has not resulted in any fatigue failures of these valves.·The most susceptible sub-components to wear would be the valves stem or the actuator limit switches, torque switch or rotor.·Quad Cities FW system engineer has noted loose components on hangers supporting the condensate booster piping that feeds the LP heaters on both units; perhaps typical of what could be expected from the higher flows through the 3202 valves. For the actuator/motor, the current PM inspects the actuator every 6 refuel cycles. However, if sub-components are now excited at EPU operating frequencies or vibration amplitudes increased, then potential wear patterns on these sub-components could be detected after one operating cycle and if significant, replaced prior to failure.·Visual inspection of the valve stem external surfaces should also be performed. If significant wear is detected, replaced worn components prior to failure and adjustment PM cycle to control inspection/replacement frequencies of identified susceptible sub-components.
RISK
The Feedwater heater system is designed to close the inlet and outlet isolation valves and open the bypass valve based upon high heater level due to tube rupture to prevent turbine water induction. Water induction into the turbine could cause turbine failure in which a scram would likely occur prior to the water induction if the valves do not operate properly.

Valves - Motor Operated	Unexpected Acc Degradation	Vibrations	N/A	N/A	N/A	Wear	Yes	M

Master EPN : 2-3201-A 2A RFP DISCH VLV

Rec: Perform external valve/internal operator/motor inspection for wear on one of the 3201 MOVs during next refuel outage.

Comments: BASIS OF RECOMMENDATION
The effects of vibration and wear at increase EPU flows cannot be easily evaluated on the sub-components of these valves. The likely cause of failure would be due to wear or loosening of fasteners since operation at EPU condition of one cycle has not resulted in any fatigue failures of these valves .The most susceptible sub-components to wear would be the valves body to bonnet fasteners, valve stem or the actuator limit switches, torque switch or rotor. Body to bonnet fasteners have been found to be loose, in the past, due to vibrations. If sub-components are now excited at EPU operating frequencies or vibration amplitudes increased, then potential wear patterns on these sub-components could be detected after one operating cycle and if significant, replaced prior to failure. Visual inspection of the valve stem external surfaces should also be performed If significant wear is detected, replaced worn components prior to failure and adjustment PM cycle to control inspection/replacement frequencies of identified susceptible sub-components.
RISK
The Feedwater discharge valves are designed to provide isolation of the feedwater pump for maintenance during plant operations. In the event of an unisolable leak on the discharge check valve the unit may have to, at the extreme conditions, shutdown to repair a leak. The risk associated with a failure of the valve to operate would at least require an operator work around to manually close the MOV in the event the Feedwater Pump needs to be isolated during operations

Valves - No Operators	Unexpected Acc Degradation	Flow	N/A	N/A	N/A	Wear	Yes	L

Master EPN : 2-3208-A 2A RFP DISCH CK VLV

Rec: Inspect one discharge check valve at the next outage and perform check valve monitoring or increase the frequency of the PM for the discharge check valve to manage the potential for accelerated wear on the hinge bushings.

Comments: BASIS FOR RECOMMENDATION
Due to the lower flow rate per pump there is potential for accelerated wear on the feedwater pump discharge check valve flapper hinge bushings; due to potential fluctuations in the position of the valve flapper. If this condition persists, then the PM for the discharge check valve should be increased to manage the potential for accelerated wear on the hinge bushings
RISK
Avoid the potential of reverse flowthrough the feedwater pumps, when a pump is in standby, which would require operator work around to manage feedwater flow to the reactor.

EPU Extent of Condition Review

Vulnerability of Components/Subcomponents

Dresden
System 32 Feedwater (FW)

System Parameter

Parameter	Analytical Pre-EPU Level	Analytical Post-EPU Level	System Design Basis Value	Limiting Pre-EPU Plant Data	Limiting Post-EPU Plant Data
Pressure - FW Pmp Discharge (PSIG)	N/A	1338	1850	1501	1506

Related Vulnerabilities of Components (See Master/Sub Component Report for listing of all components evaluated)

Component Type	Failure Vulnerablilty:	Critical Characteristic	Expect Value for Critical Characteristic	Actual Value for Critical Characteristic	Limit	Failure Mechanism	GAP	Risk Priority
Pump	De-rate	Pressure	N/A	N/A	N/A	Wear	Yes	M

Master EPN : 2-3201-A (PUMP) 2A REACTOR FEED PUMP

Rec: Recommend that the seal PM be increased and a review of three-pump operation be performed in order to prevent a de-rate.

Comments: BASIS OF RECOMMENDATION
The Feedwater Pumps seal performance has been reviewed in the past due to many failures. The maintenance plan specifies a replacement of the seals every eight years. CMO suggests that the failure of the seals has increased due to the increased discharge pressure due to three pump EPU operation.
RISK
At the minimum the plant would derate to 100% of pre-EPU conditions if a feedwater pump was placed out of service.

Valves - Air Operated	Orp Work Around	Vibration	N/A	N/A	N/A	Wear	Yes	L

Master EPN : 2-3201-A 2A RFP MIN FLOW VLV

Rec: Perform valve, actuator and positioner inspection for wear on the AO 3201 during the next refuel outage. Recommend overall of the valve and operator taking particular care to document as found conditions and recommend vibration analysis of the valve, particularly of the Air Operator, attached air piping, and the solenoid valve.

Comments: BASIS FOR RECOMMENDATION
There may be additional vibrations imposed on the valve based on the operating experience gained from the system engineers. It was stated that the vibrations have increased due to three feedwater pumps in operations. Note that the consequences of a failure of the valve should be minimal because the minimum flow valves fail open on a loss of air. This would provide pump protection. The Feedwater System has the capability to makeup for an inadvertent opening of a single minimum flow control valve based on the additional flow available with three feedwater pumps in operation. The CMO identified that these valves are trouble-prone to leakage past their seats. The effects of vibration and wear at EPU conditions cannot be easily evaluated on the sub-components of this valve. The likely cause of failure would be wear since operation at EPU condition for one cycle has not resulted in any fatigue failures of these valves. Note, however, that the minimum flow valve (AO 3201) is normally closed at full power operation; therefore, the number of cycles resulting in fatigue may not have occurred at EPU operating conditions.· However, if sub-component are now excited at EPU operating frequencies or vibration amplitudes increased, then potential wear patterns on the valves sub-components could be detected after one operating cycle and if significant replaced prior to failure.· Visual inspection of the valve stem external surfaces should also be performed. If significant wear is detected, replaced worn components prior to failure and adjustment PM cycle to control inspection/replacement frequencies of identified susceptible sub-components.
RISK
During a scram or recirculation runback with closure of FWRV's, the feedwater system is near or at deadhead conditions and over pressurizes without the minimum flow valve opening. Pump damage or component degradation may occur if this valve does not open.

EPU Extent of Condition Review

Vulnerability of Components/Subcomponents

Dresden
System 32 Feedwater (FW)

System Parameter

Parameter	Analytical Pre-EPU Level	Analytical Post-EPU Level	System Design Basis Value	Limiting Pre-EPU Plant Data	Limiting Post-EPU Plant Data
Pressure - FW Pmp Suction (PSIG) Min	N/A	166.7	N/A	193	175

Related Vulnerabilities of Components (See Master/Sub Component Report for listing of all components evaluated)

Component Type	Failure Vulnerability:	Critical Characteristic	Expect Value for Critical Characteristic	Actual Value for Critical Characteristic	Limit	Failure Mechanism	GAP	Risk Priority

Master EPN :

Rec:

Comments:

System Parameter

Parameter	Analytical Pre-EPU Level	Analytical Post-EPU Level	System Design Basis Value	Limiting Pre-EPU Plant Data	Limiting Post-EPU Plant Data
Temperature - FW Heater D Discharge (Max/Min) (F)	340/N/A	352/152	356/152 Transient Analysis 350 Pipe Design	343/322	351/346

Related Vulnerabilities of Components (See Master/Sub Component Report for listing of all components evaluated)

Component Type	Failure Vulnerability:	Critical Characteristic	Expect Value for Critical Characteristic	Actual Value for Critical Characteristic	Limit	Failure Mechanism	GAP	Risk Priority
Piping Specialty,Fittings	LER	Stress	352	351	350	Design Errors	Yes	L

Master EPN : EPU-DRE-32-0001 FEEDWATER PIPING

Rec: Revise Specification K-4080 o reflect a new design temperature of 356 F between the D Feedwater heaters and the Reactor.

Comments: BASIS OF RECOMMENDATION
The original design temperatures of 350 F downstream of the "D" FW heaters have been exceeded at EPU conditions. The EPU analytic temperature is 352 F in which the actual operating temperatures are just below the analytic limit (356 F).
RISK
The non-conservative temperatures in the specification can result in a design error or a design control issue.

EPU Extent of Condition Review

Vulnerability of Components/Subcomponents

Dresden
System 32 Feedwater (FW)

System Parameter

Parameter	Analytical Pre-EPU Level	Analytical Post-EPU Level	System Design Basis Value	Limiting Pre-EPU Plant Data	Limiting Post-EPU Plant Data
Temperature - FW Pmp Suction (F)	300	318	N/A	307	315

Related Vulnerabilities of Components

(See Master/Sub Component Report for listing of all components evaluated)

Component Type	Failure Vulnerability:	Critical Characteristic	Expect Value for Critical Characteristic	Actual Value for Critical Characteristic	Limit	Failure Mechanism	GAP	Risk Priority
Piping Specialty,Fittings	LER	Stress	318	315	300	Design Errors	Yes	L

Master EPN : EPU-DRE-32-0002 FEEDWATER MIN FLOW PIPING

Rec: Revise Specification K-4080to reflect a new design temperature of 318 F between the Feedwater Pump minimum flow line and the condenser.

Comments: BASIS OF RECOMMENDATION
The original design temperatures of 300 F downstream of the Feedwater Pump minimum flow valves have been exceeded at EPU conditions.
RISK
The non-conservative temperatures in the specification can result in a design error or a design control issue.

EPU Extent of Condition Review

Master/Sub Component

Quad Cities

System FW Feed Water

Master Component Type	Master EPN/ECode	Equipment Name
Air Operated Valve Assembly (AOVA)	*1-3201-A 0000993619*	*ASSY - VALVE 1A RFP MIN FLOW AO PCV*

Cmp EPN		Equipment Name	Sub Cmp Type
1-3201-A	Sub Cmp	ACTUATOR 1A RFP MIN FLOW AO FCV	Actuator,Air (Valves) (A12)
1-3201-A	Mstr Cmp	ASSY - VALVE 1A RFP MIN FLOW AO PCV	Air Operated Valve Assembly (AOVA)
1-3201-A	Sub Cmp	REGULATOR IA TO PCV 1-3201A FILT PRV SET	Regulators (R17)
1-3201-A	Sub Cmp	SOLENOID 1A RFP MIN FLOW AO PCV	Solenoids (S45)
1-3201-A	Sub Cmp	VALVE 1A RFP MIN FLOW AO PCV	Valves - Air Operated (V05)
1-3201-B	Sub Cmp	ACTUATOR 1B RFP MIN FLOW AO FCV	Actuator,Air (Valves) (A12)
1-3201-B	Sub Cmp	ASSY - VALVE 1B RFP MIN FLOW AO PCV	Air Operated Valve Assembly (AOVA)
1-3201-B	Sub Cmp	REGULATOR IA TO PCV 1-3201B FILT PRV SET	Regulators (R17)
1-3201-B	Sub Cmp	SOLENOID 1B RFP MIN FLOW AO PCV	Solenoids (S45)
1-3201-B	Sub Cmp	VALVE 1B RFP MIN FLOW AO PCV	Valves - Air Operated (V05)
1-3201-C	Sub Cmp	ACTUATOR 1C RFP MIN FLOW AO FCV	Actuator,Air (Valves) (A12)
1-3201-C	Sub Cmp	ASSY - VALVE 1C RFP MIN FLOW AO PCV	Air Operated Valve Assembly (AOVA)
1-3201-C	Sub Cmp	REGULATOR IA TO PCV 1-3201C FILTER PRV SET	Regulators (R17)
1-3201-C	Sub Cmp	SOLENOID 1C RFP MIN FLOW AO PCV	Solenoids (S45)
1-3201-C	Sub Cmp	VALVE 1C RFP MIN FLOW AO PCV	Valves - Air Operated (V05)
2-3201-A	Sub Cmp	ACTUATOR 2A RFP MIN FLOW AO PCV	Accumulators (A10)
2-3201-A	Sub Cmp	ASSY - VALVE 2A RFP MIN FLOW AO PCV	Air Operated Valve Assembly (AOVA)
2-3201-A	Sub Cmp	SOLENOID 2A RFP MIN FLOW AO PCV	Solenoids (S45)
2-3201-A	Sub Cmp	VALVE 2A RFP MIN FLOW AO PCV	Valves - Air Operated (V05)
2-3201-A	Sub Cmp	VALVE SOLENOID OP TO 2A RFP MIN RECIRC	Valves - Solenoid (V27)
2-3201-B	Sub Cmp	ACTUATOR 2B RFP MIN FLOW AO PCV	Accumulators (A10)
2-3201-B	Sub Cmp	ASSY - VALVE 2B RFP MIN FLOW AO PCV	Air Operated Valve Assembly (AOVA)
2-3201-B	Sub Cmp	REGULATOR IA TO PCV 2-3201B FILT PRV SET	Regulators (R17)
2-3201-B	Sub Cmp	SOLENOID 2B RFP MIN FLOW AO PCV	Solenoids (S45)
2-3201-B	Sub Cmp	VALVE 2B RFP MIN FLOW AO PCV	Valves - Air Operated (V05)
2-3201-B	Sub Cmp	VALVE SOLENOID 2B RFP MIN FLOW	Valves - Solenoid (V27)
2-3201-C	Sub Cmp	ACTUATOR 2C RFP MIN FLOW AO PCV	Accumulators (A10)
2-3201-C	Sub Cmp	ASSY - VALVE 2C RFP MIN FLOW AO PCV	Air Operated Valve Assembly (AOVA)
2-3201-C	Sub Cmp	SOLENOID 2C RFP MIN FLOW AO PCV	Solenoids (S45)
2-3201-C	Sub Cmp	VALVE 2C RFP MIN FLOW AO PCV	Valves - Air Operated (V05)
2-3201-C	Sub Cmp	VALVE SOLENOID 2C RPF INST AIR FCV	Valves - Solenoid (V27)
ZS 2-3201-A	Sub Cmp	SWITCH 2A RFP RECIRC VLV POSITION	Position Switch (ZS)
ZS 2-3201-B	Sub Cmp	LIMIT SWITCH 2B RFP MIN FLOW AO PCV	Position Switch (ZS)
ZS 2-3201-C	Sub Cmp	LIMIT SWITCH 2C RFP MIN FLOW AO PCV	Position Switch (ZS)

Master Component Type	Master EPN/ECode	Equipment Name
Air Operated Valve Assembly (AOVA)	*1-3241-62 0000993622*	*ASSY - REACTOR FEED WATER*

Cmp EPN		Equipment Name	Sub Cmp Type
1-3241-62	Mstr Cmp	ASSY - REACTOR FEED WATER	Air Operated Valve Assembly (AOVA)
1-3241-62	Sub Cmp	REACTOR FEED WATER	Valves - Air Operated (V05)
2-3241-62	Sub Cmp	ASSY - REACTOR FD. WATER	Air Operated Valve Assembly (AOVA)
2-3241-62	Sub Cmp	REACTOR FD. WATER	Valves - Air Operated (V05)

Master Component Type	Master EPN/ECode	Equipment Name
Motor Operated Valve Assembly (MOVA)	*1-3201 0000993623*	*ASSY - VALVES RX FEED PUMP DISCHARGE*

Cmp EPN		Equipment Name	Sub Cmp Type
1-3201	Mstr Cmp	ASSY - VALVES RX FEED PUMP DISCHARGE	Motor Operated Valve Assembly (MOVA)
1-3201	Sub Cmp	VALVES RX FEED PUMP DISCHARGE	Valves - Motor Operated (V20)
1-3201-A	Sub Cmp	*1A RFP DSCH VLV (HW)	Motor Operated Valve Assembly (MOVA)
1-3201-A	Sub Cmp	LIMITORQUE 1A RFP DISCH VLV	Mov Valve Operator (L05)
1-3201-A	Sub Cmp	1A RFP DSCH VLV	Valves - Motor Operated (V20)
1-3201-B	Sub Cmp	+1B RFP DSCH VLV (HW)	Motor Operated Valve Assembly (MOVA)
1-3201-B	Sub Cmp	LIMITORQUE 1B RFP DISCH VLV	Mov Valve Operator (L05)
1-3201-B	Sub Cmp	1B RFP DSCH VLV	Valves - Motor Operated (V20)
1-3201-C	Sub Cmp	+1C RFP DSCH VLV (HW)	Motor Operated Valve Assembly (MOVA)
1-3201-C	Sub Cmp	LIMITORQUE 1C RFP DISCH VLV	Mov Valve Operator (L05)
1-3201-C	Sub Cmp	1C RFP DSCH VLV	Valves - Motor Operated (V20)
2-3201-A	Sub Cmp	+2A RFP DSCH VLV (HW)	Motor Operated Valve Assembly (MOVA)

EPU Extent of Condition Review

Master/Sub Component

Quad Cities

System FW Feed Water

Master Component Type	Master EPN/ECode	Equipment Name
Motor Operated Valve Assembly (MOVA)	*1-3201* *0000993623*	*ASSY - VALVES RX FEED PUMP DISCHARGE*

Cmp EPN	Type	Equipment Name	Sub Cmp Type
2-3201-A	Sub Cmp	LIMITORQUE 2A RFP DISCH	Mov Valve Operator (L05)
2-3201-A	Sub Cmp	2A RFP DSCH VLV	Valves - Motor Operated (V20)
2-3201-B	Sub Cmp	*2B RFP DSCH VLV (HW)	Motor Operated Valve Assembly (MOVA)
2-3201-B	Sub Cmp	LIMITORQUE 2B RFP DISCH VLV	Mov Valve Operator (L05)
2-3201-B	Sub Cmp	2B RFP DSCH VLV	Valves - Motor Operated (V20)
2-3201-C	Sub Cmp	+2C RFP DSCH VLV (HW)	Motor Operated Valve Assembly (MOVA)
2-3201-C	Sub Cmp	LIMITORQUE 2C RFP DISCH VLV	Mov Valve Operator (L05)
2-3201-C	Sub Cmp	2C RFP DSCH VLV	Valves - Motor Operated (V20)

Master Component Type	Master EPN/ECode	Equipment Name
Motor Operated Valve Assembly (MOVA)	*1-3202-A* *0000993627*	**1D1 HP FW HTR INLET VLV*

Cmp EPN	Type	Equipment Name	Sub Cmp Type
1-3202-A	Mstr Cmp	*1D1 HP FW HTR INLET VLV	Motor Operated Valve Assembly (MOVA)
1-3202-A	Sub Cmp	LIMITORQUE 1D1 HP FW HTR INLET VLV	Mov Valve Operator (L05)
1-3202-A	Sub Cmp	1D1 HP FW HTR INLET VLV	Valves - Motor Operated (V20)
1-3202-B	Sub Cmp	MOTOR 1B RFP AUX OIL PUMP	Motor (M10)
1-3202-B	Sub Cmp	*1D2 HP FW HTR INLET VLV	Motor Operated Valve Assembly (MOVA)
1-3202-B	Sub Cmp	LIMITORQUE 1D2 HP FW HTR INLET VLV	Mov Valve Operator (L05)
1-3202-B	Sub Cmp	1D2 HP FW HTR INLET VLV	Valves - Motor Operated (V20)
1-3202-C	Sub Cmp	*1D3 HP FW HTR INLET VLV	Motor Operated Valve Assembly (MOVA)
1-3202-C	Sub Cmp	LIMITORQUE 1D3 HP FW HTR INLET VLV	Mov Valve Operator (L05)
1-3202-C	Sub Cmp	1D3 HP FW HTR INLET VLV	Valves - Motor Operated (V20)
1-3204-A	Sub Cmp	*1D1 HP FW HTR OUTLET VLV	Motor Operated Valve Assembly (MOVA)
1-3204-A	Sub Cmp	LIMITORQUE 1D1 HP FW HTR OUTLET VLV	Mov Valve Operator (L05)
1-3204-A	Sub Cmp	1D1 HP FW HTR OUTLET VLV	Valves - Motor Operated (V20)
1-3204-B	Sub Cmp	*1D2 HP FW HTR OUTLET VLV	Motor Operated Valve Assembly (MOVA)
1-3204-B	Sub Cmp	LIMITORQUE 1D2 HP FW HTR OUTLET VLV	Mov Valve Operator (L05)
1-3204-B	Sub Cmp	1D2 HP FW HTR OUTLET VLV	Valves - Motor Operated (V20)
1-3204-C	Sub Cmp	*1D3 HP FW HTR OUTLET VLV (HW)	Motor Operated Valve Assembly (MOVA)
1-3204-C	Sub Cmp	LIMITORQUE 1D3 HP FW HTR OUTLET VLV	Mov Valve Operator (L05)
1-3204-C	Sub Cmp	1D3 HP FW HTR OUTLET VLV	Valves - Motor Operated (V20)
2-3202-A	Sub Cmp	*2D1 HI PRESS FW HTR INLET VLV (HW)	Motor Operated Valve Assembly (MOVA)
2-3202-A	Sub Cmp	LIMITORQUE 2D1 FW HTR INLET VLV	Mov Valve Operator (L05)
2-3202-A	Sub Cmp	2D1 FW HTR INLET VLV	Valves - Motor Operated (V20)
2-3202-B	Sub Cmp	*2D2 HI PRESS FW HTR INLET VLV (HW)	Motor Operated Valve Assembly (MOVA)
2-3202-B	Sub Cmp	LIMITORQUE 2D2 HI PRESS FW HTR INLET VLV	Mov Valve Operator (L05)
2-3202-B	Sub Cmp	2D2 HI PRESS FW HTR INLET VLV	Valves - Motor Operated (V20)
2-3202-C	Sub Cmp	*2D3 HI PRESS FW HTR INLET VLV (HW)	Motor Operated Valve Assembly (MOVA)
2-3202-C	Sub Cmp	LIMITORQUE 2D3 HI PRESS FW HTR INLET VLV	Mov Valve Operator (L05)
2-3202-C	Sub Cmp	2D3 HI PRESS FW HTR INLET VLV	Valves - Motor Operated (V20)
2-3204-A	Sub Cmp	*2D1 HI PRESS FW HTR OUTLET VLV (HW)	Motor Operated Valve Assembly (MOVA)
2-3204-A	Sub Cmp	LIMITORQUE 2D1 FW HTR OUTLET VLV	Mov Valve Operator (L05)
2-3204-A	Sub Cmp	2D1 FW HTR OUTLET VLV	Valves - Motor Operated (V20)
2-3204-B	Sub Cmp	*2D2 HI PRESS FW HTR OUTLET VLV (HW)	Motor Operated Valve Assembly (MOVA)
2-3204-B	Sub Cmp	LIMITORQUE HI PRESS FW HTR OUTLET VLV	Mov Valve Operator (L05)
2-3204-B	Sub Cmp	2D2 HI PRESS FW HTR OUTLET VLV	Valves - Motor Operated (V20)
2-3204-C	Sub Cmp	*2D3 HI PRESS FW HTR OUTLET VLV (HW)	Motor Operated Valve Assembly (MOVA)
2-3204-C	Sub Cmp	LIMITORQUE 2D3 HI PRESS FW HTR OUTLET VLV	Mov Valve Operator (L05)
2-3204-C	Sub Cmp	2D3 HI PRESS FW HTR OUTLET VLV	Valves - Motor Operated (V20)

Master Component Type	Master EPN/ECode	Equipment Name
Motor Operated Valve Assembly (MOVA)	*1-3203* *0000993630*	**HI PRESS FW HTR MO BYP VLV*

Cmp EPN	Type	Equipment Name	Sub Cmp Type
1-3203	Mstr Cmp	*HI PRESS FW HTR MO BYP VLV	Motor Operated Valve Assembly (MOVA)
1-3203	Sub Cmp	LIMITORQUE HP FW HTR MO BYP VLV	Mov Valve Operator (L05)
1-3203	Sub Cmp	HI PRESS FW HTR MO BYP VLV	Valves - Motor Operated (V20)
2-3203	Sub Cmp	*HI PRESS FW HTR MO BYP VLV (HW)	Motor Operated Valve Assembly (MOVA)
2-3203	Sub Cmp	LIMITORQUE HP FW HTR MO BYP VLV	Mov Valve Operator (L05)

EPU Extent of Condition Review

Master/Sub Component

Quad Cities

System FW Feed Water

Master Component Type	Master EPN/ECode	Equipment Name
Motor Operated Valve Assembly (MOVA)	*1-3203* *0000993630*	**HI PRESS FW HTR MO BYP VLV*

Cmp EPN		Equipment Name	Sub Cmp Type
2-3203	Sub Cmp	HI PRESS FW HTR MO BYP VLV	Valves - Motor Operated (V20)

Master Component Type	Master EPN/ECode	Equipment Name
Motor Operated Valve Assembly (MOVA)	*1-3205-A* *0000993634*	**VALVE FW HDR A TO RX ISOL*

Cmp EPN		Equipment Name	Sub Cmp Type
1-3205-A	Mstr Cmp	*VALVE FW HDR A TO RX ISOL	Motor Operated Valve Assembly (MOVA)
1-3205-A	Sub Cmp	LIMITORQUE FW HDR 'A' TO RX ISOL VLV	Mov Valve Operator (L05)
1-3205-A	Sub Cmp	VALVE FW HDR 'A' TO RX ISOL	Valves - Motor Operated (V20)
1-3205-B	Sub Cmp	+FW HDR B TO REACTOR ISOL VLV	Motor Operated Valve Assembly (MOVA)
1-3205-B	Sub Cmp	LIMITORQUE FW HDR 'B' TO RX ISOL VLV	Mov Valve Operator (L05)
1-3205-B	Sub Cmp	VALVE FW HDR 'B' TO RX ISOL	Valves - Motor Operated (V20)
1-3206-A	Sub Cmp	+A FW INLET ISOL VLV	Motor Operated Valve Assembly (MOVA)
1-3206-A	Sub Cmp	LIMITORQUE 'A' FW INLET ISOL VLV	Mov Valve Operator (L05)
1-3206-A	Sub Cmp	A FW INLET ISOL VLV	Valves - Motor Operated (V20)
1-3206-B	Sub Cmp	+B FW INLET ISOL VLV	Motor Operated Valve Assembly (MOVA)
1-3206-B	Sub Cmp	LIMITORQUE 'B' FW INLET ISOL VLV	Mov Valve Operator (L05)
1-3206-B	Sub Cmp	B FW INLET ISOL VLV	Valves - Motor Operated (V20)
2-3205-A	Sub Cmp	*FW HDR A TO REACTOR MO ISOL VLV (HW)	Motor Operated Valve Assembly (MOVA)
2-3205-A	Sub Cmp	LIMITORQUE FW HDR 'A' TO REACTOR MO ISOL VLV	Mov Valve Operator (L05)
2-3205-A	Sub Cmp	VALVE FW HDR 'A' TO REACTOR MO ISOL	Valves - Motor Operated (V20)
2-3205-B	Sub Cmp	*FW HDR B TO REACTOR MO ISOL VLV (HW)	Motor Operated Valve Assembly (MOVA)
2-3205-B	Sub Cmp	LIMITORQUE FW HDR 'B' TO REACTOR MO ISOL VLV	Mov Valve Operator (L05)
2-3205-B	Sub Cmp	VALVE FW HDR 'B' TO REACTOR MO ISOL	Valves - Motor Operated (V20)
2-3206-A	Sub Cmp	+A FW INLET ISOL VLV (HW)	Motor Operated Valve Assembly (MOVA)
2-3206-A	Sub Cmp	LIMITORQUE 'A' FW INLET ISOL VALVE	Mov Valve Operator (L05)
2-3206-A	Sub Cmp	A FW INLET ISOL VLV	Valves - Motor Operated (V20)
2-3206-B	Sub Cmp	+B FW INLET ISOL VLV (HW)	Motor Operated Valve Assembly (MOVA)
2-3206-B	Sub Cmp	LIMITORQUE 'B' FW INLET ISOL VALVE	Mov Valve Operator (L05)
2-3206-B	Sub Cmp	B FW INLET ISOL VLV	Valves - Motor Operated (V20)

Master Component Type	Master EPN/ECode	Equipment Name
Piping Specialty,Fittings (P20)	*1-3204-A* *0000899839*	*PIPING*

Cmp EPN		Equipment Name	Sub Cmp Type
1-3204-A	Mstr Cmp	PIPING	Piping Specialty,Fittings (P20)
1-3204-B	Sub Cmp	PIPING	Piping Specialty,Fittings (P20)
1-3205-A	Sub Cmp	PIPING REACTOR FEEDWATER	Piping Specialty,Fittings (P20)
1-3205-A3	Sub Cmp	PIPING REACTOR FEED	Piping Specialty,Fittings (P20)
1-3205-B6	Sub Cmp	PIPING REACTOR FEEDWATER	Piping Specialty,Fittings (P20)
1-3205-C6	Sub Cmp	PIPING REACTOR FEEDWATER	Piping Specialty,Fittings (P20)
2-3204-A	Sub Cmp	PIPING A LOOP FEED WATER	Piping Specialty,Fittings (P20)
2-3204-B	Sub Cmp	PIPING B LOOP FEED WATER	Piping Specialty,Fittings (P20)

Master Component Type	Master EPN/ECode	Equipment Name
Piping Specialty,Fittings (P20)	*2-3205-A3F* *0000942456*	*PIPING 'A' RFP MIN FLOW LINE*

Cmp EPN		Equipment Name	Sub Cmp Type
2-3205-A3F	Mstr Cmp	PIPING 'A' RFP MIN FLOW LINE	Piping Specialty,Fittings (P20)
2-3205-A6	Sub Cmp	LINE RFP MIN FLOW LINE	Piping Specialty,Fittings (P20)
2-3205-B3F	Sub Cmp	PIPING 'B' RFP MIN FLOW LINE	Piping Specialty,Fittings (P20)
2-3205-B6	Sub Cmp	PIPING 'B' RFP MIN FLOW LINE	Piping Specialty,Fittings (P20)
2-3205-C	Sub Cmp	PIPING 'C' RFP MIN FLOW LINE	Piping Specialty,Fittings (P20)
2-3205-C3F	Sub Cmp	PIPING 'C' RFP MIN FLOW LINE	Piping Specialty,Fittings (P20)

Master Component Type	Master EPN/ECode	Equipment Name
Probs, Temp, Levels, Sampling (P27)	*2-3299-167A* *0000942730*	*PROBE FEEDWATER INJECTION*

Cmp EPN		Equipment Name	Sub Cmp Type

EPU Extent of Condition Review

Master/Sub Component

Quad Cities

System FW Feed Water

Master Component Type	Master EPN/ECode	Equipment Name
Probs, Temp, Levels, Sampling (P27)	*2-3299-167A 0000942730*	*PROBE FEEDWATER INJECTION*

Cmp EPN		Equipment Name	Sub Cmp Type
2-3299-167A	Mstr Cmp	PROBE FEEDWATER INJECTION	Probs, Temp, Levels, Sampling (P27)
2-3299-167B	Sub Cmp	PROBE FEEDWATER INJECTION	Probs, Temp, Levels, Sampling (P27)
2-3299-167C	Sub Cmp	PROBE FEEDWATER INJECTION	Probs, Temp, Levels, Sampling (P27)
2-3299-168A	Sub Cmp	PROBE, FEEDWATER INJECTION	Probs, Temp, Levels, Sampling (P27)
2-3299-168B	Sub Cmp	PROBE, FEEDWATER INJECTION	Probs, Temp, Levels, Sampling (P27)
2-3299-168C	Sub Cmp	PROBE, FEEDWATER INJECTION	Probs, Temp, Levels, Sampling (P27)

Master Component Type	Master EPN/ECode	Equipment Name
Pump Assembly (PMPA)	*1-3201-A 0000993638*	*ASSY - PUMP 1A REACTOR FEEDWATER*

Cmp EPN		Equipment Name	Sub Cmp Type
1-3201-A	Sub Cmp	GEAR 1A RX FD PUMP SPEED INCREASER	Gear Reducer (G34)
1-3201-A	Sub Cmp	MOTOR 1A REACTOR FEED PUMP	Motor (M10)
1-3201-A	Sub Cmp	1A REACTOR FEED PUMP	Pump (P30)
1-3201-A	Mstr Cmp	ASSY - PUMP 1A REACTOR FEEDWATER	Pump Assembly (PMPA)
1-3201-A1	Sub Cmp	PUMP 1A RFP SHAFT DRIVEN LUBE OIL	Pump (P30)
1-3201-A1	Sub Cmp	ASSY - PUMP 1A RFP SHAFT DRIVEN LUBE OIL	Pump Assembly (PMPA)
1-3201-B	Sub Cmp	GEAR 1B RX FD PUMP SPEED INCREASER	Gear Reducer (G34)
1-3201-B	Sub Cmp	MOTOR 1B REACTOR FEED PUMP	Motor (M10)
1-3201-B	Sub Cmp	PUMP 1B REACTOR FEED	Pump (P30)
1-3201-B	Sub Cmp	ASSY - PUMP 1B REACTOR FEEDWATER	Pump Assembly (PMPA)
1-3201-B1	Sub Cmp	PUMP 1B RFP SHAFT DRIVEN LUBE OIL PUMP	Pump (P30)
1-3201-B1	Sub Cmp	ASSY - PUMP 1B RFP SHAFT DRIVEN LUBE OIL PUMP	Pump Assembly (PMPA)
1-3201-C	Sub Cmp	GEAR 1C RX FD PP SPEED INCREASER	Gear Reducer (G34)
1-3201-C	Sub Cmp	MOTOR 1C REACTOR FEED PUMP	Motor (M10)
1-3201-C	Sub Cmp	PUMP 1C REACTOR FEED	Pump (P30)
1-3201-C	Sub Cmp	ASSY - PUMP 1C REACTOR FEEDWATER	Pump Assembly (PMPA)
1-3201-C1	Sub Cmp	PUMP 1C RFP SHAFT DRIVEN LUBE OIL	Pump (P30)
1-3201-C1	Sub Cmp	ASSY - PUMP 1C RFP SHAFT DRIVEN LUBE OIL	Pump Assembly (PMPA)
2-3200-C	Sub Cmp	SEAL 2C REACTOR FEED PUMP INBOARD	Seals Pwr (S12)
2-3201-A	Sub Cmp	GEAR 2A RX FD PP SPEED INCREASER	Gear Reducer (G34)
2-3201-A	Sub Cmp	MOTOR 2A REACTOR FEED PUMP	Motor (M10)
2-3201-A	Sub Cmp	PUMP 2A REACTOR FEED	Pump (P30)
2-3201-A	Sub Cmp	ASSY - PUMP 2A REACTOR FEED	Pump Assembly (PMPA)
2-3201-A1	Sub Cmp	PUMP 2A RFP SHAFT DRIVEN LUBE OIL	Pump (P30)
2-3201-A1	Sub Cmp	ASSY - PUMP 2A RFP SHAFT DRIVEN LUBE OIL	Pump Assembly (PMPA)
2-3201-B	Sub Cmp	GEAR 2B RX FD PP SPEED INCREASER	Gear Reducer (G34)
2-3201-B	Sub Cmp	MOTOR 2B REACTOR FEED PUMP	Motor (M10)
2-3201-B	Sub Cmp	PUMP 2B REACTOR FEED	Pump (P30)
2-3201-B	Sub Cmp	ASSY - PUMP 2B REACTOR FEED	Pump Assembly (PMPA)
2-3201-B1	Sub Cmp	PUMP 2B RFP SHAFT DRIVEN LUBE OIL	Pump (P30)
2-3201-B1	Sub Cmp	ASSY - PUMP 2B RFP SHAFT DRIVEN LUBE OIL	Pump Assembly (PMPA)
2-3201-C	Sub Cmp	GEAR 2C RX FD PP SPEED INCREASER	Gear Reducer (G34)
2-3201-C	Sub Cmp	MOTOR 2C REACTOR FEED PUMP	Motor (M10)
2-3201-C	Sub Cmp	PUMP 2C REACTOR FEED	Pump (P30)
2-3201-C	Sub Cmp	ASSY - PUMP 2C REACTOR FEED	Pump Assembly (PMPA)
2-3201-C1	Sub Cmp	PUMP 2C RFP SHAFT DRIVEN LUBE OIL	Pump (P30)
2-3201-C1	Sub Cmp	ASSY - PUMP 2C RFP SHAFT DRIVEN LUBE OIL	Pump Assembly (PMPA)

Master Component Type	Master EPN/ECode	Equipment Name
Temp Special (TX)	*TX 1-3241-22A 0000900060*	*1D1 FW HTR FW INLET*

Cmp EPN		Equipment Name	Sub Cmp Type
TE 1-3241-8A	Sub Cmp	RX FEED PUMP INLET	Temp Element, Primary (TE)
TE 1-3241-8B	Sub Cmp	RX FEED PUMP INLET	Temp Element, Primary (TE)
TE 2-3241-21	Sub Cmp	2D FW HTRS FW OUTLET HEADER	Temp Element, Primary (TE)
TE 2-3241-4	Sub Cmp	2D FW HEATERS FW INLET HEADER	Temp Element, Primary (TE)
TE 2-3241-5	Sub Cmp	2D1 FW HTR FW OUTLET	Temp Element, Primary (TE)
TE 2-3241-6	Sub Cmp	2D2 FW HTR FW OUTLET	Temp Element, Primary (TE)
TE 2-3241-7	Sub Cmp	2D3 FW HTR FW OUTLET	Temp Element, Primary (TE)

EPU Extent of Condition Review

Master/Sub Component

Quad Cities

System FW Feed Water

Master Component Type	Master EPN/ECode	Equipment Name
Temp Special (TX)	*TX 1-3241-22A 0000900060*	*1D1 FW HTR FW INLET*

Cmp EPN		Equipment Name	Sub Cmp Type
TX 1-3241-22A	Mstr Cmp	1D1 FW HTR FW INLET	Temp Special (TX)
TX 1-3241-22B	Sub Cmp	1D2 FW HTR FW INLET	Temp Special (TX)
TX 1-3241-22C	Sub Cmp	1D3 FW HTR FW INLET	Temp Special (TX)
TX 1-3241-23A	Sub Cmp	1D1 FW HTR FW OUTLET	Temp Special (TX)
TX 1-3241-23B	Sub Cmp	1D2 FW HTR FW OUTLET	Temp Special (TX)
TX 1-3241-23C	Sub Cmp	1D3 FW HTR FW OUTLET	Temp Special (TX)
TX 2-3241-22A	Sub Cmp	2D1 FW HTR FW INLET	Temp Special (TX)
TX 2-3241-22B	Sub Cmp	2D2 FW HTR FW INLET	Temp Special (TX)
TX 2-3241-22C	Sub Cmp	2D3 FW HTR FW INLET	Temp Special (TX)
TX 2-3241-23A	Sub Cmp	2D1 FW HTR FW OUTLET	Temp Special (TX)
TX 2-3241-23B	Sub Cmp	2D2 FW HTR FW OUTLET	Temp Special (TX)
TX 2-3241-23C	Sub Cmp	2D3 FW HTR FW OUTLET	Temp Special (TX)

Master Component Type	Master EPN/ECode	Equipment Name
Valves - Hydraulic Operated (V10)	*EPU-QDC-FW-0001 300000372*	*FEEDWATER REG VALVE*

Cmp EPN		Equipment Name	Sub Cmp Type
EPU-QDC-FW-0001	Mstr Cmp	FEEDWATER REG VALVE	Valves - Hydraulic Operated (V10)

Master Component Type	Master EPN/ECode	Equipment Name
Valves - No Operators (V25)	*1-3208-A 0000899925*	*1A RFP DSCH CK VLV*

Cmp EPN		Equipment Name	Sub Cmp Type
1-3208-A	Mstr Cmp	1A RFP DSCH CK VLV	Valves - No Operators (V25)
1-3208-B	Sub Cmp	1B RFP DSCH CK VLV	Valves - No Operators (V25)
1-3208-C	Sub Cmp	1C RFP DSCH CK VLV	Valves - No Operators (V25)
2-3208-A	Sub Cmp	2A RFP DSCH CK VLV	Valves - No Operators (V25)
2-3208-B	Sub Cmp	2B RFP DSCH CK VLV	Valves - No Operators (V25)
2-3208-C	Sub Cmp	2C RFP DSCH CK VLV	Valves - No Operators (V25)

EPU Extent of Condition Review

Vulnerability of Components/Subcomponents

Quad Cities
System FW Feed Water

System Parameter

Parameter	Analytical Pre-EPU Level	Analytical Post-EPU Level	System Design Basis Value	Limiting Pre-EPU Plant Data	Limiting Post-EPU Plant Data
Flow - Total FW System (Mlbs/hr)	9.72	11.71	11.71	9.74	11.57

Related Vulnerabilities of Components

(See Master/Sub Component Report for listing of all components evaluated)

Component Type	Failure Vulnerability:	Critical Characteristic	Expect Value for Critical Characteristic	Actual Value for Critical Characteristic	Limit	Failure Mechanism	GAP	Risk Priority
Air Operated Valve Assembly	Orp Work Around	Vibration	N/A	N/A	N/A	Wear	Yes	L

Master EPN : 1-3201-A ASSY - VALVE 1A RFP MIN FLOW AO PCV

Rec: Perform valve, actuator and positioner inspection for wear on the AO 3201 during the next refuel outage. Recommend overall of the valve and operator taking particular care to document as found conditions and recommend vibration analysis of the valve, particularly of the Air Operator, attached air piping, and the solenoid valve.

Comments: BASIS FOR RECOMMENDATION
There may be additional vibrations imposed on the valve based on the operating experience gained from the system engineers. It was stated that the vibrations have increased due to three feedwater pumps in operations. Note that the consequences of a failure of the valve should be minimal because the minimum flow valves fail open on a loss of air. This would provide pump protection. The Feedwater System has the capability to makeup for an inadvertent opening of a single minimum flow control valve based on the additional flow available with three feedwater pumps in operation. The CMO identified that these valves are trouble-prone to leakage past their seats. The effects of vibration and wear at EPU conditions cannot be easily evaluated on the sub-components of this valve. The likely cause of failure would be wear since operation at EPU condition for one cycle has not resulted in any fatigue failures of these valves. Note, however, that the minimum flow valve (AO 3201) is normally closed at full power operation; therefore, the number of cycles resulting in fatigue may not have occurred at EPU operating conditions.· However, if sub-component are now excited at EPU operating frequencies or vibration amplitudes increased, then potential wear patterns on the valves sub-components could be detected after one operating cycle and if significant replaced prior to failure.· Visual inspection of the valve stem external surfaces should also be performed. If significant wear is detected, replaced worn components prior to failure and adjustment PM cycle to control inspection/replacement frequencies of identified susceptible sub-components.
RISK
During a scram or recirculation runback with closure of FWRV's, the feedwater system is near or at deadhead conditions and over pressurizes without the minimum flow valve opening. Pump damage or component degradation may occur if this valve does not open

Motor Operated Valve Assembly	SCRAM	Stroke Time	N/A	N/A	N/A	Binding	No

Master EPN : 1-3202-A *1D1 HP FW HTR INLET VLV

Rec: Review sizing calculation of isolation valves MO 3202 A, B, C and MO 3204 A, B, C at the EPU operating differential pressure conditions.

Comments: BASIS FOR RECOMMENDATION
The feedwater system MOVs consist of heater string isolation MO 3202A, B, C and MO 3204 A, B, C for the inlet and outlet isolation of the HP heaters strings. In addition, a bypass MOV, MO 3203, exists around the LP heater string. In the event of high heater level due to tube rupture, the inlet and outlet isolation valves close and the bypass valve opens to prevent turbine water induction. A higher flow results in a higher differential pressure for valve closure, which may impose additional loads on the motor, operated valves. It is understood that the margins for the motor operator motors are in the range of 200%. With an increased differential pressure across the valves, the primary potential is to affect the stroke time of the valve. The are no assumptions made of these valves in the transient analysis of the plants, however an assumption is made that the feedwater temperature change due to a feedwater heater isolation will take a minute or more. Therefore no affect on the affect on the loss of feedwater heating event and there is no foreseen loss of margin. However, in the event of a heater tube rupture, a failure of the valves to close could result in water backing up into the main turbine. In the event of high heater level due to tube rupture, the inlet and outlet isolation valves close and the bypass valve opens to prevent turbine water induction. ·Typically, motor operated valves are sized based upon design conditions (shutoff conditions that provide the

maximum differential pressure at which the valves must open), however; actual operating conditions could have been used in the actuator sizing of MO 3202A, B, C and MO 3204 A, B, C.· EPU has increased the operating differential pressure across the D Feedwater Heaters by as much as 27 psid. Adjustment of the torque switch setting may be required based upon this sizing review.
RISK
The Feedwater heater system is designed to close the inlet and outlet isolation valves and open the bypass valve based upon high heater level due to tube rupture to prevent turbine water induction. Water induction into the turbine could cause turbine failure in which a scram would likely occur prior to the water induction if the valves do not operate properly.

Motor Operated Valve Assembly	SCRAM	Stroke Time	greater than 1 minute	N/A	no faster than 1 minute	Binding	Yes	M

Master EPN : 1-3203 *HI PRESS FW HTR MO BYP VLV

Rec: Evaluate the capability of the motor operator to operate at the peak expected differential pressure. Review sizing calculation of bypass valve MO 3203 to ensure adequate margin exists at the EPU operating differential pressure conditions.

Comments: BASIS FOR RECOMMENDATION
A higher flow results in a higher differential pressure for valve closure, which may impose additional loads on the motor, operated valves. It is understood that the margins for the motor operator motors are in the range of 200%. With an increased differential pressure across the MO-3203, D' HEATERS BYP VLV, the primary potential is to affect the stroke time of the valve. The assumptions made of this valve, in the transient analysis of the plants, is that the performance of the valve will be no better than a stroke time of one minute. Therefore no affect on the affect on the loss of feedwater heating event and there is no foreseen loss of margin. However, in the event of a heater tube rupture, a failure of the valve to open, along with failure of the D heater isolation valves to close due the increased differential pressure, could result in water backing up into the main turbine. In the event of high heater level due to tube rupture, the inlet and outlet isolation valves close and the bypass valve opens to prevent turbine water induction. ·Typically, motor operated valves are sized based upon design conditions (shutoff conditions that provide the maximum differential pressure at which the valves must open), however; actual operating conditions could have been used in the actuator sizing of MO 3203.· EPU has increased the operating differential pressure across MO 3203 by up to 27 psid.. Adjustment of the torque switch setting may be required based upon this sizing review.
RISK
The Feedwater heater system is designed to close the inlet and outlet isolation valves and open the bypass valve based upon high heater level due to tube rupture to prevent turbine water induction. Water induction into the turbine could cause turbine failure in which a scram would likely occur prior to the water induction if the valves do not operate properly.

Motor Operated Valve Assembly	Unexpected Acc Degradation	Vibration	N/A	N/A	N/A	Wear	Yes	M

Master EPN : 1-3205-A *VALVE FW HDR A TO RX ISOL

Rec: Perform external valve/internal operator/motor inspection for wear on one of the 3206 MOVs during next refuel outage.

Comments: BASIS OF RECOMMENDATION
The effects of vibration and wear at increase EPU flows cannot be easily evaluated on the sub-components of these valves. The likely cause of failure would be due to wear or loosening of fasteners since operation at EPU condition of one cycle has not resulted in any fatigue failures of these valves .The most susceptible sub-components to wear would be the valves body to bonnet fasteners, valve stem or the actuator limit switches, torque switch or rotor. Body to bonnet fasteners have been found to be loose, in the past, due to vibrations. If sub-components are now excited at EPU operating frequencies or vibration amplitudes increased, then potential wear patterns on these sub-components could be detected after one operating cycle and if significant, replaced prior to failure. Visual inspection of the valve stem external surfaces should also be performed If significant wear is detected, replaced worn components prior to failure and adjustment PM cycle to control inspection/replacement frequencies of identified susceptible sub-components.
RISK
The Feedwater Regulation isolation valves are designed to provide isolation of the control valves for out of service conditions. In the event of an unisolable leak of valve, the unit may have to, at the extreme conditions, shutdown to repair a leak.

Motor Operated Valve Assembly	Unexpected Acc Degradation	Vibration	N/A	N/A	N/A	Wear	Yes	M

Master EPN : 1-3201 ASSY - VALVES RX FEED PUMP DISCHARGE

Rec: Perform external valve/internal operator/motor inspection for wear on one of the 3201 MOVs during next refuel outage.

Comments: BASIS OF RECOMMENDATION
The effects of vibration and wear at increase EPU flows cannot be easily evaluated on the sub-components of these valves. The likely cause of failure would be due to wear or loosening of fasteners since operation at EPU condition of one cycle has not resulted in any fatigue failures of these valves .The most susceptible sub-components to wear would be the valves body to bonnet fasteners, valve stem or the actuator limit switches, torque switch or rotor. Body to bonnet fasteners have been found to be loose, in the past, due to vibrations. If sub-components are now excited at EPU operating frequencies or vibration amplitudes increased, then potential wear patterns on these sub-components could be detected after one operating cycle and if significant, replaced prior to failure. Visual inspection of the valve stem external surfaces should also be performed If significant wear is detected, replaced worn components prior to failure and adjustment PM cycle to control inspection/replacement frequencies of identified susceptible sub-components.
RISK
The Feedwater discharge valves are designed to provide isolation of the feedwater pump for maintenance during plant operations. In the event of an unisolable leak on the discharge check valve the unit may have to, at the extreme conditions, shutdown to repair a leak. The risk associated with a failure of the valve to operate would at least require an operator work around to manually close the MOV in the event the Feedwater Pump needs to be isolated during operations.

Probs, Temp, Levels, Sampling	De-rate	Vibration	N/A	N/A	N/A	Wear	Yes	M

Master EPN : 2-3299-167A PROBE FEEDWATER INJECTION

Rec: Replace the feedwater sample probes (Note Unit 2 is complete - found to be in good condition).

Comments: BASIS OF RECOMMENDATION
In response to several concerns, one of which would be the damage to the Dresden feedwater sparger. The shorter design should be less subject to failure. Dresden had sample probe failures.Three FW probes and one condensate sample probe failed due to high cycle fatigue induced by flow-induced vibration. Quad Cities Operability Evaluation CR 190513 addresses the consequences of a probe failure as a result of Dresden's three FW probe failures. ·GE-NE-000-0024-2731 determined ratio of Vortex Frequency for various sample probes but did not include thermowells· GE-NE-000-0024-2731 determined frequency ratios of 1.026, 0.87, 0.86, 0.3, 0.9 and 0.71 for various sample probe cases. Per ASME Appendix N - Below 0.76 acceptable however other documents more restrictive. GE-NE-000-0024-2731 has determined that for certain sample probe cases, the vortex shedding frequency is sufficiently close to a sample probe first natural frequency during various operating conditions that lock-on could occur (i.e. resonant condition). GE-NE-000-0024-2731 did not include thermowells ·Continued successful operation of existing probes or thermowells for an entire operating cycle cannot be assured. If it is determined by analysis that the vortex shedding frequency is sufficiently close to a thermowell's first natural frequency during various operating conditions that lock-on could occur, then the thermowell shall be re-designed.
RISK
Probe and thermowell failure could occur and either enter a feedwater heater and become lodged or enter a feedwater pump and possibly cause pump damage. A failure of a thermowell will also likely cause a breach in the system. In both cased, a unit derate or shutdown may be required.

Pump Assembly	De-rate	Number of Pumps	3	3	3	Wear	Yes	H

Master EPN : 1-3201-A ASSY - PUMP 1A REACTOR FEEDWATER

Rec: Evaluate if there is some (probably small) potential that full-range testing of pump capability might support operation at 100% EPU power with one of the pumps in standby or might support identification of cost-effective modifications (e.g., change in impeller design) that would allow restoration of the pre-EPU stand-by pump configuration. Perform testing to determine the optimum operating conditions at which to start and stop feedwater pumps for providing optimal power production and running conditions of these pumps. This would be a plant specific analysis.

Comments: BASIS OF RECOMENDATION
Per Task Report T0701, when operating with the modified high pressure turbine with only two feed and three condensate / condensate booster pumps on line, the flow passing capability of the feedwater flow control valves will become limiting for operation near 100% of 2527/2511 MWth (pre-EPU 100% power level). While at 110 % of 2527/2511 MWth, the condensate / condensate booster pump motors will exceed their full load amp ratings. In order to achieve maximum operational flexibility a third feed pump should be placed on-line prior to exceeding 100% of 2527/2511 MWth (85.5% LPU), and the fourth condensate/condensate booster pump placed on-line prior to reaching 110% of 2527/2511 MWth (94 % LPU).·These above values for placing additional pumps in service at 100% and 110% Pre-EPU power levels are also true if a feedwater or condensate / condensate booster pump is lost or removed from service which is not significantly dependent upon time of season which

is based upon theoretically pump crves. Maintenance or loss of condensate, condensate booster or feedwater pumps/motors will result in unit derates during non-outage periods or to avoid derates will have to occur during outage periods. No testing has been performed to verify achievable power with less than full complement of pumps. Per discussion with system engineers, there has been no full-range testing to confirm/identify achievable power with less than the full complement of pumps. There are no spare gear boxes available that would be needed in the event of a failure.
RISK
At the minimum the plant would derate to 100% of pre-EPU conditions if a feedwater pump was placed out of service. Maintenance will result in unit derates during non-outage periods or to avoid derates will have to occur during outage periods. This require outage resources.

Pump Assembly	De-rate	Vibration	N/A	N/A	N/A	Wear	Yes	M

Master EPN : 1-3201-A ASSY - PUMP 1A REACTOR FEEDWATER

Rec: Measure the vibrations of FW pump components.

Comments: BASIS OF RECOMMENDATION
System engineers and Exelon Vibration experts recognize noticably increased vibration (possibly flow-change or equipment line-up induced) in small-bore pipe and tubing associated with the Feedwater System
RISK
At the minimum, the plant would derate to 100% of pre-EPU conditions if a feedwater pump was placed out of service

Temp Special	De-rate	Vibration	N/A	N/A	N/A	Wear	Yes	H

Master EPN : TX 1-3241-22A 1D1 FW HTR FW INLET

Rec: Evaluate Feedwater system temperature element thermowells to eliminate concern with vortex shedding failure vulnerability..

Comments: BASIS OF RECOMMENDATION
Dresden has had sample probe failures. Three FW probes and one condensate sample probe failed due to high cycle fatigue induced by flow-induced vibration. GE-NE-000-0024-2731 determined ratio of Vortex Frequency for various sample probes but did not include thermowells. GE-NE-000-0024-2731 determined frequency ratios of 1.026, 0.87, 0.86, 0.3, 0.9 and 0.71 for various sample probe cases. Per ASME Appendix N - Below 0.76 acceptable however other documents more restrictive. GE-NE-000-0024-2731 has determined that for certain sample probe cases, the vortex shedding frequency is sufficiently close to a sample probe first natural frequency during various operating conditions that lock-on could occur (i.e. resonant condition). GE-NE-000-0024-2731 did not include thermowells . Continued successful operation of existing probes or thermowells for an entire operating cycle cannot be assured. If it is determined by analysis that the vortex shedding frequency is sufficiently close to a thermowell's first natural frequency during various operating conditions that lock-on could occur, then the thermowell shall be re-designed
RISK
Probe and thermowell failure could occur and either enter a feedwater heater and enter a feedwater sparger and possibly cause damage. A failure of a thermowell will also likely cause a breach in the system. In both cased, a unit derate or shutdown may be required.

Valves - Hydraulic Operated	De-rate	Vibration	N/A	N/A	N/A	Wear	Yes	M

Master EPN : EPU-QDC-FW-0001 FEEDWATER REG VALVE

Rec: Perform actuator/valve inspection for wear on both of the Feedwater Reg valves during next refuel outage.

Comments: BASIS OF RECOMMENDATION
The effects of vibration and wear at increase EPU flows cannot be easily evaluated on the sub-components of these valves. The likely cause of failure would be due to wear since operation at EPU condition of one cycle has not resulted in any fatigue failures of these valves. The most susceptible sub-components to wear would be the valves stem or the actuator parts. Quad Cities FW system engineer has noted lan increase in vibrations. The current PM inspects the valve and actuator, one each outage. However, if sub-components are now excited at EPU operating frequencies or vibration amplitudes increased, then potential wear patterns on these sub-components could be detected after one operating cycle and if significant, replaced prior to failure. Visual inspection should also be performed and documeneted to determine significant wear is detected. The goal is to replace worn components prior to failure and adjust PM cycles to control inspection/replacement frequencies of identified susceptible sub-components.
RISK

At the minimum, failure of a regulating valve would require a de-rate to repair.

Valves - No Operators	Orp Work Around	Flow	N/A	N/A	N/a	Wear	No

Master EPN : 1-3208-A 1A RFP DSCH CK VLV

Rec: Inspect one discharge check valve at the next outage and perform check valve monitoring or increase the frequency of the PM for the discharge check valve to manage the potential for accelerated wear on the hinge bushings.

Comments: BASIS FOR RECOMMENDATION
Due to the lower flow rate per pump there is potential for accelerated wear on the feedwater pump discharge check valve flapper hinge bushings. In addition Quad has detected pressure fluctuations in the feedwater system; if the pressure oscillations result in flow oscillations that affect the position of the valve flapper. If this condition persists, then the PM for the discharge check valve should be increased to manage the potential for accelerated wear on the hinge bushings
RISK
Avoid the potential of reverse flow through the feedwater pumps, when a pump is in standby, which would require operator work around to manage feedwater flow to the reactor.

System Parameter

Parameter	Analytical Pre-EPU Level	Analytical Post-EPU Level	System Design Basis Value	Limiting Pre-EPU Plant Data	Limiting Post-EPU Plant Data
Pressure - FW Pmp Discharge (PSIG)	N/A	1338	1850	1336	1410

Related Vulnerabilities of Components (See Master/Sub Component Report for listing of all components evaluated)

Component Type	Failure Vulnerability:	Critical Characteristic	Expect Value for Critical Characteristic	Actual Value for Critical Characteristic	Limit	Failure Mechanism	GAP	Risk Priority
Pump Assembly	De-rate	Pressure	N/A	N/A	N/A	Wear	Yes	M

Master EPN : 1-3201-A ASSY - PUMP 1A REACTOR FEEDWATER

Rec: *Recommend that the seal PM be increased and a review of three-pump operation be performed in order to prevent a de-rate.*

Comments: BASIS OF RECOMMENDATION
The Feedwater Pumps seal performance has been reviewed in the past due to many failures. The maintenance plan specifies a replacement of the seals every eight years. CMO suggests that the failure of the seals has increased due to the increased discharge pressure due to three pump EPU operation.
RISK
At the minimum the plant would derate to 100% of pre-EPU conditions if a feedwater pump was placed out of service

EPU Extent of Condition Review

Vulnerability of Components/Subcomponents

Quad Cities
System FW Feed Water

System Parameter

Parameter	Analytical Pre-EPU Level	Analytical Post-EPU Level	System Design Basis Value	Limiting Pre-EPU Plant Data	Limiting Post-EPU Plant Data
Pressure - FW Pmp Suction (PSIG) Min	N/A	166.7	N/A	221	204

Related Vulnerabilities of Components (See Master/Sub Component Report for listing of all components evaluated)

Component Type	Failure Vulnerablilty:	Critical Characteristic	Expect Value for Critical Characteristic	Actual Value for Critical Characteristic	Limit	Failure Mechanism	GAP	Risk Priority

Master EPN :

Rec:

Comments:

System Parameter

Parameter	Analytical Pre-EPU Level	Analytical Post-EPU Level	System Design Basis Value	Limiting Pre-EPU Plant Data	Limiting Post-EPU Plant Data
Temperature - FW Heater D Discharge (Max/Min) (F)	340/N/A	352/152	356/152 Transient Analysis, 350 Pipe Design	339	351

Related Vulnerabilities of Components (See Master/Sub Component Report for listing of all components evaluated)

Component Type	Failure Vulnerablilty:	Critical Characteristic	Expect Value for Critical Characteristic	Actual Value for Critical Characteristic	Limit	Failure Mechanism	GAP	Risk Priority
Piping Specialty,Fittings	LER	Stress	N/A	N/A	N/A	Design Errors	Yes	L

Master EPN : 1-3204-A PIPING

Rec: Revise Specification R-4411 o reflect a new design temperature of 356 F between the D Feedwater heaters and the Reactor.

Comments: BASIS OF RECOMMENDATION
The original design temperatures of 350 F downstream of the "D" FW heaters have been exceeded at EPU conditions. The EPU analytic temperature is 352 F in which the actual operating temperatures are just below the analytic limit (356 F).
RISK
The non-conservative temperatures in the specification can result in a design error or a design control issue.

EPU Extent of Condition Review

Vulnerability of Components/Subcomponents

Quad Cities
System FW Feed Water

System Parameter

Parameter	Analytical Pre-EPU Level	Analytical Post-EPU Level	System Design Basis Value	Limiting Pre-EPU Plant Data	Limiting Post-EPU Plant Data
Temperature - FW Pmp Suction (F)	300	318	N/A	304	313

Related Vulnerabilities of Components (See Master/Sub Component Report for listing of all components evaluated)

Component Type	Failure Vulnerability:	Critical Characteristic	Expect Value for Critical Characteristic	Actual Value for Critical Characteristic	Limit	Failure Mechanism	GAP	Risk Priority
Piping Specialty,Fittings	LER	Stress	318	313	300	Design Errors	Yes	L

Master EPN : 2-3205-A3F PIPING 'A' RFP MIN FLOW LINE

Rec: Revise Specification R-4411 to reflect a new design temperature of 318 F between the Feedwater Pump minimum flow line and the condenser.

Comments: BASIS OF RECOMMENDATION
The original design temperatures of 300 F downstream of the Feedwater Pump minimum flow valves have been exceeded at EPU conditions.
RISK
The non-conservative temperatures in the specification can result in a design error or a design control issue.

9. Miscellaneous Documents

The following documents are attached:

- Previously Identified Issues for the FW system

Known Feedwater System Vulnerabilities

Issue	Source	Document	Impact/Recommendation/Solution	Previous EPU Failure	Remarks
The Heat Exchange Institute Standards for Closed Feedwater Heaters and the Tubular Heat Exchanger Manufactures Association Standards have established criteria limiting tube flow velocities in order to minimize tube end erosion. For stainless steel tubes the applicable limit is 10 ft/s(corrected to 60 degree F). At 115% uprate, all the heaters except heater A2 will exceed a tube velocity of 10ft/s.	Margin Reduction Report & Pinch Points & Task Report Recommendations & Quad Cities Concerns (Q-034) & Post EPU Assessment	T0701	While not a cause for immediate concern, operation at velocities above 10 ft/s can be expected to result in increased rates of tube end erosion and should continued be to monitored at the current frequencies for tube end erosion related wall thinning. This information was transmitted to the site system engineering personnel. Currently, the "D" heater flows are approximately 11 ft/sec prior to EPU. The lifetime average number of tubes plugged per heater on the "D" heaters at Dresden with these flows has been 16 out of 910 tubes or 1.7%. At EPU, the "D" heater flows will increase to approximately 12.5 ft/sec and the other heater flows to approximately 11 ft/sec or below. Dresden 3C2 FW heater degraded and should have tube bundle replaced	No	
When operating with the modified high pressure turbine with only two feed and three condensate / condensate booster pumps on line, the flow passing capability of the feedwater flow control valves will become limiting for operation near 100% of 2527/2511 MWth (pre-EPU 100% power level). While at 110 % of 2527/2511 MWth, the condensate / condensate booster pump motors will exceed their full load amp ratings. In order to achieve maximum operational flexibility a third feed pump should be placed on-line prior to exceeding 100% of 2527/2511 MWth (85.5% LPU), and the fourth condensate/condensate booster pump placed on-line prior to reaching 110% of 2527/2511 MWth (94 % LPU). These above values for placing additional pumps in service at 100% and 110% Pre-EPU power levels are also true if a feedwater or condensate / condensate booster pump is lost or removed from service which is not significantly dependent upon time of season.	Margin Reduction Report & Pinch Points & Quad Cities Concern (Q-054) & Post EPU Assessment	T0701	Maintenance or loss of condensate, condensate booster or feedwater pumps/motors will result in unit derates during non-outage periods or to avoid derates will have to occur during outage periods. No testing to verify achievable power with less than full complement of pumps (i.e. unknown).	No	
Feedwater and Condensate Booster Pump on line work will be recoded to refuel outages to avoid derates	Extent of Condition Report in November 2003 & Margin Report	CR D138767	SR's created on for the condensate and condensate booster pumps for outage pump work. Feedwater SR's be handled under AR 91238 task 61 for PM outage work.	No	Need to ensure that PM plan for future bundles PM appropriately to minimize derates that will be required to perform maintenance on line. See AT 75977-02 tracking resolution of PM scope review.

Issue	Source	Document	Impact/Recommendation/Solution	Previous EPU Failure	Remarks
Reactor Feed Pump Suction RV lines. 2B RFP suction relief valve developed leak and required derate to repair	Extent of Condition Report in November 2003	CR D116918 CR D124640	Small Bore piping developed leak due to vibration - high cycle fatigue. Performed vibration reading on all relief valves. All Unit 3 RVs and the Unit 2A and C were not susceptible to high cycle fatigue. The Unit 2 B was susceptible to high cycle fatigue. Vibration measurements were also performed on the piping. The Unit 3 B piping and the Unit 2 B and C piping was determined to be susceptible to vibration. WRs 75469, 66377 and 66382 were created to address these piping areas. The Unit 3 RV were installed with 2x1 welds during D3R17 to address the valve susceptibilities. Cat ID 1385353 was created to order RV with 2x1 welds in the future. The cracked weld on the 2B RV was repaired with a 2x1 weld.	Yes	2:1 welds on 2A and 2C were not performed. What about off normal conditions and Quad Cities Cat ID. Verify status of WRs
Reactor Feed Pump Suction Low pressure alarm	Extent of Condition Report in November 2003	CR Q172060 Ref CR Q150946, CR Q 155349, CR Q 164360	Received Reactor Feed Pump Low pressure suction alarms several times when demin dp at 29 psid (normal). Alarm is 200 psig. EC 344329 issued to change alarm from 200 psig to 180 psig	Yes	Verify if Unit 1 had this done.
Feedwater suction pressure trips setpoints should be staggered and timed delayed to avoid a complete loss of feedwater	Task Report Recommendation	T0701	Need to verify if staggered setpoints and timed delays implemented	No	Verify if done
Feedwater Piping Vibrations	Pinch Points & Quad Cities Concern (Q-058) & Post EPU Assessment & DQ Final Margins	T0701	The flow velocity in the Feedwater piping will increase under EPU and result in higher flow induced vibration. Failure to meet pre-determined acceptance limits may limit operation and may require modifications to reduce the vibration.	Yes	
Loss of condensate pump at full power will result in remaining pumps being over amp limit until runback. May also result in the trip of all FW pumps due to trip of one condensate pump.	Quad Cities Concern (Q-036)	T0701	Need to resolve potential damage to condensate pumps due to running at low flow with resultant high amps, and FW suction pressure trips at same setpoint. Need to implement modifications.	No	
Increased Feedwater flow will potentially accelerate erosion/corrosion rates	Post EPU Assessment & DQ Final Margins	T0701	Checkworks program changes are in place to ensure appropriate inspections are performed	No	Need to validate flows in Checkworks
Replace Dresden FW Heater Pressure Transmitters	Post EPU Assessment	T0701 & T0700	Actual Pressures at EPU conditions exceed T0700 VWO pressures. Pressure indication went off scale, no PI data available	Yes	Need to verify if installed
More rigorous Feedwater pump trip analysis is currently done to show there is adequate margin for trip avoidance	EPU Task Scope and Methods Comparisons	T0701	Dresden and Quad Cities had substantial margin and a more rigorous evaluation is probably not necessary; however, it could be done for completeness if desired	No	
Achievable power level with less than full complement of pumps is unknown	DQ Final Margins		No testing has yet been performed on Unit 3 to determine the achievable power level with selected pumps off line for maintenance. Unit 2 testing was performed in Feb. 2002 to determine appropriate power levels.	No	It is recommended that a special test be conducted when the next maintenance window presents to optimize the recommended stable power level for operation with various pump

Issue	Source	Document	Impact/Recommendation/Solution	Previous EPU Failure	Remarks
					combinations. It is also recommended to continue with the upgrades to match the impellers on all three feedwater pumps.
Required CV for flow control valves marginal at high power levels with only 2 FW pumps on-line.	DQ Final Margins & Margin Report	T0701	PM and current performance is acceptable. With a higher dP, valves will be less open. Currently valves are okay although neither will handle 100% flow. The performance of the FW regulating valves is as predicted. They are controlling at a less open condition than prior to EPU.	No	No actions are required beyond normal performance monitoring.
FW Pump cooling will be reduced due to running all three pumps. This results in potential reduced reliability and reduced motor life.	Margin Report	Item D-004/Q-004	Projection is that 30 years life still remain. GE evaluation documented in DRF-A61-00052- Tab 6 concluded that the pumps will be acceptable for an additional 30 years at EPU conditions, with all three pumps running. No change to the HVAC system was required.	No	Results should be factored into the PM plans.
Margin of available FW flow capacity to that required for handling transients. This makes potential pump degradation more significant and monitoring and trending for degradation more important.	Margin Report	T0701	Original design required 10% margin, revised analysis by GE determined that 5% was sufficient. EPU levels provide 7%. All appropriate available parameters are being monitored, adequate monitoring plans are in place. Current pump performance has been good.	No	Need to ensure that PM plan for future bundles PM appropriately to minimize derates that will be required to perform maintenance on line. See AT 75977-02 tracking resolution of PM scope review.

ATTACHMENT 2D

Reactor Internals and Recirculation Systems Report

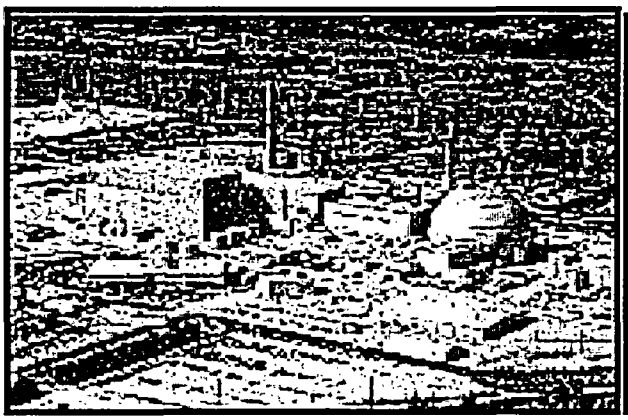
Exelon Corporation

and

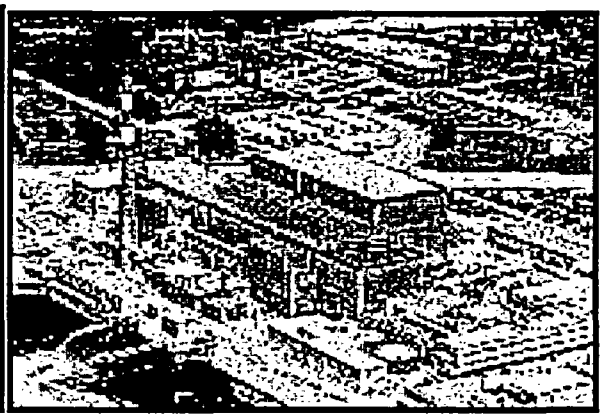
GE Energy

Extended Power Uprate

Extent of Condition
System & Component Evaluation



Dresden



Quad Cities

Reactor Internals and Recirculation Systems

SYSTEM NAME:	*Reactor Internals and Recirculation Systems*
STATION(S):	☒ Dresden ☒ Quad Cities
SYSTEM REVIEWERS:	Bob Geier

Table of Contents

1. Executive Summary

Process

The review of the Recirculation system and reactor internals was conducted using guidance in Extended Power Uprate Extent of Condition Review instruction Revision 0. These two topics are combined into one report since Jet pumps are internal to the reactor and are directly affected by changed recirc system parameters.

A total of 2304 components within the Dresden and Quad Cities recirculation system and reactor internals were considered in this review. A listing of these components included in this review is provided in Section 9 of this report.

EPU Task Reports, system descriptions, system specific trend data, system component list, previous EPU evaluations, and UFSARs were reviewed to identify EPU-related vulnerabilities (LER, ESF actuation, Reactor SCRAM, derate, OWA, Tech Spec entries, unexpected accelerated degradation) within the Reactor Recirculation system and Reactor Internals at Dresden and Quad Cities. In addition, the site Recirculation System Engineers (Bill Poppe-Dresden and Greg Houldson-Quad Cities and Jim Trettin-Quad Cities Systems Engineering Primary Lead) were interviewed as part of the Recirculation system evaluation. Keith Moser assisted with the reactor internals evaluation.

The System Engineers were questioned about the current operation and maintenance history of the Recirculation system and Reactor Internals, and limited trend data was considered. The Recirc System Engineers stated that there had been essentially no changes in Recirculation system operation or maintenance due to EPU. Though reactor power has increased, there have been no observable changes in flow controller positions, temperatures, pressures, or flow conditions attributed solely EPU. Of course, this is not the case for reactor internals.

Additionally, a second review was performed in San Jose with several GENE engineers of the Quad Cities Unit 2 Steam Dryer Failure Determination of Root Cause and Extent of Condition report during the week of April 27, 2004. The GENE team members participating in this review included Henry Hwang, Richard Wu, Hwang Choe, Maharaj Kaul, S. J. Lin and Sam Sundaram.

Based on these discussions, it was identified that components of the Recirculation system and Reactor Internals are subject to potential vulnerabilities as a result of operation under EPU conditions. A detailed component level evaluation was performed. The effect of system changes due to EPU on key system components is evaluated as part of the system level evaluation.

Parameter Evaluation

The changes in system parameters resulting from EPU are summarized in the table in Section 4 of this report.

The most important changes in system parameters affecting the reactor vessel and its internals and recirculation system are the 17% increase in reactor power and associated steam and feed flow which result in an increased fluence, and changes in the recirculation flow conditions as a result of changed core design and core flow resistance.

Vulnerabilities Identified

Potential vulnerabilities were identified during the system and component level reviews and were evaluated as documented in Section 6 of this report. Specific vulnerabilities are documented in Recommendations Section 2, and in the data base in Section 8 of this report.

The primary vulnerabilities center around the following issues:

- Increased resistance to flow in the core due to EPU core designs and resulting effect on Jet Pump performance, and limitations on Recirc pump speeds
- Changed flow conditions in the Moisture Separator and dryer and the effects of potential vibration,
- Reduced core flow window at 100% thermal power and effects on component life due to increased operation time at near full rated recirc pump speed,

Recommendations

During the next few fuel cycles, core designs will increasingly challenge the Jet Pumps and Recirc M-G sets and pumps as flow resistance increases at power. A theoretical threshold exists where bypass leakage at the Jet Pump slip joint bypass leakage (SJBL) will initiate and where vibration will drastically increase. This vibration will result in a sudden increase in the wear rate of the Jet Pump restrainer wedges. In turn, wedge wear will result in increased clearances at the restrainer set screws allowing the vibration amplitude to increase. This will cause fatigue damage to the Jet Pump riser braces and eventually after a predictable number of cycles, cracking will initiate. This mechanism has occurred at some BWR5s, including LaSalle, and one known BWR4. There has been no history of wedge wear or movement at Dresden, but there was some indication of movement during Q2R17 this winter. There is an investigation in progress at GE to determine the appropriate parameter that should be evaluated and for which a threshold value should be established to predict SJBL for the Dresden/Quad Cities Jet Pumps. If operation is anticipated close to this threshold, either mitigation actions need to be taken or contingency planning needs to occur for these actions.

The Recirc pumps and M-G set components are all operating close to their maximum capabilities. As core flow resistance continues to increase, this equipment will be pushed closer to those limitations with resulting impact to operational flexibility and to component life. Also, as the M-G sets are operated at higher speeds, steady state control will be challenged due to limitations of the fluid coupling's positioners. Currently, adjustable speed drives (ASD's) are being considered for installation at Quad Cities station. This modification should be supported for all four units.

There has been a history at pre-EPU conditions of vibration of small-bore Recirc loop flow instrumentation line high cycle fatigue failures. As the pumps are operated at speeds where they have not continuously operated in the past, there is a possibility that these lines could be vulnerable. Therefore, these lines should be examined at each opportunity during the next several refuel outages.

It has been concluded that increased steam line velocities under EPU conditions are responsible for damage that has resulted in catastrophic failures on the Quad Cities steam dryers and cracking at the same location on the Dresden dryers. The Moisture separators are not thought to be susceptible to the same fatigue mechanism. However, the flow conditions have changed to a point where it is prudent asset management to perform examinations at locations on the Moisture separator that have been vulnerable to fatigue cracking at other plants.

References:

1. Dresden 2-Diagram of Nuclear Boiler and Recirculation Piping, drawing M-26 sheets 1 and 2.
2. Dresden 3-Diagram of Nuclear Boiler and Recirculation Piping, drawing M-357 sheets 1, 2 and 3.
3. Dresden 2-Diagram of Recirculation Pump Motor and Generator (M-G) sets oil Piping, drawing M-174 sheet 1.
4. Dresden 3-Diagram of Recirculation Pump Motor and Generator (M-G) sets oil Piping, drawing M-174 sheet 2.
5. Quad Cities 1-Diagram of Nuclear Boiler and Recirculation Piping, drawing M-35 sheets 1 and 2.
6. Quad Cities 2-Diagram of Nuclear Boiler and Recirculation Piping, drawing M-77 sheets 1 and 2.
7. Quad Cities 1-Diagram of Recirculation Pump Motor and Generator (M-G) sets oil Piping, drawing M-35 sheet 4.
8. Quad Cities 2-Diagram of Recirculation Pump Motor and Generator (M-G) sets oil Piping, drawing M-77 sheet 4.
9. TO300 EPU Task Report, Nuclear Boiler System (Dresden 2 and 3), GENE-A22-00103-19-01, October 2000.
10. TO301 EPU Task Report, RPV Fracture Toughness, GENE-A22-00103-20-01, October 2000.
11. TO300 EPU Task Report, Nuclear Boiler System (Quad Cities 1 and 2), GENE-A22-00103-19-02, October 2000.
12. TO303 EPU Task Report, RPV Internals Structural Integrity Evaluation, GENE-A22-00103-05-01, October 2000.
13. T0304 EPU Task Report, Reactor Internal Pressure Differences, GENE-A22-00103-06-01, October 2000
14. TO305 EPU Task Report, Flow Induced Vibration, Reactor Internals, GENE-A22-00103-07-01, October 2000.
15. TO307 EPU Task Report, Reactor Recirculation System, GENE-A22-00103-23-01, October 2000.

16. TO308 EPU Task Report, Recirculation System Piping, GENE-A22-00103-23-01, October 2000.
17. BWR Jet Pump Assembly Maintenance Issues, S. Ranganath, M. Schrag, S. Kaylor, June 2002.
18. System Lesson Plan Dresden station, Module DRE202LN002, ILT: 12ILYSY, July 03.
19. System Lesson Plan Quad Cities Station, ModuleTRNOPSLP\LF-0202.R07, May 99.
20. Exelon document "Long Term Asset Management Strategy", Section 9, Reactor Coolant/Recirculation Pumps, April, 2003.
21. Exelon document "Long Term Asset Management Strategy", Section 10, Reactor Recirculation Motor Generator Sets, April, 2003.
22. Exelon document MA-AA-716-210-1001, High and Medium Voltage Electric Motors, PM Template, A. Mantey and W.Vargas, dated1/21/2002.
23. Dresden Abnormal Operating procedure DOA 0202-01, "Recirculation (Recirc) Pump Trip-One or Both Pumps", revision 24.
24. Dresden General procedure DGP 03-03, "Single Recirculation Loop Operation", revision 21.
25. Dresden "Invessel Visual Inspection Basis Document", revision 2
26. General Electric Report GENE-0000-0018-3358, Rev. 0, "Quad Cities Unit 2 Steam Dryer Failure- Determination of Root Cause and Extent of Condition", dated July 2003
27. General Electric Report GENE-0000-0018-3359-P, Rev. 1, "Quad Cities Unit 2 Steam Dryer Failure- Determination of Root Cause and Extent of Condition", dated August 2003
28. General Electric report NEDC-32984P-C, Joint Owners Group 09-02, part 5.3, Shroud Head Bolts
29. General Electric SIL No.35, Shroud Head Bolts, dated November 30, 1973
30. General Electric SIL No.433, Shroud Head Bolt Cracks, dated February 7, 1986

2. Recommendations and Bases (Project Instruction Attachment 5)

Reactor Internals		
Recommendations	**Basis**	**Risk**
Perform a one time visual examination of the shroud head bolt locking pin window for evidence of wear. Perform a one time visual inspection of the shroud head bolt mid-span and top support ring gussets at next refuel. Perform a one time inspection of a sample of the separator standpipe welds to the top of the shroud head for fatigue cracking.	Increased feed flow velocity could potentially cause high cycle and thermal fatigue of the shroud head bolts and their support ring gusset welds. The SHB has a 90° rotation limiting pin integral to the bolt shank in a "window" on the SHB sleeve. If the bolt is vibrating or loosens, the pin will rub and wear against the sleeve. Moisture separators have been vulnerable to fatigue cracking at other BWRs. Specifically, the mid-span and upper support ring gusset welds have experienced fatigue damage at other BWRs and these areas have not been examined at either plant since the 14th outages at Dresden. Increased feedwater flow velocity could potentially cause high cycle and thermal fatigue cracking on the standpipe welds and support ring gussets.	Failure of the shroud head bolts or the support ring gussets would result in a loss of function of shroud head bolts and de-tension the shroud head resulting in steam carry-under. This would not allow increased power with recirc pump flow and resulting in a derate (5). The probability of the occurrence of a gusset weld failure in light of vibration failures of the dryer is medium (5). The failure of the shroud head would be detectable immediately by the power-flow anomaly and is assigned a detect ability number of 5. Vulnerability Prioritization: 5x5x5=125 M
Perform a one time inspection of the feedwater sparger and end bracket pin hardware for evidence of vibration.	EPU conditions have increased feedwater flow. As a result, vibration modes may have changed. During Q2R17, one of the eight sparger end bracket attachment locking bolts was observed to have a missing nut. Others were found loose. There is a history of moving and rising pins at other BWRs. A review of the Dresden IVVI tapes verified that the nuts were in-place during the D2R18 inspections specified for NUREG 0619.	If an end bracket bolt is lost, a sparger could loose its seal in the safe-end bore if the end is not secured to the vessel bracket. When the damage was discovered, it would result in significant outage extension to repair and an LER (10). The probability of this occurrence is very small (1). The failure would not be detected and is assigned a detect ability number of 5. Vulnerability Prioritization: 10x1x5=50 M
Establish a threshold value of the appropriate parameter at which slip joint bypass leakage initiates Jet Pump	The pressure drop across the core is expected to increase as additional fuel is changed to new designs that support	Damaged Jet Pump riser braces would result in costly repairs and outage extensions

Reactor Internals		
Recommendations	**Basis**	**Risk**
vibration. Accelerate the BWRVIP-41 recommended inspection of restrainer gate wedges (WD-1) to verify that there is no evidence of vibration and wear. Perform this inspection activity every cycle utile confidence is developed that slip joint leakage will not cause degradation of the restrainer gate clearances.	twenty-four month EPU cycles. Eventually, increases in recirc pump speed will be necessary to achieve 100% core flow at end of cycle EPU conditions. It is anticipated that this change in pressure drop across the core will result in decreased M-ratios. A correlation has been postulated between M-ratio and the threshold at which increased slip joint bypass leakage initiates. Parameters other than M-ratio may be more suitable for indicating this operational threshold. Slip-joint bypass leakage causes Jet Pump vibrations that results in increased restrainer set-screw wear and accelerates wedge wear. As these clearances at the restrainer brace open, fatigue damage at leaf attachment welds at the vessel wall quickly accumulates and cracking initiates. There are two mitigation schemes that could be implemented to address this problem.	(5). Since core dP is increasing with every core change, the probability of slip joint leakage is increasing each cycle (10). Wedge wear would be revealed during BWRVIP-41 IVVI (5). Vulnerability Prioritization: 5x10x5=250 H

Reactor Recirculation		
Recommendations	**Basis**	**Risk**
Procure a spare Recirc pump motor and replace the motors in accordance with the High Voltage motor maintenance template once every ten years. Adjustable speed drives (ASD's) are being considered for installation at Quad Cities station. This modification should be supported for all four units.	Post EPU, the M-G sets and recirc pumps are operating close to their maximum capabilities. As core flow resistance continues to increase, the M-G sets will be operated more closely to their limits. The width of the operational window under EPU and MELLLA has been decreased for 100% power. It requires the minimum core flows of 95% vs 78% for Pre-EPU full power operation. Therefore, increased duty will be required of the recirc system. Most of these motors have never been refurbished. The remaining reliable life of these components is being challenged under EPU.	When the M-G set and pumps reach limitations, more frequent control rod pattern changes and associated derates (5) will be required. The failure of any motor of these motors will result in an outage. The probability of the occurrence of these limitations is medium (5). The operating limitation would be detectable immediately and is assigned a detect ability number of 5. Vulnerability Prioritization: 5x5x5=125 M
After Recirc pumps speeds are increased to levels not previously attained, inspect the Recirc Loop Flow sensing lines in the drywell and other small-bore piping attached to the Recirc system during each of the upcoming outages.	As pump speeds increase to values not previously seen, the resonant frequency of the lines may be reached. The small-bore sensing lines on the D3 reactor recirc loop flow instrumentation have failed due to high cycle fatigue at lower recirc pump vane passing frequencies on multiple occasions.	Failure of the sensing lines is an ASME Class 1 "structural integrity" issue and would cause an LER and is therefore a severity of 10. The probability of the occurrence of a failure is very small (1). The failure of the sensing line would result in immediately recognizable parameter changes and is assigned a detect ability number of 5. Vulnerability Prioritization: 10x1x5=50 M

3. Known Vulnerabilities Checklist Review

System Name: Reactor Internals and Recirculation *Instructions: Place a check mark for each unit for which the document was reviewed.*				
Reviewed Documents	**DRE U2**	**DRE U3**	**QDC U1**	**QDC U2**
Pinch Points	☒	☒	☒	☒
Margin List	☒	☒	☒	☒
Focus Review	☒	☒	☒	☒
CHIP	☒	☒	☒	☒
SHIP	☒	☒	☒	☒
Single Point Vulnerability	☒	☒	☒	☒
Scram De-Rate Challenge	☒	☒	☒	☒
Industry Feedback	☒	☒	☒	☒
GE Task Reports	☒	☒	☒	☒
Common Cause Analysis	☒	☒	☒	☒
EPU Evaluation Changes	☒	☒	☒	☒

4. Summary of System Parameter Changes

a. Evaluation of Parameter Changes with Plant Data

PI data for selected parameters was captured from point history for this review of the four units. Data is compared for the summer and winter seasons both prior to and after implementation of EPU. The parameters selected included Total Jet Pump flow, Recirc Pump Motor Power and Core Differential Pressure. These three data points were reviewed to attempt to validate the task reports to actual changes in recirculation flow. Recirc Pump Outlet Temperature was used to detect actual temperature changes.

Total Jet Pump Flow

The comparison of the four units for this indicator revealed the following information. At Dresden, full power operation prior to EPU during the summer months was often significantly curtailed due to lake temperatures. The addition of cooling towers during the first EPU cycle served to cloud this evaluation and as a result D2 trends seem meaningless. For reasons that were not established during this evaluation, D3 was impacted significantly less than D2. D3 recirc flow was also less than either Quad Cities unit. Quad Cities 1 total jet pump flow decreased significantly under EPU conditions. This may have been due to operation at pre-EPU after the dryer failure and being off-line during the summer for repairs

Quad Cities 2 did not enter coast-down conditions at the end of this cycle. Prior to EPU, Quad Cities 2 had licensed ICF up to 108% and had operated up to 102M#/hr at maximum pump speed (about 96.5% indicated). The other units are limited to 98M#/hr. As a result, QC 2 was able to operate with higher flow during the summer under EPU conditions.

Winter season data revealed that total jet pump flow decreased on all four units by a percent to over five percent on Quad Cities 2 from pre- to post-EPU. This is not consistent with predictions that core flow would increase under EPU.

MELLLA provides a higher rod line than previously used so that the units operate at lower core flows at a given power. Under EPU conditions, to achieve increased power, the unit operates further up the slope. At pre-EPU, 100% rated power was achieved (under ELLLA) at 87% core flow. Under EPU conditions along the MELLLA rod line, 100% thermal power is not reached until 95% core flow and the range of % core flow available at 100% thermal power is only 5% of core flow. Therefore, the minimum pump speed, where the plant would prefer to operate early in the cycle at EPU power is higher. The operating window under MELLLA and EPU conditions is actually smaller and requires higher core flow for rated power. In addition, the resistance to flow across the core is increased.

Core Differential Pressure

The comparison of the four units for this indicator revealed the following information. Again, at Dresden, full power operation prior to EPU during the summer months was often significantly curtailed due to lake temperatures. Core differential pressure (dP) increased on both Quad Cities

units under EPU conditions. On Quad Cities 2, there was a greater than 10% increase in dP and an increase in Jet pump flow. No other unit displayed that consistency in the predicted direction. In the winter months, Dresden dP increased and Quad Cities dP decreased although all Jet Pump flows decreased. No clear and constant trends are identified.

Recirc Pump Motor Power

The comparison of the four units for this indicator revealed the following information. Again, at Dresden, full power operation prior to EPU during the summer months was often significantly curtailed due to lake temperatures. Dresden 2 motor power was up post EPU reflecting the increased power that the unit was capable of producing with cooling towers. There was essentially no change in pump power for either D3 or QC1 during the summer months. Motor power increased by over 10% on QC 2. This is consistent with the discussions under total core flow.

Motor power decreased across all four plants in the winter months from pre- to post-EPU as did total jet pump flow.

In summary, for the three parameters evaluated above, it can be concluded that the task report estimation of an increase of required recirc pump speed was not consistently demonstrated. Under EPU conditions, motor power and Jet Pump flow generally decreased and core dP generally increased.

Recirculation Temperature

The comparison of the four units for this indicator revealed the following information. During the summer months, Dresden average and maximum temperatures consistently decreased, and except for QC1A, temperatures consistently increased at Quad Cities. During the winter months, temperatures consistently decreased by two degrees or less except for QC1B. The Task report for temperature decrease under EPU condition can be demonstrated.

Recirc System Temperature Reduction

As a result of increased feedwater flow and constant recirculation flow, there is an increase in the dilution of recirculation flow by relatively cool feedwater resulting in a decrease in recirculation pump suction temperature of 2.5°F to 528.6°F. Feedwater temperatures have increased by approximately 8°F. This improves subcooling to the suction of the recirculation pumps will reduce cavitation and benefit recirculation pump performance.

Winter Conditions

Unit-Process Computer Point	P&ID Point	Description	Unit	Pre-EPU Average	Pre-EPU Maximum	Pre-EPU Minimum	Post-EPU Average	Post-EPU Maximum	Post-EPU Minimum
D2-C202	A2709	Total Jet Pump	M#/HR	92.70177047	95.63320923	73.98322296	91.82031892	93.46582031	82.14011383
D3-C302	A3709	Total Jet Pump	M#/HR	94.14979621	95.31526184	93.13673401	93.73176502	95.35189056	92.71601868
Q1-V_C128	A1709	Total Jet Pump	M#/HR	95.91870639	96.99048615	94.30877686	91.80377367	92.76532745	90.47369385
Q2-V_C228	A2709	Total Jet Pump	M#/HR	97.33920337	100.0422745	61.10993958	90.97178227	91.79974365	90.43103027
D2-'C208	A2743	Recirc Pump A Outlet Temp	DEG. F	510.7951524	511.2767639	504.0869446	#DIV/0!	0	0
D2-'F289	A2744	Recirc Pump B Outlet Temp	DEG. F	527.5476002	528.1021729	521.1825562	526.922066	527.5625	522.0859375
D3-'C308	A3743	Recirc Pump A Outlet Temp	DEG. F	523.5233403	523.9909058	522.9876099	523.4056238	523.890625	522.796875
D3-'F389	A3744	Recirc Pump B Outlet Temp	DEG. F	524.6775375	524.9578857	524.4862061	523.7405114	524.265625	523.2890625
Q1-V_W126	A1745	Recirc Pump A Outlet Temp	DEG F	533.1995979	533.5731201	532.7059326	531.3624679	531.5579224	531.1958008
Q1-V_W128	A1747	Recirc Pump B Outlet Temp	DEG F	528.1283357	528.4981689	527.5789795	530.2185973	530.3883057	530.0322876
Q2-V_W226	A2745	Recirc Pump A Outlet Temp	DEG F	528.634808	529.3355713	518.2432251	525.8818233	526.1090698	525.7312622
Q2-V_W228	A2747	Recirc Pump B Outlet Temp	DEG F	528.9726515	529.7496338	518.7221069	526.4040195	526.6120605	526.2213745
D2-C201	A2694	Core Differential Pressure	PSID	16.14446121	17.02295494	10.40097237	16.67216299	17.33614922	13.1888485
D3-C301	A3694	Core Differential Pressure	PSID	16.57196807	16.8451004	16.33537865	17.09764762	17.61946678	16.84005356
Q1-V_C125	A1694	Core Differential Pressure	PSID	17.68545913	18.02289963	17.12502098	17.37348568	17.53021049	17.21990013
Q2-V_C225	A2694	Core Differential Pressure	PSID	16.46124121	17.31905937	6.078884602	15.92504746	16.2236824	15.80533123
D2-E231	A2725	Recirc Pump Motor Power	MW	3.837671245	4.219514847	1.901763201	3.630303546	3.857153893	2.497956514
D2-E232	A2726	Recirc Pump Motor Power	MW	4.008142213	4.419769287	1.945305705	3.878779096	4.094808102	2.658258677
D3-E331	A3725	Recirc Pump Motor Power	MW	3.592389682	3.682448387	3.504469395	3.694365274	3.858342648	3.615107298
D3-E332	A3726	Recirc Pump Motor Power	MW	3.609386272	3.70570302	3.519202709	3.643044731	3.811739206	3.567734957
Q1-V_E131	A1725	Recirc Pump Motor Power	MW	3.778168783	3.898148298	3.537899256	3.514826263	3.62611413	3.394503117
Q1-V_E132	A1726	Recirc Pump Motor Power	MW	3.794894206	3.924225569	3.583292246	3.535316565	3.558689117	3.50801158
Q2-V_E231	A2725	Recirc Pump Motor Power	MW	3.702888828	4.013677597	0.643042266	3.250649231	3.40348506	3.180674314
Q2-V_E232	A2726	Recirc Pump Motor Power	MW	3.683691665	4.03543663	0.62641263	3.251896873	3.325912952	3.169280767

Summer Conditions

Unit-Process Computer Point	P&ID Point	Description	Unit	Pre-EPU Average	Pre-EPU Maximum	Pre-EPU Minimum	Post-EPU Average	Post-EPU Maximum	Post-EPU Minimum
D2-C202	A2709	Total Jet Pump	M#/HR	64.2398455	93.44664001	47.39886093	92.91130205	94.39006805	47.39886093
D3-C302	A3709	Total Jet Pump	M#/HR	94.70946882	96.06912994	74.74697113	92.42101062	93.53484344	74.74697113
Q1-V_C128	A1709	Total Jet Pump	M#/HR	97.92489155	98.69844818	96.69194031	94.94632476	96.83203125	78.1762085
Q2-V_C228	A2709	Total Jet Pump	M#/HR	95.03381595	98.84989929	66.61985779	96.49954496	97.76865387	80.79436493
D2-'C208	A2743	Recirc Pump A Outlet Temp	DEG. F	497.0337086	510.7617188	490.4222717	492.4065857	492.4065857	492.4065857
D2-'F289	A2744	Recirc Pump B Outlet Temp	DEG. F	518.7127621	527.75	512.4508057	526.0698278	526.6640625	514.8312988
D3-'C308	A3743	Recirc Pump A Outlet Temp	DEG. F	527.35262	527.8242188	520.5500488	521.5261133	521.8746338	521.1934204
D3-'F389	A3744	Recirc Pump B Outlet Temp	DEG. F	525.4148686	526.1185303	518.5115356	523.8936926	524.3640137	523.5935669
Q1-V_W126	A1745	Recirc Pump A Outlet Temp	DEG F	535.4267227	535.6152344	535.2261963	531.046505	531.5901489	525.300293
Q1-V_W128	A1747	Recirc Pump B Outlet Temp	DEG F	528.6467475	528.8276978	528.3475342	530.8417528	531.630188	525.4386597
Q2-V_W226	A2745	Recirc Pump A Outlet Temp	DEG F	525.1797256	526.2043457	517.1166382	527.5299332	528.1057739	522.5474854
Q2-V_W228	A2747	Recirc Pump B Outlet Temp	DEG F	526.2481283	527.3581543	519.005127	527.7492021	528.2483521	523.0712891
D2-C201	A2694	Core Differential Pressure	PSID	7.857740485	16.11454391	3.826586485	16.66192251	17.02040291	3.920143604
D3-C301	A3694	Core Differential Pressure	PSID	16.42334552	16.87566948	10.5481348	15.90956549	16.12119865	15.69331932
Q1-V_C125	A1694	Core Differential Pressure	PSID	18.1195219	18.42063904	17.7437191	18.36188348	18.90261078	12.54195118
Q2-V_C225	A2694	Core Differential Pressure	PSID	15.81823737	16.93633461	7.005197048	17.66018311	18.14486694	12.49350834
D2-E231	A2725	Recirc Pump Motor Power	MW	1.310832971	3.870424271	0.380921006	3.759041877	3.87655139	0.390501618
D2-E232	A2726	Recirc Pump Motor Power	MW	1.387254871	4.010406971	0.403454959	4.01921649	4.150903225	0.416361868
D3-E331	A3725	Recirc Pump Motor Power	MW	3.549743334	3.728581905	1.73702395	3.545166732	3.6222229	3.47679615
D3-E332	A3726	Recirc Pump Motor Power	MW	3.530373295	3.723742962	1.729502082	3.638027853	3.735108376	3.569796324
Q1-V_E131	A1725	Recirc Pump Motor Power	MW	3.96959742	4.080327511	3.834770918	3.80329151	4.012353897	2.172073603
Q1-V_E132	A1726	Recirc Pump Motor Power	MW	3.992554085	4.123438358	3.850306749	3.795122231	4.04059267	2.113226652
Q2-V_E231	A2725	Recirc Pump Motor Power	MW	3.346537492	3.711092949	0.829988062	3.852412951	4.02753973	2.260179043
Q2-V_E232	A2726	Recirc Pump Motor Power	MW	3.423657895	3.795499086	0.847927392	3.798995962	3.986114502	2.127665997

b. Evaluation of Plant Parameter Changes using Calculations

Some parameters were not directly verifiable with plant instrumentation. The effects of these changes required involved calculations to determine the effects of EPU. These changes included increased fluence and the effects of increased flow on components. These are discussed below.

Fluence Increase and effect on the Reactor Vessel

An increase in power subjects the reactor vessel to an increase in neutron fluence as described in EPU Task report 0301. Independent of EPU, recent revisions to the ASME code had allowed Exelon to recalculate the effects of temperature change rates on the integrity of the reactor vessel shell material.

The Dresden and Quad Cities Pressure-Temperature (P-T) curves were revised and submitted, and received acceptance for use (TAC # 8353 and 8354). As a result of the License Renewal project, these curves were again revised to evaluate the effects of an additional twenty years of operation up to 54 EFPY (TAC # 7851) as described in General Electric report GE–NE-0000-0002-9629-01 (D2, typical of four units) and consider the effects of increased fluence under EPU conditions. In order to obtain acceptance of the improved P-T curves, the methodology employed by GE was refined in accordance with Guide 1.99. The updated evaluation concluded that the impact of increased fluence under EPU conditions is actually less than that previously evaluated in the pre-EPU analysis. The pre-EPU analysis had been unnecessarily conservative. These new curves allow pressure testing to be performed at lower temperatures and should allow the plants to start-up in a decreased time-frame. The performance of these evaluations to determine the effect of EPU on the material properties of the reactor vessel has relieved operational constraints.

Originally, the effects of fluence on the materials of the reactor vessel shell were evaluated in accordance with the Appendix G program described in the D&QC UFSARs. The NRC has recently approved the use of an alternate material surveillance program as described in BWRVIP-78 and 86, known as the Integrated Surveillance Program. This program is used to evaluate the effects of fluence on the fleet of reactor vessel shells. The effect of this change is that Exelon will be able to rely on material surveillance test programs from several reactors to predict the changes of material properties of the Exelon BWRs. This will decrease the frequency and cost of testing the coupons contained in the Exelon reactors. The effects of increased fluence on the Integrated Surveillance program has been evaluated and factored into the schedule for coupon removal described in BWRVIP-86.

The Dresden and Quad Cities (D&QC) reactor internals are routinely inspected as recommended in the BWRVIP Inspection and Evaluation documents, and as required by ASME Section XI. There are no changes as a result of EPU in inspection types or frequency described in either of these sets of documents. The document recommending top guide inspection, BWRVIP-26, did not recommend inspection of the surfaces of the "egg-crate" as a function of fluence. However, as a result of increased fluence under EPU conditions, the License Renewal project has committed D&QC to inspect the top guide on an increased frequency in response to cracking originally observed at Oyster Creek. CRD guide tube welds are inspected as recommended in BWRVIP-38. Since access to the guide tubes is through the top guide, top guide surfaces will be examined on the same frequency as the guide tubes (5% in 6 years).

Recirculation Flow (Pump speed) Increase

The GE task report T0307 estimated that 100% core flow for pre-EPU core designs corresponds to 94% of actual recirc pump speed (1627rpm) and estimated under EPU conditions, recirc pump speeds will be required to exceed the previous 94% of rated speed limit up to 95.9% (1654 rpm). Prior to D2R17 in 2001, this 94% of rated pump speed limitation was in place at all four units due to the susceptibility of Jet Pump flow sensing lines to resonance. This resonance phenomenon was identified in 1987 and described in SILs 420 and 551 and is described in task Report T0305. The susceptible lines were supported during each of the four unit's following refueling outages and the lines are no longer susceptible to recirc pump speed vane pass frequency.

Additionally, to avoid coast-down at the end of each cycle, increased flow was predicted to be necessary. This prediction was not realized during D2C18 where the unit operated at EPU conditions for about 700 days without a coast-down. Conservatisms were identified in the D2C18 (the first cycle of EPU) core design that allowed the unit to avoid coast-down. However, D2 did operate with indicated speeds of 97% (94.5% actual) at just below 97% core flow during the summer of 2003. To date, the other units have not operated for a full twenty-four month cycle and therefore the coast-down prediction has not been validated. The nuclear engineers at Dresden are now predicting that the first unit that will experience coast-down conditions under EPU will be Dresden 3 during the end of fuel cycle nineteen in August of 2006.

Dresden 2 had limitations in-place since original construction when start-up testing identified concerns with the original riser brace design that were left unchanged. An additional and redundant brace was installed on Dresden 3 and a different design was installed on the Quad Cities units. The Quad Cities brace is also somewhat unique in that the later BWR4's used a different design. Modifications have been performed on the Dresden 2 Jet Pump riser braces (mitigation clamps) that have technically removed this limit (Task Report T0307) and will permit operation up to 100% of rated pump speed.

Quad Cities 2 has had significant non-EPU speed control issues that have resulted in speed oscillation and unexpected power changes. As a result, recirc pump speed control is being updated to a digital system on Quad Cities unit 2 during Q2R17 in 2004.During the Exelon System Engineer interviews, GE reviewers identified that there appears to be a larger than normal speed mismatch conditions between recirc pump speeds exist on all of the units. In the past, Tech Specs required that speed be controlled to limit mismatch. Recently, the Tech Specs were revised to require flow match. Quad Cities matches recirc pump flow by operating in individual manual and has not previously had speed-matching circuitry similar to Dresden where-as Dresden operates in Master Manual control. On Dresden 2, flows are different between the two loops at matched speeds since the recirc pump impellers are different diameters. Station records and those of the pump manufacturer, Byron-Jackson, do not agree on the exact impeller diameters/configurations on Dresden 2. This is not a significant difference or a problem.

A review of recirc pump motor and M-G set motor and generator ratings was performed. The pump motor ratings have been changed at Dresden and are being changed at Quad Cities to uprate from the previous limit of 724 amps to 750 amps. Under increased motor current duty,

the critical variable for these motors becomes winding temperature. Winding temperature is influenced by drywell temperature. Drywell temperature was predicted to increase with EPU conditions, but material condition improvements with drywell cooler coils and blowers has actually decreased drywell temperatures. The recommended GE limit for motor temperature is (248°F) is referenced in the Dresden Operating procedure DAN 902(3)-4 F-9. This procedure limits temperatures to 230°F on D2 and 238°F on D3. The warmest motor is currently operating at 210°F on D2B (the largest impeller size). The projected temperature at 103% of rated speed, a limited ICF condition, is 220°F or an 18°F margin.

The M-G set motor and generator capability and ratings were also examined. GE has performed an evaluation of the recirculation system rotating equipment (DRF.B31-00287, dated November 19, 1999). The conclusions of this report were that M-G set fluid drive slip would limit speed to below 100.4% speed and that thermal aging of components (insulation breakdown) rapidly accelerates with increased speed above this point. The recirculation system Task report T0307 also states that operation over 100.7% rated speed will need additional evaluation. Due to issues with these aging and predicted breakdown of motor insulation, and due to obsolescence of speed control components, there is a corporate initiative currently being pursued to replace the M-G sets with Adjustable Speed Drives (ASD). The first unit targeted for ASD installation is Q2R18 in 2006. Eventually, the objective of this initiative to upgrade M-G sets will be to replace M-G sets with ASDs on all four units. In summary, as a result of increased recirc pump motor ratings, pump speeds can be increased to 1654 rpm corresponding to 100% core flow under EPU conditions with acceptable margin for the near term.

Increased Core Flow (ICF) to over 103% rated core flow is also under review. It is anticipated that ICF conditions will require the recirc pump to operate at full rated speed. At ICF conditions, T0307 states that the M-G set motor current and input power exceeds their rated values by 26 amps (3%) and 277kW (5%). Currently, ICF is not permitted and D&QC do not allow core flow above 100% rated core flow and it is not likely that the reliability of these components would be unchallenged at these conditions.

Dresden 2's M-G set ventilation dampers were modified prior to EPU to reduce cooling air recirculation to address the anticipated higher heat load on the M-G motor. The generator output current is limited to the recirc pump motor amperage rating. Ventilation issues will be addressed during the group 2 EPU EOC system reviews.

With regard to components in the reactor vessel that would be subject to different loading as a result of increased recirculation flow, the access hole covers are situated on the shroud support plate abd would experience an increase in pressure difference due to flow. The Task report T0303 discusses the bolted cover but is silent on the one remaining original welded design that has not yet been repaired on Dresden unit 3. Therefore, it needs to be determined if the welded cover is bounded by the repaired bolted covers for this parameter change.

Main Steam and Feedwater Flow Increase

Although core flow is not increased, the Feedwater and Main Steam systems have increased flow to 117% of rated. Evaluations of the effects of increased flow in these systems are performed by Teams 1 and 2 under group 1 EPU EOC reviews.

Dryer

The effects of increased flow in these systems internal to the reactor are evaluated in this report. The moisture separator is experiencing changed flow conditions and the steam dryer is experiencing an increase in flow. The steam dryer was modified prior to EPU with the "perforated plate" mod to improve moisture removal performance (Task Report 303 paragraph 3.1.9). The steam dryer was evaluated for the effects of Flow Induced Vibrations in Task report T0305. Since this evaluation, the steam dryers at Quad Cities have each experienced repeated and severe cracking as a result of increased steam line velocities under EPU conditions. The driving mechanism and its source are currently under investigation. Acoustic circuit analysis methods are being used to evaluate pressure pulse measurement data to determine the relative magnitudes of the loading between the units. As illustrated by the differences in inspection outcomes between the units, the apparent level of loading on the Quad Cities units is larger than that at the Dresden units. During Q2R17 in March of 2004, the outboard hood vertical plate was repaired for a second time. Limited cracking was identified at each of the Dresden dryers during inspections during the winter 2003 and pre-emptive repairs were installed to prevent failures that have caused forced outages on each of the Quad Cities units. Enhanced inspection activities as recommended in GE SIL-644 will continue to monitor the progression of this cracking during the upcoming Dresden and Quad Cities refueling outages with dryer replacements being considered.

Dryer Drain Channels

GE SIL 474 described dryer drain channel cracking problems identified at other BWRs. The Quad Cities and Dresden dryer drain channels are constructed of ¼" thick plate with 1/8" leg fillet welds and are heavier than the standard design channel and are thought to be less susceptible these problems. Cracking has been identified for the first time at Dresden and Quad Cities during the extensive dryer inspections recommended by SIL 644. These inspections were performed in response to the Quad Cities dryer hood failures and will continue to be performed until the dryers are replaced.

Dryer Support Lugs

Dryer rocking has been identified to be a cause of cracking in some BWR4 dryers. Dresden and Quad Cities have leveling screws that assure a uniform contact between all four support points. IVVI inspections have revealed that contact does exist and this is not a vulnerability on these units.

Feedwater Spargers

The feedwater spargers are experiencing increased flow and have been evaluated for these conditions (Task Report 303 paragraph 3.1.10). During the recent Q2R17 Quad Cities outage, one sparger end bracket stop (nut) was identified to be missing. In addition, there was evidence of rotation and lifting on other end bracket hardware. This was discussed with GENE personnel during the April session at GENE. Similar evidence of end bracket hardware loosening that is apparently caused by vibration has been identified at other BWRs (Brown's Ferry) with the triple seal interference fit spargers.

The original testing of this design sparger at the Moss Landing facility in the middle 1970s identified that the predominant source of sparger vibration was due to the effects of feedwater leakage past the fit in the safe-end. The Moss Landing tests used segments of various design spargers with a high capacity feed pump with actual full-flow velocity conditions. It was identified that the relative contribution of vibration caused by the turbulence associated with increased flow is not significant. This would not have been expected since turbulent vibration is proportional to the square of flow rate. Nozzles were plugged as necessary to model actual pressure differences at the seal.

The current design uses a 304 series stainless steel thermal sleeve with an interference fit in the bore of a carbon steel safe-end. Eventually, degradation of the seal between these two materials in the form of corrosion of the safe-end is expected and vibration will be initiated. The increased flow and pressure under EPU conditions would increase any established leakage and result in increased bypass leakage vibration. As a result, there is a concern that these components may be degrading and this represents a potential vulnerability at all four units.

After the installation of these spargers, thermal-couples were installed and temperature data was collected and analyzed by Nutech Engineers to assess the sealing performance of the sparger. Leakage is present when a temperature differential occurs in the area of the seal from the top to the bottom of the safe-end. These devices were removed within a few cycles of their installation since the fragile wiring was subject to damage by the removal of reflective insulation and ISI activities. If the sparger end brackets continue to demonstrate signs of excessive vibration, the station should consider installing additional thermo-couples to evaluate the sparger seal and evaluate the necessity of repairing the sparger.

The increase in feedwater flow could also effect components of the moisture separator including the shroud head bolt, their mid and top support rings and gussets, and the moisture separator standpipes. These conditions were also discussed at length during the April evaluation session.

Steam Separator/Shroud Head

The effects of flow on the side of the steam separator/shroud head were not discussed in the task reports. A separate analysis was prepared for the effects of vibration due to vortex shedding around the shroud head bolts is discussed in GE report GENE-0018-3359-P, rev 1 after the Quad Cities 2 failure. This report addresses the effects of EPU on all of the reactor internal

components that are believed to be effected in section 3.1. It considered components in the steam and feedwater flow paths. It concludes that there is no impact to the shroud head from the increase in feedwater flow. This and other components were discussed at length during the April session at GENE. The components of the steam separator and shroud head that were evaluated in this report by GE include the shroud head bolts and the separator standpipe assembly.

Vortex Shedding and Resonance of Shroud Head Bolts

The effects of increased feedwater velocity as it passes across the shroud head bolt (SHB) were evaluated in terms of the added loading due to vortex shedding. The resonant frequency of the SHB was calculated and compared to the calculated vortex shedding frequency resulting from the increased feedwater flow velocity. The desired outcome is that the vortex shedding frequency be one-third or less of the resonant frequency. If the vortex shedding frequency is at or above the resonant frequency, the primary or higher vibration mode frequency could match the vortex shedding frequency and initiate resonance and eventual fatigue usage if the resulting stresses are adequate. For the case of the SHBs, it was determined that these frequencies were close enough that it should be assumed that the SHB assembly is locked-in at resonance. The resulting stresses in the bolt were calculated as described in the ASME Code, Section III Appendice N-1321. Several conservatisms were identified with this analysis. For example, the assumed fixed end-condition at the bottom of the SHBs and the assumed clearance at the mid and top support rings conservatively influence the resonant frequency calculation. Also, the load is assumed to act over the length of the bolt and it does not.

It was the opinion of the GENE team that the gusset is sufficiently removed from the flow stream and would not be impacted by vortex shedding and that the clearance between the SHB and the ring would not transfer vibration loading to the gussets.

The effects of the increased feedwater temperature and decreased flow-rate of separated liquid phase after moisture separation resulting in reduced dilution of cold feedwater was discussed with thwe GENE team. Stated another way, the larger fraction of feedwater flow is mixed at a smaller temperature difference after EPU. If it is assumed that mixing is not complete in the vicinity of the sparger, the result of this change could be adverse thermal cycle fatigue at components directly in the flow stream including the SHBs and the standpipes. GENE stated that the net effect of this change with regard to high cycle thermal fatigue of the SHB and moisture separator standpipe is improved after EPU since the temperature difference is reduced and that mixing can be safely assumed to be complete. The basis for the recommendation that SHBs and support ring gusset welds be examined is industry experience at LaSalle and other plants where SHB failure and wear and gusset weld cracking due to vibration have occurred.

Separator Standpipe Assembly

Performance and design features of three separator standpipe assemblies were compared in GE report GENE-0018-3359-P, rev 1. Separator standpipe assemblies are fixed axial flow steam separators that swirl the mixture and centrifugally separate liquid from vapor phase steam. The older style Quad Cities/Dresden model 65M assemblies are thicker and structurally stiffer than the latest BWR6 model AS2B assemblies. A third model 67M from Browns Ferry was also

evaluated. Finite element models were prepared for each and the resonant frequencies of the 65M units were almost twice as high as the AS2B. During initial product development testing of the 65M style assembly, only thermal hydraulic and no vibration testing was performed. The BWR6 AS2B assembly has been extensively tested. AS2B testing revealed that the maximum vibration stresses were below the GE acceptance criteria (10ksi) that is below the code allowed (endurance) limit of 13ksi.

The effects of changed feedwater flow rates and temperatures were examined for the standpipe assembly. GENE-0018-3359-P evaluates the cases of complete and no mixing on thermal gradients across the assembly and concludes that the result of the high gradient case on the standpipe assembly is not significant. The effect increased feedwater velocity and turbulence on the standpipe and its connection to the top of the shroud head is not evaluated. In the opinion of the GENE team, the structure of the moisture separator/shroud head with reinforcing matrix at the mid span and top, as well as the thick connection at the shroud head and the SHB support ring is stiff enough that detrimental effects are not expected. It is a recommendation of this evaluation that an inspection of the connection point to the top of the shroud be performed of the peripheral assemblies to confirm that no fatigue damage has occurred.

Core Spray Piping

Core spray piping after it enters its RPV nozzle runs around the inside diameter of the vessel and then drops down to the shroud penetration. In this piping run, the Core Spray line is in the flowpath of slightly increased flow under EPU conditions. The effects of this increased flow are evaluated in GENE-0018-3359-P. In fact, total core flow has not increased and so this evaluation provides additional conservatism. This evaluation was discussed with the GENE team during the April session. The effects of increased feedwater velocity as it passes across the Core Spray piping were evaluated in terms of the added loading due to vortex shedding. The resonant frequency of the Core Spray piping was calculated and compared to the calculated vortex shedding frequency resulting from the increased feedwater flow velocity. The desired outcome is that the vortex shedding frequency be one-third or less of the resonant frequency. The calculation concluded that vortex shedding frequency is about 3 hz and the resonant frequency of the Core Spray piping spool piece is about 30 hz. Therefore the condition is met.

Core Spray Spargers

The core spray spargers are sheltered from steam flow as it exits the top of the core since the spargers are positioned outside of the diameter of the top guide. Secondary flow eddies are conservatively expected present around the sparger. After EPU, the radial power peaks are reduced and the flat power distribution results in increased flow in the peripheral areas from maximums of 7 to 9 fps. The magnitude of secondary flow velocity is between 10% to 20% of these values. The resulting vortex shedding frequency of these low velocities is very low and there would be no vibration due to flow.

Nozzles

The RPV nozzles evaluated in the GENE-0018-3359-P report included the three vessel head nozzles, the Main Steam, and water level instruments. The effects of increased steam and feedwater flow were evaluated as applicable and the resulting vortex shedding frequency was compared the resonant frequencies of the nozzles. As it would be expected, the resonant frequencies of the nozzles is high and is higher than the acoustic emissions that have been recorded during the acoustic circuit analysis effort. Therefore, there is not expected to be a problem resulting from high steam line flow under EPU with these components. This evaluation was discussed with the GENE team for each of these nozzles during the April session.

In the case of the three vessel head nozzles, the expected main steam flow across the inlet surface of the nozzles is zero. The nozzles are along a line that runs parallel to the center divider of the dryer. CFD models illustrating this concept were prepared by GENE. The lowest nozzle frequency on the head are the instrument nozzle (spare) and the head spray nozzle which have an identical configuration. The calculated resonant frequency is 238hz. Thus, the calculated stress due to vortex shedding is negligible. The head spray thermal sleeve and spray nozzle assembly was not evaluated since it is believed to not be present on the Quad Cities RPV heads (the subject unit of the report) although it is shown on the illustrations in the report. This piece is present on the Dresden units. Dr. Sundaram has agreed to evaluate this difference. As mentioned above, the expected cross flow velocity is negligible and there is not expected to be a problem with this omission.

The steam line nozzles are directly in the increased steam velocity flow path. The ANSYS computer program was used to calculate resonant frequency of the steam line nozzles. The calculated value is very high (775hz) as would be expected considering its mass. This model did not consider the attached piping due to the complexity of the piping configuration. Instead, the system was instrumented to get actual inputs to nozzle. Most other nozzles do not have flow so the effects of flow are secondary elsewhere. Acoustic loading frequencies are below 230 hz and therefore the nozzle is not subject to flow vibration loading under EPU conditions. The loadings on the attached piping are relevant and are addressed under the Main Steam system evaluation.

The water level instrumentation nozzles are two-inch alloy 600 pipe material seal welded on the vessel ID to the cladding with alloy 82/182. The pipe is interference fit at temperature and 0.020" at cool conditions. For potential dynamic response, the nozzle will act as if it is integral with the RPV wall as so it will be very stiff. The level instrument nozzles are located in stagnant flow regions just above the water line adjacent to the skirt (reference leg) and the variable leg is located near the top of the shroud head below the feedwater sparger. The low flow velocities and high resonant frequencies of the nozzles assure that there will be no effects due to FIV.

Dryer and Moisture Separator Guide Rods

A unique feature of the Dresden and Quad Cities units is two sets of guide rods.

The two dryer guide rods are located at 0° and 180° and as such, are similar to the head nozzles in that steam flow is symmetrically away from the centerline of the dryer. Therefore, the flow velocity and consequently the vortex shedding frequency are very low. The guide rods are 3"

diameter solid 304 stainless steel bar. The calculated resonant frequency of the guide rod is 15.2 hz.

The two moisture separator/shroud head guide rods are located at 20° and 200°. The upper end is attached to a lug welded to the vessel ID and the lower end is welded to a bracket on the shroud cylinder. The guide rods are 3" diameter solid 304 stainless steel bar and have an unsupported length of 89". This guide rod extends from stagnant steam regions above the feedwater spargers and extends into higher flow areas near the shroud head where velocities are calculated to be about 8 fps. Core flow rates are unchanged for EPU but increased carry-under results in increased water velocities by 2.5%. By conservatively assuming that this flow velocity occurs as cross-flow, the resulting vortex shedding frequency is calculated to be 14 hz. The calculated resonant frequency of this guide rod is 60.2 hz and therefore flow induced vibration of this guide rod is expected to be very small.

c. Evaluation of Parameter Changes based on Start-up Test Results

Several reactor internals components were instrumented during original plant start-up and the base-line data collected at that time can be extrapolated to evaluate the loads due to vibration under EPU conditions. General Electric report GENE-0018-3359-P evaluated the following components in that way. These methods were discussed with GENE personnel during the April reactor internal EPU EOC review sessions at GENE. The plant safety related components that were instrumented during start-up included the CRD guide tubes and In-core instrument guide tubes, the core shroud, the shroud head and separator, fuel channels, Jet Pumps, and Jet Pump sensing lines and riser braces. In order to monitor the vibration amplitudes and frequencies during start-up testing, strain gauges, accelerometers and displacement transducers were installed on these selected components prior to initial operation. The data was analyzed and compared to allowable limits derived from natural frequency modes of the subject component. Locations of high stress were identified and vibration limits for all of the sensor locations were established from the corresponding analytical mode shape. The vibration limits were determined by setting the maximum zero to peak stress amplitude of the mode to 10,000 psi, including the effect of stress concentration. This is conservatively below the ASME code limit for service in excess of 10^{11} cycles of 13,600 psi.

Vibration amplitudes measured during start-up testing at 75% and 100% power were used to extrapolate vibration amplitudes up to 2957 MW_t and 108% core flow. The extrapolated value was then compared to the original peak amplitude acceptance criteria. Sensors were oriented in different directions at the same locations between the first Dresden unit (D2) and the subsequent D3 and QC 1&2 unit start-ups. This allowed measurements of different components of vibration modes.

<u>Core Shroud</u>

The maximum vibration amplitudes for the core shroud were less than 25% of the acceptance criteria at any plant. There were four displacement gauges to measure shroud tangential motion on Dresden 2. On D3, one displacement gauge was installed to measure radial motion and two

velocity sensors were installed to measure tangential motion. Because of the low levels of vibration, there is no concern for vibration under EPU conditions.

Shroud Head and Separator Assembly

There were four velocity sensors to measure tangential motion on D2.

There are two sources of vibration associated with the Separator assembly: flow turbulence and periodic forces. Because there are no distinct peaks in turbulence excitation, the assembly will vibrate at the structural natural frequencies of the components. These frequencies do not change with higher steam flows. Therefore, EPU conditions will increase vibration magnitudes at these frequencies.

Periodic forces are generated by the swirling motion of the flow through the separator. The magnitude of the periodic excitation forces increases with the square of flow velocity while the forcing frequency increases linearly with flow. Examination of the actual response spectrum shows that there is no assembly response at the calculated periodic forcing function frequency of 21.8 hz. Thus it can be concluded that the vibration in the shroud head/separator assembly is mainly from flow turbulence.

Separator velocity increases by 15.6% under EPU conditions and therefore the resulting vibration will increase by 33.6%. Since the maximum vibration amplitudes for the core shroud were less than 25% of the acceptance criteria at any sensor during start-up testing, an increase of 33.6% will remain well below the acceptance criteria. Because of the low levels of vibration, there is no concern for vibration under EPU conditions.

Control Rod Drive Guide Tubes

During D2 start-up testing, three strain gauges were installed on guide tubes to measure axial strain. The following plants were not instrumented. The measured strains were less than 10% of the acceptance criteria. Since stresses in this area are mainly a function of core flow, stresses are expected to remain very low.

In-Core Instrument Guide Tube

During D2 start-up testing, three strain gauges were installed on two in-core instrument guide tubes. Again, since the measured strains were less than 10% of the acceptance criteria, the following plants were not instrumented. Since stresses in this area are mainly a function of core flow, stresses are expected to remain very low.

Jet Pump Assembly, Sensing Lines and Riser Brace

Eight sensors including two displacement and six strain gauges were installed on the Dresden units during start-up testing. The displacement sensors measured the vibration of the Jet Pump inlet elbow and the strain gauges measured the response of the riser brace and riser motion. The response of the D2 riser brace at vane pass frequency was very high. The D2 riser braces have

been repaired since this analysis was performed and are no longer an area of concern with regard to vane passing frequency (VPF). In addition, the Jet Pump sensing lines were also determined to be vulnerable to VPF at speeds near full rated speed (172 hz). The susceptible line on all four units have since been stiffened by the addition of support clamps which effectively reduced the free lengthy of tubing and raised the calculated resonant frequency of the piping to VPF exceeding 103% of rated speed.

The variation of the designs of the riser braces between Dresden 2, Dresden 3 and the two Quad Cites units is also evaluated in the GENE-0018-3359-P report. The report concludes that there are no other issues impacted by increased pump speeds or operation under EPU conditions and therefore no EPU vulnerabilities.

5. System Boundaries

The reactor recirculation boundary in this evaluation includes the recirculation piping and valves in the drywell, the recirc pump and motor, the M-G set and its oil circulation system and speed control system and Jet Pumps and associated Jet Pump flow sensing lines. Flow instrumentation electronics and its input to nuclear instrumentation are not included in this review. The sample system and associated PCIS is also not included. Runback and minimum speed features are included under the feedwater level control and feedwater systems.

The reactor internals boundary in this evaluation includes components inside of the reactor vessel. CRDs, neutron monitoring, and reactor protection (RPS) and pressure sensing and control (EHC) and level sensing instrumentation interfaces with ECCS and FWLCS are not included under this review.

The boundaries for the Reactor Recirculation system is as shown on the applicable P&IDs (e.g., M-26 and M-357 for Dresden and M-35 and M-77 for Quad Cities).

6. Vulnerability Review (Project Instruction Attachment 4)

6a. System Vulnerability Review

System-level worksheets are attached for the recirculation system and reactor internal.

6b. Component Vulnerability Review

EXTENT OF CONDITION - EVALUATION CRITERIA
VULNERABILITY REVIEW

System: Reactor Internals and Recirc
Parameter(s): Flow
Component(s): Jet Pumps, Recirc Pumps and M-G Sets, Moisture Separator/Steam Dryer
Subcomponent(s): N/A

Instructions: The purpose of this form is to assess potential vulnerabilities related to a changed system parameter or parameters. For each attribute, the associated questions should be evaluated relative to the effect of the changed parameter on a system component or components. For the system level evaluation, if "Yes" is indicated in the "Action?" column, a component level evaluation is required for that changed parameter. For the component level evaluation, if "Yes" is indicat

Attribute	Evaluation	Action?
Component ratings: Does the changed parameter exceed a component rating or design value, e.g., with regard to capacity, duty, pressure, flow, temperature, fluid state, fluid chemistry or voltage?	The Dresden and Quad Cities (D&QC) reactor internals are routinely inspected as recommended in the BWRVIP Inspection and Evaluation documents, and as required by ASME Section XI. There are no changes as a result of EPU in inspection types or frequency described in either of these sets of documents. As a result of increased fluence, the license renewal project has committed D&QC to inspect the top guide on an increased frequency. CRD guide tube welds are inspected as recommended in BWRVIP-38. Since access to the guide tubes is through the top guide, top guide surfaces will be examined on the same frequency as the guide tubes (5% in 6 years). The document recommending top guide inspection, BWRVIP-26, did not recommend inspection of the surfaces of the "egg-crate" as a function of fluence. Originally, the effects of fluence on the reactor vessel shell was evaluated in accordance with the Appendix H program described in the D&QC UFSARs. The NRC has recently approved Integrated Surveillance Program as described in BWRVIP-78 and 86. This program now is used evaluate the effects of fluence on the fleet of reactor vessel shells. The vessel P-T curves have also been recently revised for both D&QC in the License Renewal project (D2, typical of four units, GE–NE-0000-0002-9629-01) and consider the effects of increased fluence under EPU conditions. Recirc flow was predicted to increase under EPU by about 2%. The actual recirc system flow decreased about 0.5% under EPU and MELLLA to less than 100% rated core flow. Under EPU conditions, the pressure drop across the core is expected to increase as additional fuel is changed to new designs. Also, to avoid coast-down at the end of each cycle, increased flow was predicted to be necessary. This prediction was not realized during D2C18 where the unit operated at EPU conditions for about 700 days. Rated (100%) core flow for the D&QC current core designs corresponds to about 94% of rated recirc pump speed (1627rpm). It is conservatively expected that at end of cycle EPU conditions, recirc pump speeds will be required to exceed the previous limit 94% of rated speed up to 95.9% (1654 rpm). Increased Core Flow (ICF) to over 103% rated core flow is also under review. It is anticipated that ICF conditions will require the recirc pump to operate at full rated speed, 1750 rpm. Modifications have been performed on the D&QC Jet Pump Sensing lines (clamps) and the Dresden 2 Jet Pump riser braces (mitigation clamps) that have technically removed this limit (Task Report T0307) and will permit operation at 100% or more of rated pump speed. Currently, ICF is not permitted and D&QC do not allow core flow above 100% rated core flow. Eventually, after the next fuel reloads of GE-14 or other higher power fuel, an increase in recirc pump speed will be necessary to achieve 100% rated core flow since the resistance to flow across the core will be increasing. It is anticipated that this will result in significantly decreased M-ratio's. GE has identified that there is a threshold at which decreasing M-ratio could potentially result in increased slip joint bypass leakage. This leakage will result in Jet Pump vibrations that could degrade restrainer set-screw clearance and wedge wear. After this leakage initiates, and restrainer brace clearances open, fatigue damage at leaf attachment welds at the vessel wall will quickly accumulate and initiate cracking. There are two accepted mitigation schemes that could be implemented to address this problem. The Jet Pump mixers can be removed and a "labyrinth seal" cut into the lower end of the mixer that inserts into the slip fit at the top of the diffuser. A second option would be to install a mixer/diffuser side load clamp assembly as was performed at LaSalle unit 1 in 2004. (LaSalle RCR # 197310-20). Recirc pump speed control is being updated to a digital system on Quad Cities unit 2 during Q2R17 in 2004. During Q2R18 in 2006, Quad Cities station plans to install	

Attribute	Evaluation	Action?
	Adjustable Speed Drives (ASD) to replace the M-G sets. Eventually, because of obsolescence, the M-G sets will be replaced by ASDs on all four units. Quad Cities 2 has had significant non-EPU speed control issues that have resulted in speed oscillation and unexpected power changes. These conditions have occurred at Dresden to a limit ed extent. There are also larger than normal mismatch between recirc pump speeds and flows on all of the units. On Dresden 2, flows are different between the two loops at matched speeds since the recirc pump impellers are a believed to be different diamet ers. Station records and those of the pump manufacturer, Byron-Jackson, do not agree on the exact impeller diameters/configurations. Tech Specs require that speed be controlled to limit mismatch. The recirc motor generator set (M-G) and the recirc pump motor ratings have been reviewed with the system engineers and evaluated under Task Report T0307. The limiting component for this equipment is the recirc pump motor current. Dresden has increased the rating on the recirc pump motors to 750 amps and up to 775 during coastdown from 724 (T0307 states pump motor limit is 735 A). At these increased currents, the winding temperatures are critical. The temperature limit (GE recommended limit referenced in Operating Procedures) for recirc pump winding temperature is 248°F. The (DAN) ALARM points for winding temperature are 230°F. The motor that is currently running the warmest is D2B peaking at 210°F at 94% speed during the summer under the highest drywell temperatures. There is 20°F of margin below the alarm point. Quad Cities is in the process of increasing their pump motor ratings to this same value. Dresden has had a pre-EPU degraded winding insulation condition on the 3B recirc pump motor. This motor will be replaced during D3R18 in November of 2004. n	
	As a result of these increased ratings, it is projected that pump speeds could be increased to 1654 rpm which corresponds to 100% core flow under EPU conditions. The recirc pump motors are projected to approach 220°F at ICF speeds. At ICF conditions, the M-G set motor current and input power exceeds their rated values by 26 amps (5%) and 277kW (5%). Dresden 2's M-G set ventilation dampers were modified prior to EPU to reduce cooling air recirculation to address the anticipated higher heat load on the M-G motor. The generator output current is limited to the recirc pump motor amperage rating. The Feedwater and Main Steam systems have increased flow to 117% of rated, and therefore the moisture separator and Steam Dryer are experiencing an increase in flow. The steam dryer was modified prior to EPU with the "perforated plate" mod to improve moisture removal performance (Task Report 303 paragraph 3.1.9). The steam dryers at Quad Cities each experienced severe cracking as a result of increased steam line velocities under EPU conditions. The QC dryers were repaired and similar pre-emptive repairs were installed on each Dresden dryer. GE SIL-644 addresses additional inspection activities to address this condition. The feedwater spargers are experiencing increased flow and have been evaluated for these conditions (Task Report 303 paragraph 3.1.10).	
Operating margin: Does the changed parameter reduce the operating margin for a component to an unacceptable level? Does it require that standby equipment now operate during normal conditions such that the margin is reduced to an unacceptable level?	The increase in steam flow has resulted in D&QC Main Steam line velocities that are the highest in the fleet. The effects of high steam velocity are under continued evaluation. The root cause evaluation of the three steam dryer failures experienced at Quad Cities units 1 and 2 is high cyclic loading caused by the effects of high steam velocities. Structural enhancements were installed on the Dresden dryers to address limited cracking identified at the locations predicted in finite element analyses for the Quad Cities. The access hole cover experiences an increase in pressure difference due to flow. The Task report T0303 discusses the bolted cover but is silent on the one remaining original welded design that has not yet been repaired on Dresden unit 3. Therefore, this parameter change cannot be ruled out on a system basis in which known vulnerabilities exists.	Yes, evaluate on compone nt level

Attribute	Evaluation	Action?
Material condition: Does the changed parameter have an adverse effect on the material condition of a component such that the performance of the component is significantly affected or the margin for degradation is reduced to an unacceptable level? Consider whether the material condition was degraded pre-EPU and whether EPU could aggravate this degradation unacceptably.	It is postulated that the steam dryers were not degraded prior to EPU, and that flaws initiated at high steam velocities under EPU conditions. Quad Cities unit 1 has a clamp repair installed on a jet pump riser thermal sleeve to elbow weld (RS-1). This repair was reviewed for under task report TO303 (paragraph 31.1.11) and determined to be acceptable. Other RS-1 flaws are present including the flaw on Dresden 2 Jet Pump 15/16. An existing GE flaw analysis (GE-NE-B13-02044-00) does not evaluate this weld for ICF or EPU conditions. All BWR3 style Jet Pump beams on all four units have recently been replaced. During D2R18, after one cycle of EPU condition operation, replaced 17 original BWR3 beams. Cracking has been identified on Core Spray piping on Quad Cities 1 and Dresden 2 and 3. The effect of increased Core flow would be increased fluid drag across the core spray piping. The "flaw handbooks" for Core Spray contain analyses for load combinations including this parameter, DRG1, and address the effects of increased flow under EPU conditions (D2 report, typical of four reports, GENE-B13-021326-00-01 revision 1). The Dresden and Quad Cities core shrouds either contain or are assumed to contain large flaws in the horizontal welds. The shroud repair hardware was evaluated for the change in loads due to EPU. The increased faulted condition loads in the middle lateral springs and the tie-rod assembly exceeded the original shroud repair faulted condition design loads. The increased loads however are within the allowable stresses in the faulted condition (Appendix A, Task Report T0303). See "Component Ratings" and "Operating Margin" Attributes (above)	Yes, for RS-1 weld flaws
Load and load combinations: Does the changed parameter result in loads on a component (e.g., thermal or dynamic) such that the margin to failure is reduced to an unacceptable level?	Yes. The steam dryers have already been affected by this parameter change. See "Component Ratings", "Material Condition" and "Operating Margin" Attributes (above)	Yes
Environmental conditions and environmental qualification: Does the changed parameter have an adverse effect on a component relative to EQ, for example, due to changes in pressure, temperature, humidity, corrosiveness, EMI or nuclear radiation (including duration exposure)?	An increase in core flow does not affect the environmental qualification of any equipment. See "Component Ratings", "Material Condition" and "Operating Margin" Attributes (above) N/A	N/A
Interface considerations: Does the changed parameter have a potentially adverse effect on an interfacing system (including functional and physical interfaces)? [If "Yes," this information must be communicated to the EPU review team reviewing the affected system.]	Increased Main Steam Flow will be evaluated under at least the Main Steam, EHC and Turbine systems. Increased Feedwater flow will be evaluated in the Feedwater and Condensate systems reviews. Recirc runback, a feature of feedwater level control, is discussed under the Feedwater Level control (643 system) evaluation. N/A See "Component Ratings", "Material Condition" and "Operating Margin" Attributes (above)	N/A

Attribute	Evaluation	Action?
Hydraulic considerations: Does the changed parameter have an adverse effect on pump net positive suction head, allowable pressure drop, allowable fluid velocity, fluid state, margin to flashing or steam quality?	Yes, this is the parameter under consideration in this document. All of the items listed under this parameter are under consideration. These are discussed above under the "Component Ratings" , "Material Condition" and "Operating Margin" Attributes.	Yes
Mechanical/structural considerations: Does the changed parameter have an adverse effect on component vibration, shock, reaction forces, equipment foundations or pipe supports?	Yes. Increased Main Steam line velocities are the parameter that has resulted in cyclic pressure variations that initiated cracking and fatigue resulting in excessive damage to the Quad Cities dryers. Also, during the forced QC outages, evidence of damage to Main Steam line components due to vibration were observed and evaluations are in progress. See "Component Ratings", "Material Condition" and "Operating Margin" Attributes (above)	Yes
Electrical considerations: Does the changed parameter have an adverse effect on component power, voltage, raceway requirements, fault current, electrical insulation or motor requirements? Does the changed parameter adversely affect interfacing electrical systems (e.g., motor control center)?	Recirc pump and M-G set electrical rating are discussed under the "Component Ratings" above. See "Component Ratings", "Material Condition" and "Operating Margin" Attributes (above)	Yes
Instrumentation and control considerations: Does the changed parameter affect component setpoints, instrument ranges or calibrations?	Yes, Main Steam and Feedwater system flow instrumentation was affected. See "System Interface" Attributes (above). Reactor level and recirc system instrumentation was not affected. Quad Cities is replacing their recirc speed control equipment which is discussed under the "Component Ratings" Attributes (above). Setpoints for reactor and recirc instrumentation are not affected other than the addition of recirc runback, a feature of feedwater level control, and is discussed under the evaluation.	Yes

EXTENT OF CONDITION - EVALUATION CRITERIA
VULNERABILITY REVIEW

System: Reactor Recirculation (0200) and Reactor Internals
Parameter(s): Decreased temperature (-2°F) of reactor coolant
Component(s): N/A
Subcomponent(s): N/A

Instructions: The purpose of this form is to assess potential vulnerabilities related to a changed system parameter or parameters. For each attribute, the associated questions should be evaluated relative to the effect of the changed parameter on a system component or components. For the system level evaluation, if "Yes" is indicated in the "Action?" column, a component level evaluation is required for that changed parameter. For the component level evaluation, if "Yes" is indicated, the potential vulnerability must be evaluated for potential failure mechanisms and a risk ranking must be assigned.

Attribute	Evaluation*	Action?
Component ratings: Does the changed parameter exceed a component rating or design value, e.g., with regard to capacity, duty, pressure, flow, temperature, fluid state, fluid chemistry or voltage?	No issues identified	No
Operating margin: Does the changed parameter reduce the operating margin for a component to an unacceptable level? Does it require that standby equipment now operate during normal conditions such that the margin is reduced to an unacceptable level?	No issues identified	No
Material condition: Does the changed parameter have an adverse effect on the material condition of a component such that the performance of the component is significantly affected or the margin for degradation is reduced to an unacceptable level? Consider whether the material condition was degraded pre-EPU and whether EPU could aggravate this degradation unacceptably.	No issues identified	No
Load and load combinations: Does the changed parameter result in loads on a component (e.g., thermal or dynamic) such that the margin to failure is reduced to an unacceptable level?	No issues identified	No
Environmental conditions and environmental qualification: Does the changed parameter have an adverse effect on a component relative to EQ, for example, due to changes in pressure, temperature, humidity, corrosiveness, EMI or nuclear radiation (including duration exposure)?	No issues identified	No
Interface considerations: Does the changed parameter have a potentially adverse effect on an interfacing system (including functional and physical interfaces)? [If "Yes," this information must be communicated to the EPU review team reviewing the affected system.]	No issues identified	No
Mechanical/structural considerations: Does the changed parameter have an adverse effect on component vibration, shock, reaction forces, equipment foundations or pipe supports?	No issues identified.	No
Hydraulic considerations: Does the changed parameter have an adverse effect on pump net positive suction head, allowable pressure drop, allowable fluid velocity, fluid state, margin to flashing or steam quality?	No issues identified.	No
Electrical considerations: Does the changed parameter have an adverse effect on component power, voltage, raceway requirements, fault current, electrical insulation or motor requirements? Does the changed parameter adversely affect interfacing electrical systems (e.g., motor control center)?	No issues identified.	No
Instrumentation and control considerations: Does the changed parameter affect component setpoints, instrument ranges or calibrations?	No issues identified.	No

Attribute	Evaluation*	Action?
Reliability and service life considerations: Does the changed parameter result in additional component cycles, changes in maintenance requirements (e.g., online maintenance changed to outage maintenance) or increased wear?	No issues identified.	No

* This evaluation should consider previous evaluations such that vulnerabilities that have already been identified and dispositioned (as verified by the review team) are not identified as vulnerabilities on this worksheet.

EXTENT OF CONDITION - EVALUATION CRITERIA VULNERABILITY REVIEW

System: Recirculation (0200)
Parameter(s): Increased duty of Recirc Pumps and M-G Sets
Component(s): N/A
Subcomponent(s): N/A

Instructions: The purpose of this form is to assess potential vulnerabilities related to a changed system parameter or parameters. For each attribute, the associated questions should be evaluated relative to the effect of the changed parameter on a system component or components. For the system level evaluation, if "Yes" is indicated in the "Action?" column, a component level evaluation is required for that changed parameter. For the component level evaluation, if "Yes" is indicated, the potential vulnerability must be evaluated for potential failure mechanisms and a risk ranking must be assigned.

Attribute	Evaluation*	Action?
Component ratings: Does the changed parameter exceed a component rating or design value, e.g., with regard to capacity, duty, pressure, flow, temperature, fluid state, fluid chemistry or voltage?	Theoretical increases only. Instrumentation does not show any changes in flow or temperature due to size of system compared to increased heat loads.	No
Operating margin: Does the changed parameter reduce the operating margin for a component to an unacceptable level? Does it require that standby equipment now operate during normal conditions such that the margin is reduced to an unacceptable level?	No issues identified. Other EPU-EOC system reviews will identify any loss of margin due to possible TBCCW parameter changes.	No
Material condition: Does the changed parameter have an adverse effect on the material condition of a component such that the performance of the component is significantly affected or the margin for degradation is reduced to an unacceptable level? Consider whether the material condition was degraded pre-EPU and whether EPU could aggravate this degradation unacceptably.	No issues identified.	No
Load and load combinations: Does the changed parameter result in loads on a component (e.g., thermal or dynamic) such that the margin to failure is reduced to an unacceptable level?	No issues identified.	No
Environmental conditions and environmental qualification: Does the changed parameter have an adverse effect on a component relative to EQ, for example, due to changes in pressure, temperature, humidity, corrosiveness, EMI or nuclear radiation (including duration exposure)?	No issues identified.	No
Interface considerations: Does the changed parameter have a potentially adverse effect on an interfacing system (including functional and physical interfaces)? [If "Yes," this information must be communicated to the EPU review team reviewing the affected system.]	No issues identified.	No
Mechanical/structural considerations: Does the changed parameter have an adverse effect on component vibration, shock, reaction forces, equipment foundations or pipe supports?	No issues identified.	No
Hydraulic considerations: Does the changed parameter have an adverse effect on pump net positive suction head, allowable pressure drop, allowable fluid velocity, fluid state, margin to flashing or steam quality?	No issues identified.	No
Electrical considerations: Does the changed parameter have an adverse effect on component power, voltage, raceway requirements, fault current, electrical insulation or motor requirements? Does the changed parameter adversely affect interfacing electrical systems (e.g., motor control center)?	No issues identified.	No
Instrumentation and control considerations: Does the changed parameter affect component setpoints, instrument ranges or calibrations?	No issues identified.	No

Attribute	Evaluation*	Action?
Reliability and service life considerations: Does the changed parameter result in additional component cycles, changes in maintenance requirements (e.g., online maintenance changed to outage maintenance) or increased wear?	No issues identified.	No

* This evaluation should consider previous evaluations such that vulnerabilities that have already been identified and dispositioned (as verified by the review team) are not identified as vulnerabilities on this worksheet.

7. Interview Results (Project Instruction Attachments 2 & 3)

Recirculation System and Reactor Internals interview forms are attached.

Attachment 3

Questionnaire for Interviews of Dresden and Quad Cities Personnel

D/Q EPU EOC INTERVIEWS

Date: 2-12-04

Interviewer(s): TEAM 1: John Freeman, Bob Geier, Don Knecht (GE)

8. Participants	9. Title
Bill Poppe (DR)	Bob Ross (GE)
Greg Houldson (Quad Cities)	Terry McIntyre (GE)
Gary Sozzi (GE)	

Topic(s): Recirc System portion of system 0200,

The purpose of this interview is to identify potential EPU-related vulnerabilities that could result in one or more of the following:

- License Event Report
- Engineered Safety Features Actuation
- Reactor Scram
- Plant power derate
- Unplanned entry into Technical Specification
- Operator "Work Around" that increases the risk to one of the above events.
- Unexpected accelerated equipment that increases the risk to one of the above events.

Suggested Interview Groups:

- **D/Q Plant Personnel:**
  - System Engineers
  - Reactor Engineers
  - Program Engineers
  - Design Engineers
  - Maintenance Personnel
  - Operations Personnel
- **Exelon Corporate Technical Staff**
- **Plant personnel** for utilities that are currently in the EPU evaluation process or have completed partial or full EPU implementation
- **NSSS and AE Personnel**

Note: Responses to the interview questions should consider different operating modes/configurations: normal operation (summer and winter); components out of service (e.g., FWH string, FW pump); ECCS conditions.

Systems: Recirculation System		
	Question	**Response**
1	How has system operation changed following the EPU? What parameters have changed (pressure, temperature, flow, fluid state, moisture content)?	Greg: essentially no changes. EPU introduced MELLLA, whereby core rod lines are higher flows are decreased for a given power level. Temperatures are not changed. No real problem except flow control issues with MG sets, not EPU related. RR runback mod was installed. One of 2 loops had a runback Oct 7, 2003, CR#179699, loose wire had runback Unit 1which not EPU related John: has bi-stable flow been observed (ref: SIL467)? Greg: No Bob G: What are you monitoring in "sysmon", have you seen anything with pressures that can link main steam pressure oscillations to recirc? Greg: Monitoring 4 pages of parameters on spreadsheet including seal pressure, motor temp, winding temps, most important parameter is MG amps, fluid coupler temp, winding temp, bearing temps, oil temps. Core plate DP is monitored 5 seconds and is in the PI system tag QDC_1V_C101, C201 John: Use PI or daily logs? Greg: Both get out weekly or every other day. Biggest concern is vibration on Main Steam. Have no evidence of any (vibration) on recirc but haven't looked in the drywell either. Don: Any spectral vibration analysis? Greg: No. Recirc pumps are instrumented with 2 axis prox probes and 3 axis accelerometers 3 axes. On the PI system Unit 1 has PI data point, this April U2 will have it, too. Now they go hook up to wires and download data for U2. CMO group (rotating equipment specialists) does this monitoring and evaluates output. Don: Any voltage sagging for the equipment (EPU has 3 feed pumps and recirc pumps on same buses)? Greg: Nothing significant.
2	Is system operation since the EPU as expected? Have there been any unexpected problems, disappointments, concerns or	John F: Is QC planning to use ICF? Greg: Recirc pump speed is limited by amps on M-G motor. This will be evaluated and re-rated.

Systems: Recirculation System		
	Question	**Response**
	issues?	When the new limits are documented, can do ICF. Bill P: RR speed has increased over the years , will 3 core change GE14 cause us to reach this cliff?
3	Was system operation post-EPU affected by changes in the performance of other systems?	
4	What components were affected by the EPU—either directly by changes in system parameters or indirectly by changes in interfacing systems?	Don K: Loop select plant? RR discharge valves slightly diff op conditions affect valves? Greg: Yes it is a loop select and no issues with RR discharge temp effects on the valve, per Marty Santic, previously had RR system. No changes needed.
5	What modifications were made to the system to support operation at the EPU conditions? • What failure modes or vulnerabilities were addressed by the modifications? • Did the modifications (as implemented) create any new failure modes or vulnerabilities? • Were you satisfied with the modification as implemented? • Did the modifications (as implemented) adequately address the existing material condition issues? • Has system performance post-modification been satisfactory	John: what's up with that runback mod? Greg: Runback mod was changed for EPU, bus loading and HVAC. Implemented greater than 5% per second RR runback. Motors maximum might be at 8% per second. (Actual tested from 94% to 70% in 4.5 seconds for positioner stroking, not actual core flow). (*This is consistent with teams knowledge*). Runbacks do not follow MELLLA line, must insert CRAM arrays. Runback mod addressed FW pump trip and plant scram avoidance. New Vulnerability? Redundant logic addressed single failures, if it does work, then it will scram. Run back was added to prevent scram. But does the mod add any new vulnerability? Greg: last October a single loose screw caused a runback. Wiring has terminals that can be loose too. Adding Digital RR control that is single failure proof on a system that is not single failure proof. Dresden did not experience the same number of problems with RR control as Quad has with pump speed control and did not need this upgrade as badly as Quad.. Ross: New equipment is more reliable than old so no new vulnerabilities introduced. Warning to QC station: Columbia's RR logic was faster than old logic and the faster rate could trip on low temperature suction line and trip on interlock on lookup of subcooling table (not familiar with such trip function). Greg: All JP data will be fed into digital RR control,

Systems: Recirculation System		
	Question	Response
		it will be tested in simulator before installation. Plan to put in Adjustable Speed Drive, install during Q2R18 in new location. Abandoning old MG? Yes, after outage. Size of new device(s)? Greg: Almost as big as MGs. GE tested a real MG compared to ASD. Replaces motor and fluid drive. Peach Bottom ordered, tested, received but did not install. Problem was that an outage duration of 35 days was required to remove and set new units in the place of M-G sets. PB is now looking for another place to install, same as Columbia. Browns Ferry just installed 2 ASDs. Put in along side old M-Gs

Procedures: Recirculation System, continued		
	Question	**Response**
1	Were operating procedures changed to support operation at EPU conditions? • What failure modes or vulnerabilities were addressed by the procedure changes? • Did the procedure changes create any new failure modes, vulnerabilities or operator challenges? • Were you satisfied with the procedure changes as implemented? • Has system operation with the revised procedures been satisfactory?	Greg: Extensive procedure changes for runback. Operating limits not changed. Power Flow map changed. No changes to cavitation lines. Failure modes or operator challenge? No. Satisfied with procedures? Greg: Yes pleased with procedures, operator happy.
2	Were PM procedures for systems, components and subcomponents revised to reflect , with regards to frequency and scope, accelerated wear and degradation? • Which PMs were revised? • What was the basis for the changes? • Are there additional PM changes necessary to ensure system and component reliability at the EPU conditions?	Greg: PM frequencies not changed for EPU. Examples: scoop tube position remains at 4 years, Motor work every outage, instrumentation calibration, field breakers on 6-10 year cycle, oil cooler cleaning online and valve motors with some during shutdown.
3		
4		
5		

Material Condition: Recirculation System, continued		
	Question	**Response**
1	What are the material condition issues associated with the components in the system?	Greg: Increased vibrations is a outside concern. Replaced Jet pump beams, some history of riser brace cracking. Flawed jet pump (Unit 1 RS-1 weld crack on riser). There is no indication for one jet flow rates broken instrument line. Ross: instrument line break probably happened in installation. Bob G: JPI line broke in 1974, plans to fix next outage. Some clamps were added to JP instrument lines to raise nat frequency to address concern about vane passing frequency. Freeman: QC had RR piping in drywell weld crack, RR suction, What Happened? Greg: I don't know. Bob G: several plants have not replaced recirc piping as was done on D3 and have several, maybe 20-30 weld overlays. This is common for the industry. IGSCC seems to have been arrested with H2 and Noble Chem. We don't believe that this is an EPU concern. Bill P: Plant's recirc systems are not exactly the same speed. 100% rated speed is 56 Hz at QC. Dresden has 100% speed at 57.5 Hz. Recirc pump impellers: D2a, D3 and QC all same. D2B has 29.5" tapered impeller, 28-9/16" is standard. Changed dia can change pump flow for a given speed and magnitude of vane pass frequency pressure pulse. D&QC are only known plants with B-J 6 vane pumps and therefore unique VPFs.. Greg: talked with Harry Palas about this. Bill P: Dresden has recirc pump shafts with thermal cycling fatigue cracks confirmed by inspections. QC does not have this because seal purge injection installed at a later data and lower flow. May have affected pump bonnets differently, too. Don K: Pi data mis-match on D2 that is not on D3, caused by different impeller diameters? Noticed 8% difference in flow, may not comply with Tech Spec? Bill P: Master manual Controlled by speed from a tachometer reader, flow higher on 2B and pump power Greg: QC used individual manual. Digital system

Material Condition: Recirculation System, continued		
	Question	Response
		will allow a bias in speed. Only two times time had harmonic vibration in 2 MG sets links up on local equipment vibration. Bill P: 55% has flow oscillation also its very bad, and 85% actual indicated speed has harmonic go up 1% speed and it goes away. 3B is loudest resonance. Ross: Min speed coupler has limit cycles not resonance, but changed the min speed set-point. It happens in the coupler, not hydraulic feedback. Yes we have loop selection, pressure switches during startup can be destroyed from rapid oscillation, prefer to get 1.5 psid difference and peg them unbalance to avoid this. Greg: new Digital system will give GPM
2	Does operation at EPU conditions exacerbate any of these material condition issues?	OPEX 1: any thing at other plants that concerns you? Greg : No, just weld overlay and small bore failures from vibration. Bill P: RR runback not installed at all plants. Don K: JP M-ratio will sag to 1.7 according to Task Reports, but operating at a lower core flow, they have not seen sagging in M-ratio, actually observed higher M-ratios. Ross: M-ratio goes down as core flow goes up. Freeman: John F: Why are we not looking at M-N ratio to compare to design numbers? Ross: Pump flow at a given speed can be used. N-ratio has to be calculated. Plot sqrt of core DP resistance vs time, then look at drive flow required, then you can determine if it changes. If reduced, usually crudding. RR Pump speed vs flow is also a good indication of hydraulic resistance. Freeman do sites look at these? Bill P: Rx engineers look at these Bob G: We have not been focused on tracking M-ratio, by default or not, should have detected the symptoms of LaSalle-type bypass leakage before a similar "JP death spiral" occurs. That is, wedge wear and set-screw gaps. We changed JP beams on all units lately, and looked at 100% of wedges, and have been inspecting Jet Pumps per BWRVIP recommendations. Every 6 years look at 50% of the wedges and set-screws as necessary. Dresden, and

Material Condition: Recirculation System, continued		
	Question	Response
		Quad I believe, has not seen evidence that there is a definite trend as LaSalle has experienced. Bill P: Will GE be providing data from LaSalle on minimum M-Ratio for degradation and recommendations for our JP minimum M-ratios? Ross: Yes they are working on it trying to figure out where the cliff is. McIntyre will find out who is working on it.
3	Was accelerated degradation of components with material condition issues adequately addressed by the EPU Project?	Greg: do not know that any material conditions were addressed by EPU. Bob G: Some conditions directly addressed for EPU: sensing line clamps, and D2 riser brace repair and mitigation clamps addressed anticipated higher speeds of recirc pumps under EPU.
4		John F: Who controls max flow stops Bill P: Nuke engineer is in control of stops but these need to be increased, put up out of the way, electrical , mechanical stops at 108%, also check for FRFI rates assumed in EPU in Task 900.
5		

EPU EOC Interviews – 0200 Reactor Internals

Date: February 5, 2004

Interviewer(s): Don Knecht, Paul Doverspike

Participants	Title
Bob Geier	**Dresden IVVI Engineer**
Keith Moser	**Exelon Asset Management**

Topic(s): system 0200, Reactor Vessel Internals, not including Main Steam or Recirc systems

The purpose of this interview is to identify potential EPU-related vulnerabilities that could result in one or more of the following:

- License Event Report
- Engineered Safety Features Actuation
- Reactor Scram
- Plant power derate
- Unplanned entry into Technical Specification
- Operator "Work Around" that increases the risk to one of the above events.
- Unexpected accelerated equipment that increases the risk to one of the above events.

EPU EOC Interviews – 0200 Reactor Internals

Systems:		
	Question	**Response**
1	How has system operation changed following the EPU? What parameters have changed (pressure, temperature, flow, fluid state, moisture content)?	• Since core differential pressure has increased due to higher power while maintaining steam dome pressure unchanged, pressures are changed by some amount (minimal) in some areas including the bottom head (increased). • Fluence is increased which affects some components (shroud and rpv). • With increased steam and feed flows, there are increased flow velocities through the dryer, moisture separator and feedwater spargers. Inlet quality to the dryer is decreased • The fluid density is increased in the separator and dryer. • Feedwater temperatures are decreased and there is more dilution and therefore there is more NPSH available for the recirc pumps. • Under accident conditions, there is more energy in the core. This would change emergency/faulted loads. For example, during MSL break, there will be an increase in lift loads on the top guide.
2	Is system operation since the EPU as expected? Have there been any unexpected problems, disappointments, concerns or issues?	• Dryer problems are well known. The cause for the most recent and higher intensity Low Frequency Pressure Pulses on Quad Cities 1 and 2, and to a lesser degree, Dresden 3, are under investigation. • Jet Pump diffuser slip joint bypass leakage caused by increasing core dP at LaSalle are being evaluated for D&Q. Set screw gaps were examined after one full EPU cycle during D2R18 in October 2003 and no wedge wear or unanticipated setscrew gaps (one gap at 0.013" was found of the forty set screws examined) were identified. •
3	Was system operation post-EPU affected by changes in the performance of other systems?	• The cause of the second Dryer problem is thought to be increased Main Steam line velocity that has generated acoustic energy. •
4	What components were affected by the EPU—either directly by changes in system parameters or indirectly by changes in interfacing systems?	Existing cracks in Core Spray and Jet Pumps are being evaluated more conservatively in the fuel analysis, and therefore repairs will be necessary sooner. The D3R18 LSRs are technically required due to EPU.
5	What modifications were made to the system to support operation at the EPU conditions? • What failure modes or vulnerabilities were addressed by the modifications? • Did the modifications (as implemented) create	Initially, the only steam dryer modification was the perforated plate mod. The purpose of this change was to improve steam outlet quality. • At increased steam flows, the original dryer would not have performed acceptably. Which would have accelerated moisture erosion wear on steam

Systems:		
	Question	**Response**
	any new failure modes or vulnerabilities? • Were you satisfied with the modification as implemented? • Did the modifications (as implemented) adequately address the existing material condition issues? • Has system performance post-modification been satisfactory	line and turbine components. No dryer failure modes were addressed by this modification. • As was demonstrated at Quad Cities, no new failure modes were created by the addition of the perforated plates, since other dryer parts could cause MSL restriction. • The dryer performance after the modification exceeded all expectations. • There were no known existing material condition issues with the dryer pre-EPU. Two modifications were performed on Jet Pumps. Prior to EPU operation, the Sensing line clamps were installed to permit operation at speeds above 94%. The sensing lines were vulnerable to recirc pump vane passing frequency (VPF) induced fatigue damage. After the first cycle of EPU, the D2 Jet Pump Riser Brace mod repaired the attachment of one Jet pump to the RPV. The other braces were enhanced to eliminate them from VPF excitation. • Recirc Pump VPF renders riser brace attachment welds vulnerable to fatigue failure during operation in some Recirc pump speed ranges. • No new vulnerabilities were created. • The mods allows removal of the maximum speed restriction on Recirc pump speeds and allows operation at ICF. The exclusion zone between 67% and 82% of speed was lifted by the brace mod. • Elimination of the speed exclusion zones permits operational flexibility not previously possible.

EPU EOC Interviews – 0200 Reactor Internals

Procedures:		
	Question	**Response**
1	Were operating procedures changed to support operation at EPU conditions? • What failure modes or vulnerabilities were addressed by the procedure changes? • Did the procedure changes create any new failure modes, vulnerabilities or operator challenges? • Were you satisfied with the procedure changes as implemented? • Has system operation with the revised procedures been satisfactory?	Several changes were made to support D2 with the known Jet Pump deficiencies. Dryer performance monitoring and contingency planning was in-place following the degradation at Quad Cities.
2	Were PM procedures for systems, components and subcomponents revised to reflect, with regards to frequency and scope, accelerated wear and degradation? • Which PMs were revised? • What was the basis for the changes? • Are there additional PM changes necessary to ensure system and component reliability at the EPU conditions?	Inspection frequencies of BWRVIP components were not revised. Components such as the dryer were not covered by a BWRVIP inspection and Evaluation (I&S) document and so the inspections were driven by experience at Quad Cities. • No PMs were revised. • BWRVIP (-41) recommendations for Jet Pumps are under review by an owners group committee.

EPU EOC Interviews – 0200 Reactor Internals

Material Condition:		
	Question	**Response**
1	What are the material condition issues associated with the components in the system?	Jet Pump riser welds (RS-1) are currently cracked on D2. QC1 has several brace and RS-1 welds cracked with mechanical clamps installed. Core Spray piping is cracked at various locations on D2, D3 and QC1. The shrouds have horizontal welds cracks with tie-rod type repairs installed on all four units
2	Does operation at EPU conditions exacerbate any of these material condition issues?	The LOCA analysis with GE 14 under EPU conditions allows a reduced amount of leakage and therefore margins are reduced.
3	Was accelerated degradation of components with material condition issues adequately addressed by the EPU Project?	Assumptions (probably correct) about pre-EPU material condition of the dryer were made. Many components are not easily accessible for inspection (lower plenum components) and are now being subject to changed operating conditions. Assumptions of the pre-EPU material conditions and accelerated degradation can not be validated or addressed.

EPU EOC Interviews – 0200 Reactor Internals

10. Database Report (Project Instruction Attachment 1)

A component-level review was not performed. Accordingly, there is no database report.

11. Miscellaneous Documents

The following documents are attached:

- Previously Identified Issues for

Previously Identified Issues for RBCCW System

- Post-EPU, if SW temperatures are too high, 3 SDC and RBCCW heat exchangers (instead of 2 each) are required for Dresden cooldown in 24 hours. [Task Report] *This is an infrequent operation, and would require River temperatures >95F to have an effect on the Station. However, it would not cause a scram, derate, LER, etc. The EPU-EOC review is limited to normal EPU power conditions and those plant line-ups and configurations that are not considered unusual.*
- For the emergency full core offload event at QC, the swing heat exchanger will have to be aligned. [Task Report and Pinch Points] *The system engineer (Davis) indicated that Exelon performed another evaluation that concluded the swing heat exchanger would not be needed because of the RHR Fuel Pool assist mode, which was neglected in the task report. This was confirmed per a separate interview with the EPU Engineer (Gerry Frizzel). Rev 1 of the Task Report had stated that the design max river temperature was changed to 90 deg F and the EPU Team took another look at the issue. They concluded that we did not need to operate the 5th RBCCW HX in this configuration, and that the design basis maximum river water temperature for the Service Water system would stay at 95 deg F like it had been for the first 30 years of plant life. Reference Action Tracking item #57155-07.*
- The increased heat load exceeds the design value for QC (both pre- and post-EPU). [Task Report and Pinch Points] *The table above summarizes the predicted changes in normal heat loads, based on the GE task report prior to the actual implementation of EPU. Note that the task report evaluates the system under theoretical design conditions (e.g. River temperature of 95F, etc.). The actual operating conditions shows that there is very little change in RBCCW and TBCCW temperatures (pre-EPU vs. post-EPU). Thus there have been no long-term effects on system components. This is confirmed by other system EPU-EOC system reviews that have RBCCW or TBCCW as a support system.*
- Need to confirm that the note on page 1-4 of the Task Report was added to the Dresden plant cooldown procedures. [Task Report] *Per the System Engineer, this was completed; and has no impact on EPU-EOC.*

Previously Identified Issues for TBCCW System

- "Calculated heat load on TBCCW due to bus duct is greatly underestimated in TR0607." [pilot iso-phase bus duct review] *This concern was address by the Iso-phase Bus Duct Cooling system review.*

ATTACHMENT 3

Extended Power Uprate Extent of Condition Reviews – Strategic Items, Inspections and Modifications, and Analyses, Studies, and PM Reviews

EPU Extent of Condition Review Recommendations
(Strategic Items)

#	Recommendation	Owners		Dresden						Quad						Total Cost
		DRE	QDC	Unit 3			Unit 2			Unit1			Unit 2			
				Date	Cost	Action Tracking	Date	Cost	Action Tracking	Date	Cost	Action Tracking	Date	Cost	Action Tracking	
1	Procure a spare Recirc pump motor and replace in accordance with the High Voltage motor maintenance template once every ten years.	Poppe	Gravert	12/12/06		216163-07	12/12/05		239153-07	04/29/07		216402-07	04/28/08		240081-07	0
2	Adjustable speed drives (ASD's) are being considered for installation at Quad Cities station. This modification should be supported for all four units.	Poppe	Gravert	12/12/06		216163-08	12/12/07		239153-08	04/29/07		216402-08	04/30/06		240081-08	0
3	Continue implementation and do not delay the Main Generators Material Condition Improvement Plan (MCIP) and the "Long Term Asset Management Strategy". This includes both the Dresden and Quad Cities plants.	Haarhoff	Bridges	12/12/06		216163-16	12/12/05		239153-16	04/29/07		216402-16	04/28/08		240081-16	0
4	**(Dresden Only)** Develop a cost effective solution to reduce the DP across the Unit 3 C2 FW heater.	Steckhan	N/A	12/12/06		216163-44	NA		NA	NA		NA	NA		NA	0
5	**(Dresden Only)** Establish the cause of increased Fe in Unit 3. Increase O2 levels in Unit 3, coat susceptible iron source or increase prefilter capacity	Steckhan	N/A	12/12/06		216163-49	NA		NA	NA		NA	NA		NA	0
6	Develop a plan to allow increasing the quantity in the outage template of Control Rod vacuuming if there is evidence of increased iron concentration	Chenell	Schumacher	12/15/04		240191-16	12/15/04		240191-16	12/15/04		240194-11	12/15/04		240194-11	0
7	Replace existing "scab" plates that cover the heater shell thinned areas with replacement shell sections on the B feedwater heaters.	Bezouska	Rolfes	12/12/06		216163-82	12/12/07		239153-82	04/29/07		216402-28	04/28/06		240081-82	0
8	Restore original design margin and eliminate abnormal operating condition for the heater drain system.	Bezouska	Rolfes	12/12/06		216163-84	12/12/07		239153-84	04/29/07		216402-21	04/28/08		240081-84	0
9	**(Dresden Only)** Modify the design of Circulating Water pump's discharge valves to allow more flow to pass without oscillations.	McAllister	N/A	12/12/06		216163-93	12/12/07		239153-93	NA		NA	NA		NA	0

EPU Extent of Condition Review Recommendations
(Strategic Items)

10	Identify the required set point tolerence for the MSSV that accounts for the historical performance of the valves.	Testin	Swart	12/12/04		240191-06	12/12/04		240191-06	12/12/04		240194-04	12/12/04		240194-04	0
	Total Cost				**0**			**0**			**0**			**0**		**0**

EPU EXTENT OF CONDITION REVIEW
RECOMMENDATIONS
(Inspections and Modifications)

#	Recommendation	Owners		Dresden						Quad						Total Cost
		DRE	QDC	Unit 3			Unit 2			Unit1			Unit 2			
				Date	Cost	Action Tracking	Date	Cost	Action Tracking	Date	Cost	Action Tracking	Date	Cost	Action Tracking	
1	Perform a one-time visual examination of the shroud head bolt locking pin window for evidence of wear.	Testin	Phares	12/12/04		216163-01	12/12/05		239153-01	04/29/05		216402-01	04/28/06		240081-01	
2	Perform a one-time visual inspection of the shroud head bolt mid-span and top support ring gussets at next refuel.	Testin	Phares	12/12/04		216163-02	12/12/05		239153-02	04/29/05		216402-02	04/28/06		240081-02	
3	Perform a one-time sample inspection of the separator standpipe welds to the top of the shroud head for fatigue cracking.	Testin	Phares	12/12/04		216163-03	12/12/05		239153-03	04/29/05		216402-03	04/28/06		240081-03	
4	(Dresden Only) Inspect any Jet Pump Beams that are > 4 years old.	Testin	N/A	12/12/04		240191-01	NA		NA	NA		NA	NA		NA	
5	Perform a one-time visual inspection of the Feedwater spargers and end bracket pin hardware for evidence of Vibration.	Testin	Phares	12/12/04		216163-04	12/12/05		239153-04	04/29/05		216402-04	04/28/06		240081-04	
6	Accelerate the BWRVIP-41 recommended inspection of restrainer gate wedges (WD-1) to verify that there is no evidence of vibration and wear.	Testin	Phares	12/12/04		216163-06	12/12/05		239153-06	04/29/05		216402-06	04/28/06		240081-06	
7	After Recirc pumps speeds are increased to levels not previously attained, inspect the Recirc Loop Flow sensing lines and other small-bore piping in the drywell that are attached to the recirc system during each of the upcoming outages.	Poppe	Gravert	12/12/04		216163-09	12/12/05		239153-09	04/29/07		216402-09	04/28/08		240081-009	
8	Replace MSL flow DP switches with a pressure transmitter and digital trip unit. Option: Replace one switch in each MS line and inspect for signs of degradation and adjust PM accordingly	Poppe	Gunter	12/12/04		216163-10	12/12/05		239153-10	04/29/05		216402-10	04/28/06		240081-10	
9	Perform a one-time inspection of the internals of Generator Stator Cooling Temperature Controller for any degradation caused by high vibration. Adjust PM accrordingly	Haarhoff	Bridges	12/12/04		216163-17	Complete		239153-17	04/29/05		216402-17	04/30/06		240081-17	
10	Perform a one-time boroscope examination of all 4 condensate pump impellers during the next outage. Adjust PM accrodingly	Mourikes	Kimmel	12/12/04		216163-22	12/12/05		240191-13	04/29/05		216402-22	04/28/06		240081-22	
11	Inspect one feedwater discharge check valve at the next outage and perform check valve monitoring to manage the potential for accelerated wear on the hinge bushings.	Testin	Kunzmann	12/12/04		216163-24	12/12/05		239153-24	04/29/05		216402-24	04/28/06		240081-24	

EPU EXTENT OF CONDITION REVIEW
RECOMMENDATIONS
(Inspections and Modifications)

#	Recommendation	Owners		Dresden						Quad						Total Cost
		DRE	QDC	Unit 3			Unit 2			Unit1			Unit 2			
12	Increase the frequency of the Reactor Feed Pump Seal replacement from 4 to 2 years.	Bezouska	Rolfes	12/12/04		216163-25	12/12/05		239153-25	04/29/05		216402-25	04/28/06		240081-25	
13	Perform a one-time inspection of the relays mounted in feedwater system panels located near the feedwater pump room.	Bezouska	Rolfes	12/12/04		216163-26	12/12/05		239153-26	04/29/05		216402-26	Complete		240081-26	
14	Perform UT for general pipe wall thinning near an installed thermowell on heater string downstream of a "B" or "C" heater.If erosion is occurring, then enter into FAC program	Sisk	Knapp	12/12/04		216163-29	12/12/05		239153-29	04/29/05		216402-29	04/28/06		240081-29	
15	Develop a plan and perform one-time vibration measurements at power levels of 777, 792, 807, 822, 837, 852, 867, 882 and 912 Mwe on susceptible small bore piping and instrument taps in feedwater and condensate booster Systems. Assure that data is characterized in terms of feedwater flow rates, not electrical output	Loch	Alguire	12/12/04		216163-31	Complete		239153-31	08/01/04		216402-31	Complete		240081-31	
16	Develop a plan and perform detailed inspection walk down of condensate, condensate booster and feedwater systems similar to Main Steam system piping walk down.	Loch	Alguire	12/12/04		216163-32	12/12/05		239153-32	04/29/07		216402-32	04/30/06		240081-32	
17	Obtain vibration data and perform a one-time inspection of one of the Reactor Feed Pump Min Flow Valves and operator/actuator. Adjust PM accordingly	Lintakas	Yost	12/12/04		216163-34	12/12/05		239153-34	04/29/05		216402-34	04/28/06		240081-34	
18	Perform a one-time inspection of the HP heater inlet isolation MOV operator/actuator. Adjust PM accordingly	Lintakas	Mills	12/12/04		216163-35	12/12/05		239153-35	04/29/05		216402-35	04/28/06		240081-35	
19	Obtain vibration reading and perform a one time inspection of the FRV Isolation MOV operator/actuator. Adjust PM accordingly	Lintakas	Mills	12/12/04		216163-36	12/12/05		239153-36	04/29/05		216402-36	04/28/06		240081-36	
20	Inspect valve internals and operator/actuator of one FRVs and the Low Flow Regulating Valve per outage	Lintakas	Yost	12/12/04		216163-37	12/12/05		239153-37	04/29/05		216402-37	04/28/06		240081-37	
21	Perform a one-time inspection of the LP heater string inlet MOV operator/actuator. Adjust PM accordingly	Lintakas	Mills	12/12/04		216163-38	12/12/05		239153-38	04/29/07		216402-38	04/28/08		240081-38	
22	Obtain vibrating reading and inspect one Condensate Booster Min Flow valve operator/actuator	Lintakas	Yost	12/12/04		216163-39	12/12/05		239153-39	04/29/07		216402-39	04/28/08		240081-39	
23	Inspect valve and operator/actuator of one FW Discharge MOV (external torque check and visual)	Lintakas	Mills	12/12/04		216163-40	12/12/05		239153-40	04/29/07		216402-40	04/28/08		240081-40	

EPU EXTENT OF CONDITION REVIEW
RECOMMENDATIONS
(Inspections and Modifications)

#	Recommendation	Owners		Dresden						Quad						Total Cost
		DRE	QDC	Unit 3			Unit 2			Unit1			Unit 2			
24	Perform internal inspection of the Feedwater Suction relief valves to determine if internal wear is occurring.	Testin	Kunzmann	12/12/04		216163-41	12/12/05		239153-41	04/29/05		216402-41	04/28/06		240081-41	
25	(Quad Cities Only) - Reclassify the feedwater heaters to a category 1 and develop a time directed task for internal inspections (eddy current testing).	N/A	Potts	NA		NA	NA		NA	04/29/05		216402-68	04/28/06		280194-08	
26	(Dresden Only) Inspect one demineralizer rubber boot and adjust the preventive maintenance frequency based on the inspection results	Steckhan	N/A	12/12/04		216163-46	12/12/05		239153-46	NA		NA	NA		NA	
27	(Dresden Only) Replace biased O2 probe in D3 FW piping and obtaine more accurate measurement of FW O2.	Steckhan	N/A	12/12/04		216163-47	NA		NA	NA		NA	NA		NA	
28	Perform Flow Accelerated Corrosion inspections at the condensate booster pump discharge nozzles and downstream of flow element 3441-27. Determine inspection frequency based on results	Slsk	Knapp	12/12/04		216163-51	12/12/05		239153-51	04/29/05		216402-51	04/28/06		240081-51	
29	Inspect Offgas condenser Division Plate Bypass Valve	Bonomo	Myers	12/12/04		216163-60	12/12/05		239153-60	04/29/05		216402-60	04/28/06		240081-60	
30	Inspect Gland Seal Condenser Division Plate Bypass Valve for Evidence of wear	Bonomo	Myers	12/12/04		216163-62	12/12/05		239153-62	04/29/05		216402-62	04/28/06		240081-62	
31	Monitor thermal aging of the following instruments in the Preventative Maintenance program: FT-144, PI-142, PI-24, PT-23, TI-141	Bonomo	Myers	12/12/04		216163-64	12/12/05		239153-64	04/29/05		216402-64	04/28/06		240081-64	
32	During scheduled HCU overhauls, inspect the 135 and 136 filters to determine if they are excessively plugged. Adjust PM based on inspection results.	Chenell	Schumacher	12/12/04		216163-72	12/12/05		239153-72	04/29/05		216402-69	04/28/06		240081-72	
33	Inspect valve, actuator and positioner on normal LCV 3508 or 3509. Document results in the work packages.	Bezouska	Rolfes	12/12/04		216163-75	12/12/05		239153-75	04/29/05		216402-50	04/28/06		240081-75	
34	Replace Nitrile (Buna-N) material in the feedwater heater drain valves, positioners, o-rings, etc. with a more temperature resistant material (e.g. viton).	Bezouska	Rolfes	12/12/06		216163-76	12/12/07		239153-76	04/28/07		216402-49	04/28/08		240081-76	
35	(Quad Cities Only) Perform inspections as part of the PM program on the feedwater heater start-up and normal operating motor operated vent valves actuators at a minimum frequency of every 6 refuel cycles.		Mills	NA		NA	NA		NA	04/28/07		216402-46	04/28/08		216402-47	

EPU EXTENT OF CONDITION REVIEW
RECOMMENDATIONS
(Inspections and Modifications)

#	Recommendation	Owners		Dresden						Quad						Total Cost
		DRE	QDC	Unit 3			Unit 2			Unit1			Unit 2			
36	Develop an inspection plan and perform detailed inspection walk down of the feedwater heater drain and vent system similar to the Main Steam system piping walk down.	Bezouska	Rolfes	12/12/04		216163-77	12/12/05		239153-77	04/29/05		216402-45	04/28/06		240081-77	
37	Verify sensor temperatures (through thermography, tubing/piping heat dissipation calculations, etc) in level switches and transmitters as noted in S&W EPU report.	Bezouska	Rolfes	08/30/06		216163-78	08/30/06		239153-78	08/30/06		240081-78	08/30/06		240081-78	
38	Perform Feedwater Heater and Flash Tank NDE inspections at a 3-cycle frequency per the Feedwater Heater PCM template for a category 1 component.	Bezouska	Rolfes	12/12/06		216163-81	12/12/07		239153-81	04/28/07		216402-43	04/28/08		240081-81	
39	Perform Examinations and inspections on the D, C and B feedwater heater stainless steel operating vent connections for evidence of IGSCC.	Hall	Rolfes	12/12/04		216163-83	12/12/05		239153-83	04/29/05		216402-27	04/28/06		240081-83	
40	Perform a detailed inspection of the electrical connections and mechanical linkages subjected to turbine control valve vibrations. Check for any evidence of looseness or wear. Adjust PM accordingly.	Chiou	Mendenhall	12/12/04		216163-87	12/12/05		239153-87	04/29/05		216402-19	04/28/06		240081-87	
41	Obtain detailed oscillation data for the turbine controls valves to assess impact on piping vibration.	Chiou	Mendenhall	12/12/04		216163-88	12/12/05		239153-88	04/29/05		216402-18	04/28/06		240081-88	
42	Perform a detailed inspection of a representative LP turbine's inner casing and extraction boxes. Restore the degraded material conditions.	Mourikes	Foster	12/12/04		216163-91	12/12/05		239153-91	04/29/05		216402-15	04/28/06		240081-91	
43	Perform an inspection walk down of the EHC piping and electrical connections and replace degraded support grommets that may be degraded.	Chiou	Mendenhall	12/12/04		216163-92	12/12/05		239153-92	04/29/07		216402-14	04/28/06		240081-92	
44	(Dresden Only) Perform Eddy Current testing of a sample of unstaked tubes at edge of staked region.	McAllister	N/A	12/12/04		216163-97	12/12/05		239153-97	NA		NA	NA		NA	
45	Perform a periodic examination of the extraction piping and components for effects of erosion damage.	Bezouska	Rolfes	12/12/04		216163-98	12/12/05		239153-98	04/29/05		216402-12	04/28/06		240081-98	
46	Reset the Generator Hydrogen Gas pressure LOW alarm to 58 psig and update the Alarm Procedures.	Haarhoff	Bridges	06/01/05		216163-18	06/01/05		216163-18	06/01/05		240081-18	06/01/05		240081-18	
47	Install proxy probe or ultrasonic flow measuring device to attain enhanced performance monitoring for condensate and booster pumps.	Mourikes	Kimmel	12/12/06		216163-23	12/12/05		240191-14	04/29/05		216402-23	04/28/06		240081-23	

EPU EXTENT OF CONDITION REVIEW
RECOMMENDATIONS
(Inspections and Modifications)

#	Recommendation	Owners		Dresden						Quad						Total Cost
		DRE	QDC	Unit 3			Unit 2			Unit1			Unit 2			
48	**(Quad Cities Only)** Incorporate the use of the automatic function for the Condensate/Booster Minimum Flow Control Valve, AO 3401, during normal EPU operation, as designed.	N/A	Kimmel	NA		NA	NA		NA	04/29/05		216402-67	Complete		NA	
49	Redesign and install the Condensate, Condensate Booster and Feedwater System Sample Probes.	Galanis	Porter	12/12/04		216163-30	12/12/05		239153-30	04/29/05		216402-30	04/28/06		240081-30	
50	Install flex hoses on all 4 TBCCW lines to the condensate and condensate booster pump bearing housing and the 3 Reactor Feed Pump seal cooling.	Mourikes	Kimmel	12/12/06		216163-33	12/12/05		239153-33	04/29/05		216402-33	04/28/06		240081-33	
51	Install 2x1 welds for all Feedwater Suction relief valves that are currently without this configuration.	Testin	Kunzmann	12/12/04		216163-42	12/12/05		239153-42	04/29/05		216402-42	04/28/06		240081-42	
52	**(Dresden Only)** Resize and Replace orifice in demineralizer 1/7 bypass flow line for Dresden Unit 3.	Steckhan	N/A	12/12/04		216163-50	NA		NA	NA		NA	NA		NA	
53	Modify the condensate pump seal cooling configuration to take seal cooling water from a point downstream of the demineralizer rather than from the discharge of the condensate pump.	Mourikes	Kimmel	12/12/06		216163-52	12/12/05		239153-52	04/29/07		216402-52	04/28/06		240081-52	
54	**(Dresden Only)** Improve the detectability of Unit 3 conductor high temperature by providing readily accessible temperature monitors on the transformer side of the Iso Phase Bus.	Rivera	N/A	12/12/06		216163-65	NA		NA	NA		NA	NA		NA	
55	**(Quad Citie Only)** Remove the Feed Pump Ventilation inlet dampers.	N/A	Luebbe	NA		NA	NA		NA	04/28/07		216402-65	04/28/08		216402-63	
56	**(Quad Cities Only)** Remove the Recirc M-G Set Ventilation inlet dampers at Quad Cities.	N/A	Luebbe	NA		NA	NA		NA	04/29/05		216402-61	04/28/06		216402-59	
	Total Cost				$0			0			0			0		0

EPU EXTENT OF CONDITION REVIEW RECOMMENDATIONS
(ANALYSES, STUDIES, PM REVIEWS)

#	Recommendation	Owners		Dresden						Quad						Total Cost
		DRE	QDC	Unit 3			Unit 2			Unit1			Unit 2			
				Date	Cost	Action Tracking	Date	Cost	Action Tracking	Date	Cost	Action Tracking	Date	Cost	Action Tracking	
1	Evaluate the original welded D3 Shroud Access Hole Cover to changes in normal (recirculation system flow) and accident (EPU changed) design loads.	Testin	Phares	08/30/05		240191-17	NA		NA	NA		NA	NA		NA	0
2	Establish the value of the appropriate parameter at which slip joint bypass leakage initiates Jet Pump vibration.	Testin	Phares	06/01/05		216163-05	06/01/05		216163-05	06/01/05		240081-05	06/01/05		240081-05	0
3	Provide operator guidance on 'pressure set' adjustments during a load drop.	Poppe	Mendenhall	08/30/05		216163-11	08/30/05		216163-11	08/30/05		240081-11	08/30/05		240081-11	0
4	Assess if it is feasible to increase the 30 psid pressure controller setpoint.	Poppe	Mendenhall	08/30/05		216163-12	08/30/05		216163-12	08/30/05		240081-12	08/30/05		240081-12	0
5	Investigate the calibration accuracy of the Main Steam Flow instrumentation.	Poppe	Swart	08/30/05		216163-14	08/30/05		216163-14	08/30/05		240081-14	08/30/05		240081-14	
6	Re-evaluate the Reactor Heat balance and ThermoKit for Dresden and Quad Cites focusing on: Plant Specific Parameters, Consistency between Reactor and Turbine heat balances. Acceptability of process computer heat balance input parameters: Feedwater Flow, Feedwater Temperature	Reda	Foster	08/30/05		216163-15	08/30/05		216163-15	08/30/05		240081-15	08/30/05		240081-15	0
7	(Dresden Only) Measure Iso Phase Bus cooling fan motor currents, determine the HP, and compare the values to the design values in ELMS. Reanalyze the ELMS if required.	Haarhoff	N/A	03/30/05		216163-66	03/30/05		216163-66	NA		NA	NA		NA	0
8	Upgrade the Dresden and Quad Cities operation, maintenance, and other procedures along with the lesson plan to state the relationship between the generator MVA capability via the curve and the specific Hydrogen Pressure Range of 58 psig to 64 psig.	Haarhoff	Bridges	06/01/05		216163-19	06/01/05		216163-19	06/01/05		240081-19	06/01/05		240081-19	0
9	Assess the feasibility of operating at full EPU power with 2 Reactor Feed Pump and 3 Condensate /Condensate booster pump combination.	Flick	Boline	08/30/05		216163-20	08/30/05		216163-20	08/30/05		240081-20	08/30/05		240081-20	0

EPU EXTENT OF CONDITION REVIEW RECOMMENDATIONS
(ANALYSES, STUDIES, PM REVIEWS)

#	Recommendation	Owners		Dresden						Quad						Total Cost
		DRE	QDC	Unit 3			Unit 2			Unit1			Unit 2			
				Date	Cost	Action Tracking	Date	Cost	Action Tracking	Date	Cost	Action Tracking	Date	Cost	Action Tracking	
10	For current design, perform analysis and then testing to determine the optimum operating conditions at which to start and stop condensate/condensate booster and feedwater pumps.	Galanis	Porter	08/30/05		216163-21	08/30/05		216163-21	08/30/05		240081-21	08/30/05		240081-21	0
11	Perform sizing calculation for the Condensate/Booster Min Flow Valve to verify margin at the new design pressure and temperature. Verify adequate flow capability at the EPU operating conditions with all pumps running.	Lintakas	Yost	08/30/05		216163-27	08/30/05		216163-27	08/30/05		240081-27	08/30/05		240081-27	0
12	Evaluate the temperature element thermowells in the condensate, condensate booster and feedwater systems to eliminate concern with erosion and vortex shedding failure vulnerability. Also address the H2 and O2 Injection Quills.	Galanis	Porter	06/01/05		216163-28	06/01/05		216163-28	06/01/05		240081-28	06/01/05		240081-28	0
13	Perform sizing calculation for the HP and LP Heater Inlet, Outlet and Bypass MOVs to verify margin at the new design pressure and temperatures.	Lintakas	Mills	08/30/05		216163-43	08/30/05		216163-43	08/30/05		240081-43	08/30/05		240081-43	0
14	(Dresden Only) Evaluate need to reduce pressure drop and alarm set point across the Demin resins during the bed cleaning operation.	Steckhan	N/A	06/01/05		216163-45	06/01/05		216163-45	NA		NA	NA		NA	0
15	Resolve the overpressure condition on LP Heaters and the Drain Coolers, consistent with system design and relief valve setpoint conditions	Loch	Porter	08/30/05		216163-53	08/30/05		216163-53	08/30/05		240081-53	08/30/05		240081-53	0
16	Replace Feedwater suction relief valve with set pressures of 465 psig (min.) if the system design pressure is increased to 465 psig or above (i.e., contingent on resolution of the low pressure feedwater heaters over pressure condition)	Loch	Porter	08/30/05		216163-80	08/30/05		216163-80	08/30/05		240081-80	08/30/05		240081-80	0
17	Evaluate apparent discrepancy between Dresden and Quad Cities' set pressure for Condensate Booster Pump suction relief valves 1(2)(3)-3301A, -3301B, -3301C and –3301D should be evaluated and reconciled appropriately	Loch	Porter	08/30/05		216163-57	08/30/05		216163-57	08/30/05		240081-57	08/30/05		240081-57	0
18	Perform detailed orifice sizing calculations for the low pressure B and C feedwater heater operating vents.	Bezouska	Rolfes	08/30/05		216163-79	08/30/05		216163-79	08/30/05		240081-79	08/30/05		240081-79	0

EPU EXTENT OF CONDITION REVIEW RECOMMENDATIONS
(ANALYSES, STUDIES, PM REVIEWS)

#	Recommendation	Owners		Dresden						Quad						Total Cost
		DRE	QDC	Unit 3			Unit 2			Unit1			Unit 2			
				Date	Cost	Action Tracking	Date	Cost	Action Tracking	Date	Cost	Action Tracking	Date	Cost	Action Tracking	
19	Determine appropriate system pressures and low pressure (LP) feedwater heater thermal relief valve set points such that (1) the thermal relief valves do not open under expected operating conditions, and (2) the thermal relief valve set points are such that the tube design pressure of the LP feedwater heaters is not exceeded.	Bezouska	Rolfes	06/01/05		216163-54	06/01/05		216163-54	06/01/05		240081-54	06/01/05		240081-54	0
20	Ensure that actuator PMs for critical AOVs in the FW and Condensate System are based on temperatures consistent with the expected Condensate System area ambient temperature		Yost/ Rolfes	08/30/05		240191-15	08/30/05		240191-15	08/30/05		240194-09	08/30/05		240194-09	0
21	Identify the population and Replace Nitrile (Buna-N) elastomers in susceptible valves such as backwash "W" valves in the Quad Cities demineralizer system.	Lintakas	Yost	08/30/05		216163-55	08/30/05		216163-55	08/30/05		240081-55	08/30/05		240081-55	0
22	Specifications K4080 and R4411 should be revised to provide a new piping design temperature of approximately 140 deg-F and design pressure of approximately 185 psig for the Condensate System piping between themain condenser and the Condensate Booster Pump suction.	Loch	Porter	06/01/05		216163-56	06/01/05		216163-56	06/01/05		240081-56	06/01/05		240081-56	0
23	Develop an analytical model for the FWLC system to better predict the Vessel Levels post transients and scrams (Dresden only)	Bezouska	Santic	12/12/04		216163-58	12/12/04		216163-58	NA		NA	NA		NA	0
24	Evaluate, tune and test FWLCS for desired off-rated/pump configurations (e.g. 2 pump operation). Incorporate off-normal conditions in Analytical Model	Bezouska	Santic	08/30/05		216163-59	08/30/05		216163-59	Complete		240081-59	Complete		240081-59	0
25	Re-evaluate T0801 recommendation to operate with full Offgas condenser condensate flow (Procedure Change)	Bonomo	Myers	08/30/05		216163-61	08/30/05		216163-61	08/30/05		240081-61	08/30/05		240081-61	0
26	Isolate the Preheat steam supply during full power operation	Bonomo	Myers	08/30/05		216163-63	08/30/05		216163-63	08/30/05		240081-63	08/30/05		240081-63	0

EPU EXTENT OF CONDITION REVIEW RECOMMENDATIONS
(ANALYSES, STUDIES, PM REVIEWS)

#	Recommendation	Owners		Dresden						Quad						Total Cost
		DRE	QDC	Unit 3			Unit 2			Unit1			Unit 2			
				Date	Cost	Action Tracking	Date	Cost	Action Tracking	Date	Cost	Action Tracking	Date	Cost	Action Tracking	
27	Obtain and trend temperature data for areas of the turbine and reactor building that contain equipment affected by increased temperatures under EPU conditions. Compare the data to projected values by the EPU task report	McGallian	Luebbe	12/12/06		216163-73	12/12/06		216163-73	04/28/07		240081-73	04/28/07		240081-73	0
28	**(Quad Cities Only)** Implement the industry accepted method for determining drywell bulk average temperature.	N/A	Luebbe	NA		NA	NA		NA	12/15/04		216402-57	12/15/04		216402-57	0
29	**(Quad Cities Only)** Drywell Duravent System represents a single failure vulnerability. Either increase PM or redundancy.	N/A	Luebbe	NA		NA	NA		NA	12/15/04		216402-53	12/15/04		216402-53	0
30	Determine the need for de-superheat flow requirements into the moisture separator drains (Dresden & Quad Cities). If required, reactivate the system at Dresden.	Bezouska	Rolfes	06/01/05		216163-74	06/01/05		216163-74	06/01/05		240081-74	06/01/05		240081-74	0
31	Correct inconsistencies in the maintenance/modification specification documentation for the Feedwater, Condensate and heater drain system pressures and temperatures.	Galanis	Porter	12/15/05		216163-85	12/15/05		216163-85	12/15/05		240081-85	12/15/05		240081-85	0
32	Correct inconsistencies in the controlled passport data instrument parameters' field for the heater drain system pressures and temperatures. Review setpoint calculations and control functionality, as required by the parameter changes.	Bezouska	Rolfes	08/30/05		216163-86	08/30/05		216163-86	08/30/05		240081-86	08/30/05		240081-86	0
33	**(Quad Cities Only)** update passport to reflect component classification categories and implement PMs per the PCM program templates for Heater Drain System. Align D/Q PCM programs.	N/A	Rolfes	NA		NA	NA		NA	08/30/05		216402-20	08/30/05		216402-20	0
34	Monitor cross-around relief valves for leakage after any pressure transient within the turbine boundary.	Bezouska	Foster	06/01/05		216163-89	06/01/05		216163-89	06/01/05		240081-89	06/01/05		240081-89	0
35	Implement PMs based on industry experience related to increased EHC accumulator seal leakage and loss of accumulator charge after EPU implementation.	Chiou	Mendenhall	06/01/05		216163-90	06/01/05		216163-90	06/01/05		240081-90	06/01/05		240081-90	0

EPU EXTENT OF CONDITION REVIEW RECOMMENDATIONS
(ANALYSES, STUDIES, PM REVIEWS)

#	Recommendation	Owners		Dresden						Quad						Total Cost
		DRE	QDC	Unit 3			Unit 2			Unit1			Unit 2			
				Date	Cost	Action Tracking	Date	Cost	Action Tracking	Date	Cost	Action Tracking	Date	Cost	Action Tracking	
36	Revise the appropriate procedures to perform Main Condenser tube cleaning and waterbox de-sludging if monitoring parameters indicate the presence of scale or debris.	McAllister	Peters	03/30/05		216163-94	03/30/05		216163-94	03/30/05		240081-94	03/30/05		240081-94	0
37	**(Dresden Only)** Assess the effectiveness of the modified chemical treatment plan to prevent Main Condenser scaling	McAllister	N/A	06/01/05		216163-95	06/01/05		216163-95	NA		NA	NA		NA	0
38	**(Dresden Only)** Implement lessons-learned from the Braidwood lake chemistry problems which caused tube fouling due to calcium carbonate	Austin	N/A	06/01/05		216163-96	06/01/05		216163-96	NA		NA	NA		NA	0
39	Review EPU impacts on Operational Flexibility Options in place at Dresden and Quad.	Flick	Boline	08/30/05		216163-13	08/30/05		216163-13	08/30/05		240081-13	08/30/05		240081-13	0
40	Develop an analysis to address the Torus water heat up rate during a postulated small bore line brake for HPCI operation.	Disalvo	Freidrichsen	12/12/04		240191-02	12/12/04		240191-02	12/12/04		240194-01	12/12/04		240194-01	0
41	Determin if the air wash system is credited for post-EPU ventilation.	McGallian	N/A	08/30/05		240191-03	08/30/05		240191-03	NA		NA	NA		NA	0
42	Collect data for the last two transients at Dresden Unit 3 and analyze the data as committed to the NRC as part of EPU	Alguire	Loch	06/01/05		240191-04	06/01/05		240191-04	06/01/05		240194-02	06/01/05		240194-02	0
43	Identify systems or analyses with limited margin post-EPU and evaluate/implement actions to increase margin as appropriate.	Eldridge	Eldridge	12/15/04		240191-12	12/15/04		240191-12	12/15/04		240194-05	12/15/04		240194-05	0
44	**(Quad Cities Only)** Reconstitute Design Basis Calculcation for Main Steam Line. The line is qualified but 4 support qualification calculations were lost during EPU.	N/A	Porter	NA		NA	NA		240191-02	08/30/05		240194-03	08/30/05		240194-03	0

EPU EXTENT OF CONDITION REVIEW RECOMMENDATIONS
(ANALYSES, STUDIES, PM REVIEWS)

#	Recommendation	Owners		Dresden						Quad						Total Cost
		DRE	QDC	Unit 3			Unit 2			Unit1			Unit 2			
				Date	Cost	Action Tracking	Date	Cost	Action Tracking	Date	Cost	Action Tracking	Date	Cost	Action Tracking	
	Total Cost				0			0			0			0		0
	20% for Third Party Review				0			0			0			0		0
	Total Cost, including the Third Party Review				0			0			0			0		0

ATTACHMENT 4

Target Rock Safety/Relief Valve – Root Cause Report and Operability Evaluation for Quad Cities Nuclear Power Station

4A) Quad Cities Nuclear Power Station, Unit 2 Root Cause Analysis, "Target Rock As-Found Lift Pressure High"

4B) Quad Cities Nuclear Power Station, Units 1 and 2, Operability Determination 220863-08, Revision 5, "1(2)-0203-3A Target Rock Safety Relief Valve"

ATTACHMENT 4A

Quad Cities Nuclear Power Station, Unit 2
Root Cause Analysis
"Target Rock As-Found Lift Pressure High"

NOTES

1. Extension of this assignment requires MRC approval.
2. Remove the *italicized* text below upon completion.
3. During the performance and upon completion of the investigation, investigators should reevaluate Operability, Reportability, Extent of Condition Reviews, and the need to generate a Operating Experience for the issue. Investigations can reveal new aspects of an event that may change the original disposition of an issue. For example, additional investigation such as power labs testing of a component may change the original evaluations of Operability.

Root Cause Analysis

1. **Title:** Target Rock As-Found Lift Pressure High

2. **Unit(s):** Quad Cities Station, Unit 2

3. **Event Date:** 4/19/04
   Event Time: 1500

4. **Action Tracking Item Number:** 215874-20
   Report Date: 6/9/04

5. **Investigators:**

Team Lead	Jim Trettin (RCE)
Root Cause Evaluator	Chris VanDenburgh (RCE) / Ryan Merema
Technical support	Rick Swart
	Wyle Laboratories
	Target Rock Field Services

6. **Executive Summary:**
   A Target Rock Safety Relief Valve (S/RV) Model 7467F, Serial Number 172, removed from EPN location 2-0203-3A in Q2R17, was as-found tested at Wyle Laboratories Nuclear Services S/RV Facility in Huntsville, AL, on 4/19/04. This testing identified that the Target Rock S/RV had an as-found pressure setpoint that exceeded Technical Specification (+/- 1%) and ASME Code requirements (+/- 3%). The result of the testing was an as-found lift pressure of 1213 psig against a nameplate setpoint of 1135 psig. This results in a lift pressure of 6.8% higher than nameplate.

   Arrangements were made to have Quad Cities and Exelon corporate personnel present at Wyle Laboratories to supervise and document Target Rock S/RV disassembly and inspection, which took place on 5/11/2004 and 5/12/2004. During Exelon overview of the vendor disassembly of the pilot stages of the valve, subcomponent degradation (0.008" groove in Bellows Cap Assembly) was identified (Attachments 9 and 10). Corrosion was also identified in the carbon steel components of the first stage section of the valve. The as-found testing does not support corrosion playing a factor in the pressure as-found setpoint because it would have been cleared in subsequent lifts, result in larger changes in pressure on subsequent lifts when compared to the first lift, which was not seen in the test data.

EPU related vibrations are considered the cause of the wear in the bellows cap. Shaker Table testing will help to determine the particular magnitudes and axes of vibrations that lead to this type of wear. It is clear that vibration magnitudes on the main steam lines have increased associated with operation at EPU power levels. The wear may be a result of large magnitude vibrations acting directly on the springs, or may be a result of vibration magnitude increases at critical frequencies that lead to spring oscillations or excitation.

The Root Cause of the Target Rock S/RV degraded condition is EPU-related vibration that caused material to be removed from the Bellows Cap Assembly, resulting in extra force (pressure) required to actuate the safety portion of the S/RV.

The extent of condition was addressed in an Operability Evaluation, ATI 220863-08. The Operability Evaluation identified that the 1st stage set pressure set point (Safety Function) of the Unit 1(2) Target Rock S/RV 1(2)-0203-3A may be higher than expected due to vibration induced wear between the Bellows Assembly Cap and the Pressure Adjustment Spring. The degraded condition appears to be EPU-related Power Level vibration related wear between the first stage Pressure Adjustment Spring and Bellows Assembly Cap (see Attachments 9 and 10 for pictures).

- The Quad Unit 2 valve (serial #172) was the first S/RV to experience wear of this magnitude. EPU power operation has resulted in an increase in vibration magnitudes on Main Steam Lines. Valves from previous pre-EPU Quad Unit 1 and 2 cycles have not experienced high-lift associated with this specific subcomponent degradation. Additional valves from Quad as well as 2 other Exelon sites were inspected, and indicate signs of typical contact between the spring and cap that did not result in the material loss seen at Quad Unit 2.

Based on this Operability Evaluation for both Units 1 and 2, the Operations Standing Orders were updated to limit power to 2511MWth. Actions were also initiated to implement and test S/RV subcomponent material improvements.

Corrective actions to prevent recurrence (CAPRs) for this event include disposition of S/RV vibration induced subcomponent degradation caused by operation at EPU power levels. The solution may include, but is not limited to, the following:

- Valve subcomponent modification in the form of material selection or spring guidance that either eliminates any identified component excitation, or provides wear characteristics to withstand any identified vibration related component contact
- Main Steam System modifications on Unit 2 and possibly Unit 1 as needed to eliminate EPU-related vibrations that result in S/RV degradation

Previous Quad Cities Target Rock S/RV events were attributed to setpoint drift:

- PIF 97-2204/CR Q1997-02204 – U-1 Target Rock S/RV as-found setpoint was 1095 psig, which was 3.5% below nameplate rating (SN 225, 4/25/1997).
- PIF Q1999-01389 – U-1 Target Rock S/RV as-found setpoint was 1174 psi, which was 3.4% above the 1135 nameplate rating (SN 120, 4/16/1999).

In both cases, it was determined that there was no violation of overpressure analyses. A Root Cause determination was not performed in the instances noted above. A Root Cause was initiated for the latest as-found setpoint because 6.8% deviation from nameplate is an outlier

when compared to industry peers. An OPEX / EPIX search determined that this is the first instance of EPU-related vibration that caused S/RV subcomponent degradation, causing high as-found lift pressures on the Target Rock S/RV.

Due to the existence of similar discrepancies in multiple valves, (Dresser Rand Safety Valve and Target Rock S/RV), NUREG-1022 requires an evaluation of a requirement to submit a Licensee Event Report (LER), reference AT 215874-10.

7. **Condition Statement:**
Target Rock S/RV Model 7467F, Serial Number 172, removed from EPN location 2-0203-3A in Q2R17, had an as-found pressure setpoint that exceeded Technical Specification 3.4.3 limit (+/- 1%) and ASME Code requirements (+/- 3%). The result of the testing was an as-found lift pressure of 1213 psig against a nameplate setpoint of 1135 psig. This results in a lift pressure of 6.8% higher than nameplate.

8. **Event Description:**

Event
Target Rock S/RV Model 7467F, Serial Number 172, removed from EPN location 2-0203-3A in Q2R17, was as-found tested at Wyle Laboratories Nuclear Services S/RV Facility in Huntsville, AL, on 4/19/2004. This testing identified that the Target Rock S/RV had an as-found pressure setpoint that exceeded Technical Specification (+/- 1%) and ASME Code requirements (+/- 3%). The result of the testing was an as-found lift pressure of 1213 psig against a nameplate setpoint of 1135 psig. This results in a lift pressure of 6.8% higher than nameplate. The Main Steam System Engineer initiated CR 215874 on 4/20/04 after being informed of the Target Rock S/RV test results.

Target Rock S/RV Service History
A chronological timeline of the recent service history of the Target Rock S/RV, serial number 172, is included below (summarized in Event and Causal Factor chart, see Attachment 1):

1. 2/1999 (Q2P02), Target Rock S/RV serial #172, removed from U-2, WO 97099043.
2. 4/1999, S/RV as-found tested at Wyle Laboratories 1.6% below nameplate of 1135 PSIG, Wyle Report 42774-2.
3. 6/1999 to 9/2001, Target Rock S/RV refurbished by Target Rock Field Service at Wyle Laboratories Nuclear Services S/RV Facility. The scope of work was standard and included pilot lapping.
4. 1/2001, Target Rock S/RV certified at Wyle Laboratories with lift setpoint of 1129 PSIG, 0.5% below nameplate of 1135 PSIG.
5. 2/2002, Target Rock S/RV serial #172 installed in Q2R16, WO 324872.
6. 2/2002, EPU modification installed during Q2R16.
7. 3/2002, Target Rock S/RV stroke tested under QCOS 0203-03, reference WO 324872-02.
8. 7/2002, S/RV removed to support FME retrieval during Q2M20, WO 463580.
9. 7/2002, Target Rock S/RV stroke tested under QCOS 0203-03, reference WO 463580-03.
10. 5/2003, CR 158148 documents bellows line to pressure switch broken.
11. 3/2004, valve removed during Q2R17, reference WO 478605.

12. 4/2004, Target Rock S/RV #172 as-found tested at Wyle Laboratories Nuclear Services S/RV Facility Huntsville, AL. Wyle Test Procedure number 1100 has been approved by Quad Cities and Wyle to accomplish this testing (reference Attachment 4 for Target Rock S/RV Testing Methodology) under job number 48744. The complete test included the following lifts, which were high and repeatable:

    a. 1st lift: 405 msec delay time, 1213 psi (nameplate of 1135 PSIG, as found 6.8% high)
    b. 2nd lift: 280 msec delay, 1218 psi
    c. 3rd lift: 275 msec delay, 1221 psi
    d. 4th lift with solenoid, 215 msec delay, successful lift

    Note that the first three lifts are tests of the safety function of the valve, and rely on steam pressure only to cause the valve to cycle. The final lift is a test of the relief function of the valve, which bypasses the first stage safety function, and directly acts on the second stage with an air operator.

CR 215874 was generated to document the discrepancy in Target Rock S/RV as-found setpoint on 4/20/2004.

Arrangements were made to have Quad Cities and Exelon corporate personnel present at Wyle Laboratories to supervise and document Target Rock S/RV disassembly and inspection, which took place on 5/11/2004 and 5/12/2004.

Target Rock S/RV Disassembly and Troubleshooting Findings

Reference Attachment 5 for details regarding design and operation of the Target Rock 3-stage S/RV if questions arise while reviewing this section.

Quad Cities and Exelon corporate personnel supervised and documented Target Rock S/RV disassembly and inspection, which took place between 5/11/2004 and 5/12/2004. A Senior Target Rock Field Service Manager and Valve Design Engineer were also present. The trip report in its entirety is included for reference in Attachment 3, with highlights summarized below.

The bonnet cap was removed to allow access to the 1st stage pilot assembly. Some general corrosion was present in the areas visible at this point. It was also noted that the Cap assembly was not centered in the adjusting ring, but was tilted slightly toward the direction of the air operator, reference Attachment 6. This tilt is typical of the valve assembly based on inspections of three other disassembled valves at the Wyle Laboratory facility for as-found testing and overhaul.

The Adjusting Nut was removed, allowing the Cap and Spring to be removed from the bonnet. Upon removal, there was considerable corrosion, including loose corrosion, at the base of the cap, on the spring, and in the base of the pilot section, reference Attachments 7 and 8. The valve body and the bonnet housing are constructed of carbon steel, while the pressure boundary components are constructed of stainless steel and other materials. The vendor indicated that this section experiences some corrosion, but the level of corrosion found on this valve was more extensive than typically seen.

- Consistent pressures were noted during all three test runs, with only the delay time changing between the first run and the subsequent two runs. The corrosion noted outside of the bellows in the first stage bonnet potentially caused increased friction during the first test run, during which the corrosion products were cleared from the areas of contact; however, the first run is also impacted by the collection of condensate that must be flushed through the pilot stages of the valve. On the second and third run, the delay times were both normal, and consistent. Had corrosion played a role in the setpoint drift, it would be expected that the three test runs would have displayed larger variance in lift setpoints, most likely in a downward direction, as corrosion was cleared from the contact surfaces. As this was not the case, the corrosion is not considered a contributing cause to the high pressure setpoint; the valve manufacturer concurs with this position.

The Bellows Cap section of the valve is not pressurized unless the bellows has ruptured or has a leak. Bellows leakage was eliminated as the source of corrosion in the Bellows Cap area by pressurizing and snooping. No leakage was present. The Bellows failure pressure switch is connected to a port on this chamber at Quad Cities, but at other operating BWRs, the port is open to atmosphere. The source of the corrosion is unknown at this time.

- ATI to A8430TP, 8/18/04 to investigate and disposition corrosion found in Target Rock S/RV Bellows Cap area.

Following cleaning, there were clear indications of contact between the Cap and Spring which are tensioned under the Adjusting Nut for final pressure set point adjustment. There were two wear areas noted, with one caused by the uppermost partial coil (which is ground to provide a parallel seating surface) and one caused by the first full Coil, reference Attachments 9, 10, 11. The lower of the two, caused by the first full Coil, had a depth of 0.008 inches. A small contact point was also noted on the interior of the Cap, where it was riding at the base of the Bellows; there was no apparent depth to the interior wear mark.

Inspections were made of other 3-stage SRV's, including another Quad Cities valve removed during Q1R17, a Limerick valve, and a Peach Bottom valve. All three inspections indicated the presence of contact in this same area between the Cap and Spring, but only Quad Cities Target Rock serial #172 showed any depth in the area of the wear marks.

In summary, the current valve design allows contact between the Bellows Cap Assembly and the Pressure Adjustment Spring. Operation at EPU power levels causes vibration in main steam line components. Vibration-induced movement between the above components resulted in a 0.008" groove in the Bellows Cap Assembly. The Pressure Adjustment Spring is captured in this groove and requires additional force, provided by system pressure (which causes the high as-found setpoint) to cause the Pressure Adjustment Spring to be forced out of the groove. This concept is expanded in Attachment 12 for clarity. The corrosion is discounted as a causal factor in this investigation due to consistent S/RV lift pressures required to initiate the safety function of this valve.

9. **Evaluation:**

Analysis Methodology

Recommendations provided in LS-AA-125-1001, Attachment 16 were reviewed to determine the Root Cause Tools appropriate for this investigation. An Event and Causal Factor Chart, Attachment 1, was utilized to document the timeline of events associated with the degraded condition on the Target Rock S/RV removed from EPN location 2-0203-3A, as well as operating history of this S/RV. The TapRoot software package was utilized to analyze causal factors and delineate the Root Cause. Finally, based on the investigation of an equipment failure, a Failure Modes and Effect Analysis, Attachment 2, was used to analyze findings from the QCGS System Engineering Trip Report for Valve Disassembly and Troubleshooting at Wyle Laboratories, Attachment 3. The Equipment Apparent Cause Evaluation Guide, Attachment 5 of LS-AA-125-1003, is included in Attachment 17 to ensure the RCI thoroughly evaluates all possible causes of this equipment failure.

Analysis and Root Cause Determination

There were indications of contact between the Bellows Cap Assembly and Adjustment Spring that are tensioned under the Adjusting Nut for final pressure set point adjustment. The wear area caused by the first full Coil, reference Attachments 9 and 10, would vibrate and create a groove on the Cap. EPU vibrations are being considered the source for the degradation based on the fact that this failure has not been experienced in the past, does not appear to be experienced on a sample of similar valves and a valve from pre-EPU Quad operation, and there is a known increase in vibration magnitudes at EPU operating levels.

The exact vibration amplitude and orientation to produce this specific wear pattern cannot be specified until shaker table testing is complete. Wear may be a result of direct vibration force on the spring resulting in some motion, or may involve a specific frequency causing oscillations or excitation in the spring and cap assembly.

Additional force, provided by system pressure (causing the high as-found lift setpoint) would be required to cause the spring to be forced out of the groove created by EPU-related vibration. This concept is expanded in Attachment 12 for clarity. Based on this thought process, the Root Cause of this event is identified below:

The Root Cause of the Target Rock S/RV degraded condition is EPU-related vibration that caused material to be removed from the Bellows Cap Assembly, resulting in extra force (pressure) required to actuate the safety portion of the S/RV.

Potential Causes Considered and Dispositioned

Discussions were held with Exelon Corporate Relief Valve subject matter experts, and with the Target Rock Field Service Manager, to determine possible causes for a high setpoint in this style of valve. The list below includes potential causes for high lift, and in some cases, provides immediate disposition for these causes based on the as-found steam testing. The System Engineering Trip Report for Valve Disassembly and Troubleshooting at Wyle Laboratories (Attachment 3) includes details on the remaining potential causes.

1. Pilot Seat Leakage

a. Pilot leakage not present during valve operation nor at beginning of as-found steam testing, so not likely. Disassembly showed no signs of leakage or cutting (see disassembly text, Attachment 3).

2. Pilot Stage Leakage (Bellows, Bellows O-ring, other)
   a. Bellows failure alarm not actuated during operation. No steam leakage seen during steam testing. Disassembly showed no signs of leakage during pressurized snooping and flooding (see disassembly text, Attachment 3).
3. Physical Rub or Binding at Pilot Disk to Pilot Stem
   a. See disassembly text (none found).
4. Binding of Pilot Disk to Seat
   a. This is typical of larger diameter spring over seat safety valves in PWRs, but not typical of pilot seats at BWRs.
   b. This would appear as a high lift during the as-found test, followed by normal pressure lifts on subsequent runs. All three runs on this S/RV had high lifts.
5. Loose connection in pre-load spacer to stem
   a. See disassembly text, Attachment 3 (all pins, lock wires, etc, present and tight).
6. Abutment Gap Change
   a. See disassembly text, Attachment 3 (gap acceptable as-found).
7. Unbalance of Pilot Section
   a. This addresses a hydraulic force unbalance, and is really covered by the pilot disk and bellows leakage already discussed earlier.
8. Hold Down Nut (Pre-load assembly) Movement
   a. This would typical cause a decrease in pressure, if it were loose. Found tight with lock wire in place (see disassembly text, Attachment 3).
9. Adjusting Nut Movement
   a. If the anti-rotation bar failed, it is expected that this would result in a decrease in pressure, similar to the pre-load nut. The setpoint error caused by a very slight turn of the adjusting nut would only be a few psi. The amount of as-found setpoint error would require a misadjustment of over one full turn, so this is not likely. No issues found, and high level of confidence in test personnel and training per test vendor interviews. See disassembly text.
10. FME
    a. Severe clogging of the pilot stage filter, or severe clogging in other flow areas of the valve could cause the high lift setpoint. No situations similar to this were found (see disassembly text, Attachment 3).
    b. Corrosion found in the bonnet area, outside of the pressure and flow areas, could have impact on delay time associated with first lift. Could have caused binding that would have contributed to one high lift, but then subsequent lifts would have been acceptable, so the corrosion is discounted as a cause for the high lift. The corrosion is discounted as a causal factor in this investigation due to consistent S/RV lift pressures required to initiate the safety function of this valve.
11. Component Damage
    a. No specific item was targeted prior to disassembly, just a broad based approach.
    b. Groove in Bellows Cap, caused by pressure adjusting spring, was identified (see disassembly text, Attachment 3).
        i. This was identified as a typical contact point on three other Exelon valves that were disassembled in the shop at that time, although none had any depth of degradation.

ii. Vibrations, coupled with the contact, would result in such degradation.
iii. Additional force would be required to cause spring to roll up on the side of the groove during movement of the cap.

Many of the items above could be ruled out based on the steam testing results (no initial leakage, consistent high lift setpoints). Some items had a very low probability of causing a high setpoint, but would be typical of a low setpoint (locking devices fail, allowing spring retention nuts to back out). These, and remaining items, were each reviewed during pilot section disassembly and dispositioned as not causing the problem.

The conclusion that was drawn from the investigation of the topics listed above is that the high lift setpoint is a result of the specific component damage associated with the bellows assembly cap. This damage, in the form of a 0.008" groove worn into the cap by the adjacent spring, required additional steam pressure to force the spring to be pushed in a direction out of the groove before the first stage disk could lift. While there was a large amount of corrosion present associated with the carbon steel areas of the first stage bonnet, the consistent test pressures discount the corrosion from being a cause for the lift pressures; the pressures would have varied, most likely in the downward direction, as corrosion products were cleared during subsequent lifts.

The inspections did not allow for specific vibration testing, although this will be accomplished at a later date with the use of shaker table testing. Inspections of additional 3-stage valve components available at the time of the inspection showed that there is indeed contact between the cap and spring, but the contact had not resulted in detectable material loss from the other valves. It is understood that EPU operation at Quad Cities has resulted in increased main steam line vibrations, and that these vibrations have caused degradation on other pieces of equipment, and specifically, some assemblies including springs. Degradation may be a result of direct vibration magnitudes acting on the spring, or may be a result of slight vibration magnitude increases at some critical frequency causing spring oscillations or excitation. Shaker Table testing will help to identify the magnitudes and axes that contribute to wear at this location.

A. <u>CAP Trend Codes:</u>

1. The root cause trend coding is as follows:
   Equipment difficulty (1E), Design (2D), Design Specifications (3S), Problem Not Anticipated (4PN), Equipment Environment Not Considered (5EE).

B. <u>Causal Factors:</u>
The Root Cause of the Target Rock S/RV degraded condition is EPU-related vibration that caused material to be removed from the Bellows Cap Assembly, resulting in extra force (pressure) required to actuate the safety portion of the S/RV.

1. Error Precursor: EPU installed during Q2R16 with inadequate design review. Time dependent degradation due to vibrations was not considered in the standard monitoring and assessment program for EPU.

2. Root Cause: The high lift set point on S/RV Serial Number 172 is due to vibration induced wear resulting in a 0.008 inch groove in the Bellows Assembly Cap caused by contact with the Pressure Adjustment Spring. Forces required to cause the spring to ride out of this groove were higher than forces normally required to cause the first stage pilot disk to open. See Attachment 12 for simplified diagram and detailed discussion.

3. Bases: Consistent high lift pressures during as found testing, and inspection of Target Rock S/RV components during disassembly at Wyle Laboratories by Corporate and Quad Cities personnel.

4. Corrective Actions: CR 220863 was initiated based on initial inspection findings at Wyle Laboratories. An Operability Evaluation was conducted which supported continued operation with the installed Target Rock S/RVs. Compensatory actions from the Operability Evaluation include limitation of reactor power to a pre-EPU level of 2511MWth. These actions are complete, and are documented below for completeness.

   Further corrective actions from the Operability Evaluation include test and/or evaluations into the natural frequency or frequencies associated with the Target Rock S/RV, especially the Pressure Adjusting Spring and Bellows Assembly Cap. Testing on similar equipment is planned utilizing shaker table testing at a test facility. An additional corrective action is to specify, quote, and procure a Bellows Assembly Cap made of a material more resistant to wear than the existing cap. A material of Nitronic 60 has already been discussed with the Vendor. This material will be included in the shaker table testing. A final corrective action will be to submit a modification/parts replacement request as needed to utilize any upgraded parts in the Target Rock Valve replacement for Q1R18 and any future outages.

   A new CAPR is suggested to disposition vibration induced degradation of the Target Rock S/RVs based on results of the testing noted above, elimination of EPU related vibration in the steam lines.

C. Corrective Actions:

1. Immediate and Interim Corrective Actions:

   CA: AT 220863-10, A8410OP, complete. Establish controls to limit Unit 1 reactor power to 2511MWth for any significant duration. Prior to raising power on Unit 1 above 2511MWth, contact Engineering to determine any power level restrictions that may be needed, and revise this Operability Evaluation if required. Short duration power increases above 2511 MWth for data acquisition purposes is allowed (<=72hrs).

   CA: AT 220863-11, A8410OP, complete. Establish controls to limit Unit 2 reactor power to 2511MWth for any significant duration. Prior to raising

power on Unit 1 above 2511MWth, contact Engineering to determine any power level restrictions that may be needed, and revise this Operability Evaluation if required. Short duration power increases above 2511 MWth for data acquisition purposes is allowed (<=72hrs).

2. CAPRs:

CAPR: A8430TP, 5/15/06. Disposition vibration induced degradation of Target Rock S/RV based on results of valve vibration or shaker table testing. Solution may include, but is not limited to, the following:

- Valve subcomponent modification in the form of material selection or spring guidance that either eliminates any identified component excitation, or provides wear characteristics to withstand any identified vibration related component contact
- Main Steam System modifications on Unit 1 and/or Unit 2 as needed to eliminate EPU-related vibrations that result in S/RV degradation

3. EFRs:

EFR: A8430TP, 5/15/2008. Perform and Effectiveness Review of the CAPRs identified in this report IAW LS-AA-125-1004. Due date required to achieve proper as-found data after EPU run.

4. Corrective Action Assignments:

CA: AT 220863-12, A8430TP, 6/18/04, quotation complete, procurement in progress. Specify, Submit RFQ, and Procure a Bellows Assembly Cap in wear resistant material Nitronic 60 or similar for use in shaker table tests of S/RV.

CA: AT 220863-13, A8064MW-DR, 8/12/04, preliminary arrangement for testing 6/28 to 7/2. Perform shaker table testing on the S/RV, or on appropriate subcomponents of the S/RV (as required due to radiological controls). Perform any structural or vibration analyses required to support or evaluate the shaker table testing. Testing should include identification of natural frequencies and determine correlation to existing frequencies at Quad Units 1 and 2. Aging runs at appropriate frequencies should be performed to determine wear characteristics of affected components within the valve, especially those associated with the adjustment spring and bellows cap assembly.

CA: AT 220863-14, A8430TP, 8/30/04. Submit modification requests for any S/RV upgrades as determined necessary through testing and evaluation.

CA: AT 215874-05, A8430TP, complete. Arrange for Vendor support to disassemble the S/RV in order to determine possible causes for the high lift setpoint.

CA: AT 215874-10, A8401RAPR, 6/7/04. Review the issue for reportability.

CA: AT 215874-04, A8013NFMTS, 6/9/04. Nuclear Fuels to analyze the appropriate fuel cycle accidents utilizing the as-found lift setpoints of both the S/RV and the MSSV's to determine if any fuel limitations were violated.

CA: AT 215874-29, A8401RAPR, 6/21/04. Issue an LER for the high setpoints on the S/RV and MSSV.

CA: AT 215874-22, A8430TP, 6/30/04. Review the S/RV event, including the Nuclear Fuels evaluation of the setpoint impact, against the requirements of Maintenance Rule for Functional Failure determination.

CA: AT 215874-23, A8451NESPR, 7/22/04. Verify test data is entered into the IST database.

CA: AT 215874-05, A8430TP, complete. Observe initial valve disassembly to determine cause of high lift setpoint, and to interview Valve Vendor and Test Facility personnel.

CA: AT 215874-33, A8002OEALL, complete. Issue NER on this event if determined applicable.

CA: AT 215874-35, A8002OEALL, 6/24/04. Issue final NER on this event if determined applicable

CA: AT 215874-27, A8430TP, 6/9/04. Obtain PORC approval of RCI.

CA: AT220863-15, A8430TP, 3/22/05. Implement appropriate Unit 1 actions in Q1R18 to allow full EPU operation.

CA: AT220863-16, A8430TP, 3/29/06. Implement appropriate Unit 2 actions in Q2R18 to allow full EPU operation.

CA: AT220863-37, A8430TP, 6/15/04. Draft a final NER on Target Rock S/RV as-found setpoint.

New assignments that require action tracking items:

ATI to A8430TP, 8/18/04 to investigate and disposition corrosion found in Target Rock S/RV Bellows Cap area.

ATI to A8401OPEX, 12/1/04 to issue a follow up NER on this event following shaker table testing to inform fleet BWRs of any vibration limitations to be considered for EPU analysis.

ATI to A8430TP, 7/28/04 to draft an OPEX report after completion of shaker table testing.

ATI to A8401OPEX, 8/4/04 to evaluate for applicability to industry in accordance with LS-AA-115, "Operating Experience Procedure" after completion of shaker table testing.

ATI to A8450CAP, 6/23/04 to update trend codes in TIMA017 panel in Passport.

ATI to A8430TP, 9/30/04 to review the data obtained from the Dresden SRV disassembly (reference CR 223897) for any similarity between Dresden and Quad valve conditions. Initiate a new CR should any adverse conditions parallel to the separate valves be identified.

ATI to A8430TP, 8/25/04 to verify proper cap materials by obtaining vendor history on the manufacturing of this cap.

The material of the cap was not suspect, based on the similar appearance of caps from other valves, as well as the duplication of contact points from other valves. The component manufacturing history of this cap should be investigated to insure that proper materials were utilized.

10. **Extent of Condition:**

The extent of condition was addressed in an Operability Evaluation, ATI 220863-08. Excerpts from the Operability Evaluation are contained below.

The 1st stage set pressure set point (Safety Function) of the Unit 1(2) Target Rock Safety Relief Valve (S/RV) 1(2)-0203-3A may be higher than expected due to vibration induced wear between the Bellows Assembly Cap and the Pressure Adjustment Spring. This condition was identified on an S/RV removed from Quad Unit 2 at the end of Cycle 17 when the valve was tested at +6.8% of nameplate pressure, which exceeds both Tech Spec 3.4.3 (+/- 1%) and ASME Code (+/- 3%) requirements. The subcomponent degradation (0.008" groove in Bellows Cap Assembly) was identified on 5/12/04, during Exelon overview of the vendor (Target Rock) disassembly of the pilot stages of the valve. This condition was documented in CR 220863 on 5/13/04.

The degraded condition appears to be EPU Power Level vibration related wear between the first stage adjustment spring and cap:

- The Quad Unit 2 valve (serial #172) was the first S/RV to experience wear of this magnitude. EPU power operation has resulted in an increase in vibration magnitudes on Main Steam Lines (reference table below). Valves from previous pre-EPU Quad Unit 1 and 2 cycles have not experienced this high-lift associated with this specific wear situation. Additional valves from Quad as well as 2 other Exelon sites were inspected, and indicate signs of typical contact between the spring and cap that did not result in the material loss seen at Quad Unit 2.
- Vibration data has been collected on both Quad units. The Unit 2 vibrations at the inlet to the S/RV are higher than the equivalent Unit 1 vibrations, at both EPU and Pre-EPU conditions. This is discussed in detail below.

Quad Target Rock S/RV Inlet Flange Vibrations:
Vibration data has been collected at various locations in the Main Steam System, including on the inlets of the S/RVs on both Units 1 and 2. The table below provides some of the resulting data at pre-EPU approximate full power, and EPU full power, and clearly indicates that Quad Unit 2 has notably higher EPU level vibrations than Quad Unit 1. In addition, it can be seen that Quad Unit 2 vibrations at pre-EPU conditions are generally comparable to Quad Unit 1 vibrations at full EPU conditions.

Vibrations at Target Rock S/RV (direction)	U2 @ 2815 MWth EPU (Baseline) % / grms	U2 @ 820 Mwe % / grms	U1 @ 2910 MWth EPU % / grms	U1 @ 2488 MWth % / grms
PD (perpendicular to steam flow)	100.00% / 0.1916	50.96% / 0.0976	78.53% / 0.1505	40.23% / 0.0771
V (vertical direction)	100.00% / 1.2444	44.48% / 0.5535	29.17% / 0.363	26.90% / 0.3347
PLZ (Parallel to steam flow)	100.00% / 0.3830	48.35% / 0.1852	56.92% / 0.218	61.05% / 0.2338

Inspection Results from Other Target Rock S/RVs:
In addition to the S/RV discussed above, 3 additional S/RV Bellows Assembly Caps were inspected, with all samples coming from other Exelon Units. Each cap had indications showing that this is a typical contact point of an assembled valve, but none of the other caps had any depth at these contact points.

Historical Review and Vendor Interviews:
No previous S/RV refurbishments indicated significant wear or degradation to the Cap on pre-EPU valves from Quad Units 1 or 2. Target Rock Field Service personnel indicated that this is the first time that this type of Cap degradation has been identified on this general valve model.

EPU Power Levels on Quad Units:
Quad Unit 2, which had the degraded S/RV Cap, had approximately 553 days of EPU operation prior to Q2R17; the identified S/RV subcomponent degradation took place during this time. Quad Unit 1 experienced approximately 152 days of EPU operation prior to Q1M16; the S/RV removed from this outage tested within tolerance, and had no Cap degradation noted.

Since Q1M16, Quad Unit 1 has experienced approximately 152 days of EPU operation. Since Q2R17, Quad Unit 2 has not experienced any extended periods of EPU power operation; there was a short power increase for data gathering purposes for a duration of approximately one day.

The S/RV currently installed in Quad Unit 1 is not expected to have Bellows Assembly Cap degradation that would cause an increase in its Safety Function pressure setpoint. This is based on the above discussions, including:

- No Cap degradation noted at pre-EPU power levels, on inspections of similar valves from other Exelon reactors, or from general discussions with Vendor technicians.

- No Cap degradation noted, nor setpoint drift failure, on Unit 1 S/RV removed following approximately 152 days of EPU operation.
- Unit 1 S/RV Inlet vibrations lower than Unit 2 S/RV Inlet vibrations. Furthermore, Unit 1 vibration levels at EPU power are generally comparable to Unit 2 vibration levels at pre-EPU power.

Engineering Judgment, supported by the available data detailed above, supports continued operation of the Unit 1 S/RV at pre-EPU power levels, with no impact on the Safety Function setpoint. Operation above pre-EPU power level for extended periods will not be supported until further vibration analysis and testing of an S/RV is completed.

The S/RV currently installed in Quad Unit 2 is not expected to be degraded because the Unit has not experienced any extended EPU operation. No Cap degradation has been noted at pre-EPU power levels. Historical performance supports continued operation of the Unit 2 S/RV at pre-EPU power levels, with no impact on the Safety Function setpoint. Operation above pre-EPU power level for extended periods will not be supported until further vibration analysis and testing of an S/RV is completed.

The EPU-induced vibration is a generic cause for numerous equipment problems at Quad Cities. A Corporate Investgative Team has been assembled to address EPU-related vibration issues at Quad Cities and Dresden.

11. Risk Assessment:

The plant significant risks in this event are violations of analyses concerning reactor vessel overpressure events for Quad Cities 2 Cycle 17, ASME and ATWS Overpressure events, and impacts to the Appendix R limiting events. The ASME and ATWS Overpressurization events are discussed below with references to UFSAR and Technical Specification source documents. This discussion is then followed by the analysis conducted by Nuclear Fuels Management that states: ***"The acceptance criteria for both the ATWS and ASME overpressure analyses have been met. In addition, the applicability of the established Cycle 17 fuel thermal limits has been confirmed."*** Finally, the impact to Appendix R limiting events is discussed.

<u>ASME and ATWS Overpressurization Events</u>

<u>Technical Specification Bases section 3.4.3 states the basis for the Safety Valves:</u>

"The ASME Boiler and Pressure Vessel Code requires the reactor pressure vessel be protected from overpressure during upset conditions by self-actuated safety valves. As part of the nuclear pressure relief system, the size and number of safety valves are selected such that peak pressure in the nuclear system will not exceed the ASME Code limits for the reactor coolant pressure boundary (RCPB). Each unit is designed with nine safety valves, one of which also functions in the relief mode. This valve is a dual function Target Rock safety/relief valve (S/RV)."

<u>Technical Specification Bases section 3.4.3 states the significance of Safety Valves:</u>

Applicable Safety Analyses section: "the design safety valve capacity is capable of maintaining reactor pressure below the ASME Code limit of 110% of vessel design pressure (110% x 1250 PSIG = 1375 PSIG)".

LCO section: "Operation with fewer valves OPERABLE than specified, or with setpoints outside the ASME limits, could result in a more severe reactor response to a transient than predicted, possibly resulting in the ASME Code limit on reactor pressure being exceeded".

Applicability section: "In Modes 1, 2, and 3, all safety and relief valves must be OPERABLE, since considerable energy may be in the reactor core and the limiting design basis transients are assumed to occur in these MODES".

UFSAR section 5.2.2.2.3 describes Safety Valve flow capacity and describes resulting transients that could result from Safety Valve setpoints outside of Technical Specification 3.4.3 tolerance:
The safety valves steam flow capacity is determined by assuming that the reactor is at 2957 Mwt when a MSIV closure occurs, the relief valves fail to open, direct reactor scram (based on MSIV position switches) fails, and the backup scram due to high neutron flux shuts down the reactor. This transient is reanalyzed periodically as part of each reload license analysis. Pressure increases, following this reactor isolation, until limited by the opening of the safety valves. The peak allowable pressure is 1375 psig
(according to ASME Section III equal to 110% of the vessel design pressure 1250 psig). The safety valves setpoints are spread in 10 psi increments between 1240 and 1260 psig. This satisfies the ASME Code specifications that the lowest safety valve be set at or below vessel design pressure, and the highest safety valve be set to open at or below 105% of vessel design pressure. [5.2-17]

The total safety valve capacity is equal to approximately 43% of turbine design flow.

Typical resulting transients at 2957 MWt are shown in Reference 5 and in Figure 5.2-3. The rapid pressurization caused by the isolation (about 100 psi/s) reduces the void content of the core and produces a sharp neutron flux spike before scram shuts down the reactor. Peak fuel surface heat flux is significantly slower, reaching a peak of 129% at about 3 seconds. Vessel dome pressure reaches about 1336 psig with the peak at the bottom of the vessel near 1358 psig. Therefore, the 43% capacity safety
valves provide adequate margin below the peak allowable vessel pressure of 1375 psig.

Overpressurization protection analysis is performed using the NRC approved transient code(s) each cycle. Results for these analyses are given in the Supplemental Reload Licensing Submittals.

In addition to the ASME code overpressure analysis, Anticipated Transients Without Scram can also cause an overpressure condition in the reactor pressure vessel. The ATWS scenario is described in UFSAR section 15.8:

"This section covers the events, which result in an anticipated transient without scram (ATWS). Anticipated transient without scram events are beyond design basis accidents. Anticipated transients without scram are those low probability events in which an anticipated transient occurs and is not followed by an automatic reactor shutdown (scram) when required. The failure of the reactor to scram quickly during these transients can lead to unacceptable reactor coolant system pressures and to

fuel damage. Mitigation of the lack of scram must involve insertion of negative reactivity into the reactor, thereby terminating the long-term aspects of the event.

The occurrence of a common-mode failure, which completely disables the reactor scram function, is a very low probability event. Therefore, no significant risk to public safety is presented by the combination of an infrequent event and a common-mode failure, which prevents scram. Thus, attention is focused on those transient situations, which have a relatively high-expected frequency of occurrence at a power condition at which serious plant disturbance might result.

GE has performed a plant-unique ATWS analysis using Plant Design Licensing Basis (PDLB) approach for the Dresden and Quad Cities units. The PDLB approach establishes the analysis bases applicable to the four Dresden and Quad Cities units. The analysis was performed at the Quad Cities original licensed reactor power level (2511 MWt) and Extended Power Uprate (EPU at 2957 MWt). The original Quad Cities licensed power was chosen to provide the largest change in power and the maximum effect of EPU to the ATWS compliance criteria. The GE analysis shows that pressure regulator open to maximum demand (PRFO) [Note: Pressure Regulator Fail Open – essentially an Oversteam demand event that is terminated by MSL Low Pressure Group I isolation.] is the limiting event. The results confirm that the analysis meets the acceptance criteria of peak vessel pressure, peak clad temperature, peak clad oxidation, peak suppression pool temperature and peak containment pressure for GE14, ATRIUM-9B, and GE9/0 fuel types at 2957 MWt."

Although ATWS is not a design basis event, it is part of the Quad Cities Licensing Basis and thus is included in the discussion.

Based on the above discussion, the impact of the Target Rock S/RV as-found lift pressure was evaluated by Nuclear Fuels Management (NFM) to determine how this shift in safety valve setpoint impact the ASME and ATWS overpressurization events and nuclear fuel thermal limits.

NFM Analysis of Impact of as-found Target Rock S/RV Setpoint

Nuclear Fuels Management (NFM) evaluated the impact of the as-found Target Rock S/RV setpoints on the Quad Cities 2 Cycle 17 Licensing Analysis. Excerpts from that document are included below (see Attachment 18 for GE-NE-0000-0028-6556 in its entirety).

Background

The safety valve lift setpoint tolerance used in the Cycle 17 OPL-3 is 1%. During post-Cycle 17 operational testing, two (1-SSV and 1-S/RV) of the five MSSVs tested were found to exceed their reload licensing analysis (RLA) opening pressure (lift) setpoints. As such, re-performance of impacted analyses has been requested using bounding values to support operability determinations.

NFM Evaluation

The ATWS, ASME overpressure and fuel thermal analyses applicable to Quad Cities 2 Cycle 17 were re-evaluated using revised, bounding MSSV opening setpoints. The revised setpoints were based on 3% drift for the out-of-tolerance SSV and 7% drift for the S/RV. The RLA opening setpoint values bound the other tested SSVs and are

assumed to bound the un-tested SSVs.

The ATWS analysis for the limiting event, Pressure Regulator Failure Open – Maximum Steam Demand (PRFO), was re-performed using the revised opening setpoints for the two MSSVs. The RLA RV values for the opening setpoints based on 1% drift were used as well as the opening time delay of 1.85 seconds. The ATWS peak pressure results for the PRFO event are shown below. The ATWS acceptance criteria, with respect to fuel, vessel and containment integrity, have been met. This is based on the assumption that the SLCS system can deliver the flow specified in the current analysis. SLCS system impacts are discussed in a subsection below.

Event	Fuel Type	Exposure	P_{VESSEL} (PSIG)
PRFO	Legacy	BOC	1499
PRFO	GE14	BOC	1491
PRFO	GE14	EOC	1479

The ASME Overpressure analysis for Quad Cities 2 Cycle 17 has been re-performed using the bounding MSSV opening setpoints. All 9 MSSVs were assumed in-service and the Technical Specification CRD SCRAM times were used. The results of the analysis show the licensing basis acceptance criteria have been met. The results are as follows:

Event	P_{DOME} (PSIG)	P_{VESSEL} (PSIG)	$P_{STEAMLINE}$ (PSIG)
ICF-HBB MSIV Closure (Flux Scram)	1339.1	1360.6	1335.2
MELLA-HBB MSIV Closure (Flux Scram)	1337.8	1358.1	1333.9

The out-of-tolerance MSSV setpoints do not invalidate the Cycle 17 fuel thermal limits analyses. As such, the Cycle 17 fuel thermal limits have been confirmed to remain applicable for the cycle.

NFM Summary

The ATWS, ASME overpressure and fuel thermal analyses applicable to Quad Cities 2 Cycle 17 were re-evaluated in support of operability determinations using revised inputs, which bound the as-found MSSV data. The acceptance criteria for both the ATWS and ASME overpressure analyses have been met. In addition, the applicability of the established Cycle 17 fuel thermal limits has been confirmed.

SLCS System Evaluation

The results of the GE analysis were compared against the results from the Unit 2 SBLC system evaluation conducted under EC 348232, "Analysis of Increased MSSV Setpoint Tolerance on SBLC System Operation". At the End-of-Cycle (the most limiting period of operation for the ATWS secondary peak pressure) the secondary peak reached was 1291 psig at 241 seconds into the event, which is less than the 1306 psig secondary peak pressure that was acceptable in

the Evaluation. Thus, the as-found value of the S/RV lift setting would not have caused any adverse effects on the SBLC system with regard for relief valves lifting if it were needed during the cycle for an ATWS scenario.

Based on the above discussion, the ATWS and ASME overpressure analyses have been met and the applicability of the established Cycle 17 fuel thermal limits has been confirmed. The impact to Appendix R limiting events is discussed below.

Appendix R Evaluation

The Quad Cities Units 1 and 2 Fire Protection Reports, Volume 2, Appendix R Conformance (Sections III.G, III.J, and III.L) Safe Shutdown Report, was reviewed. This report provides descriptions of systems, structures, and components considered important for safe shutdown of the reactor during specific fire scenarios.

The relief valves, including the S/RV, are used for decay heat removal and pressure control. Immediate actions include placing the ADS (Automatic Depressurization System) Inhibit switch into INHIBIT to prevent major loss of reactor inventory. This prevents the relief function of the S/RV from actuating automatically. The relief function of the S/RV, however, is not addressed in this evaluation.

The basic analysis assumptions are included in 10 CFR 50.48, 10 CFR 50 Appendix R, and the emergency operating procedure guidelines. During an assumed fire, a reactor scram would occur, and a relief valve is assumed to be open for the first ten minutes of the event. After 10 minutes, it is assumed that the operators are able to close the open relief valve, and at this point, the reactor begins to repressurize. As stated above, ADS is placed in INHIBIT to prevent the reliefs from automatically actuating from an ADS signal, but the pressure controller relief mode is still available unless the individual valve key switches are in "off". The reactor pressurizes up to the combined safety/relief (S/RV) mechanical safety setpoint and the S/RV relieves pressure down to the S/RV reset pressure. The S/RV will continue to cycle open and closed in this fashion as the S/RV is the only means of decay heat energy removal from the reactor. The S/RV cycling frequency will decrease as the decay heat energy decreases throughout the event. The reactor will undergo alternating periods of S/RV cycling and reactor water makeup operation. This relief valve cycling is postulated to continue for 69 hours.

Root Cause report conclusion for impacts to Appendix R fire scenarios:

The safety function pressure setpoint on the S/RV had increased from approximately 1135 psi to approximately 1213 psi. The typical reset pressure on the S/RV is 96% of the open pressure (1135 open, 1090 reset). Due to the type of component degradation, as opposed to a simple maladjustment of an adjusting nut, the reset pressure cannot be absolutely determined.

The impact on the Appendix R analyses would be on the suppression pool heat up rate and maximum temperature as a result of the relief valve cycling. The analyses assume a scram had been inserted, so there is no impact on fuel or reactor power. The S/RV setpoint in the degraded condition still remains below the 1240 psig Main Steam Safety Valve Setpoint, which should preclude blowdown inside of containment. The amount of energy to be removed from the reactor remains unchanged, but the rate at which it will be removed will be impacted by the S/RV setpoint. The cycling, described as being between the S/RV safety setpoint and

reset setpoint, and lasting 69 hours, would still take place as described, but the pressure region in which the cycling takes place would shift. The open setpoint would shift from approximately 1135 psi to approximately 1213 psi, and would therefore relieve an increased quantity of decay heat energy to the suppression pool with each opening at the beginning of the event.

The reset pressure cannot be determined, but would fall within the two extremes of very high (reduced reset) and normal setpoints.

- If the reset range was reduced as a result of the S/RV degradation, for example, 98% of 1213 psi (greater than 1188 psi), then the amount of heat dumped to the suppression pool would decrease with each short open cycle of the valve, but the time for pressure recovery would also be reduced. There would be more frequent cycling than the analyses postulated. The heat load to the suppression pool cannot be easily estimated (higher energy, but shorter duration), but it is expected that the overall duration of 69 hours would increase. Normal suppression pool cooling would be expected throughout the event.
- If the setpoint were high, at some location greater than 96% of the 1213 psi lift pressure (greater than 1165 psi), then the durations that the relief valve remains open would likely be shorter. The initial operation at the higher than expected lift setpoint would likely result in the duration of cycling going beyond the postulated 69 hours. There would be greater than expected heat input to the suppression pool at the beginning of the event, requiring additional heat removal at the beginning of the event.
- If the reset pressure were at a nominal setpoint of approximately 1090 psi, then each open cycle of the valve would be significantly extended (normal 1135 to 1090 range is 45 psi, degraded 1218 to 1090 range is 128 psi). The result would be that greater amounts of heat and volume would be added to the suppression pool with each cycle, but there would be a corresponding delay before the next cycle took place. Suppression pool cooling demands would increase early in the event. The duration of cycling would likely be far less than the postulated 69 hours.

In any case estimated above, it appears that early heat load to the suppression pool would be greater than that analyzed for the Appendix R events. This would result in the need to utilize additional RHR trains in Suppression Pool Cooling, and could exceed the maximum suppression pool temperature early in the event. The overall duration of relief valve cycling could increase or decrease, based on the different reset pressures. Detailed calculations would be required to determine actual impacts of both the high and normal reset pressures, and their ultimate impact on suppression pool parameters.

Should the reset pressure remain high, the Appendix R analyses may not consider RCIC injection pressures and flows, although Engineering Judgment indicates that flow will be established. The RCIC design basis discusses a flow of 400 gpm at 1120 psig reactor pressure. Surveillance QCOS 1300-05 routinely uses normal reactor pressures (1005 psi) to establish 400 gpm of flow at 1225 psig at the pump discharge. However, as reactor pressure increases, the energy available to the steam driven RCIC turbine will increase, resulting in increased output capabilities. While the Appendix R analyses may not specifically address this situation, Engineering Judgment supports the belief that RCIC will be able to inject make up water to the reactor, despite the high set pressure, and the potentially high reset pressure, of the S/RV.

The conclusion is that the S/RV degradation would clearly have caused situations other than those postulated in the Appendix R Safe Shutdown timelines. Suppression pool parameters would have been challenged, resulting in impacts on RHR, RCIC, and HPCI operation. Additional Operator challenges would have resulted in efforts to either manually control pressure within the expected reactor pressure range utilizing an ERV, or to respond to suppression pool temperatures and levels with increased frequency of RHR system adjustments. RCIC discharge head, while not specifically addressed in the Appendix R analyses in a situation of a high S/RV setpoint or reset, is expected to remain sufficient despite the higher pressures.

References: Fire Protection Reports, Volume 2, Safe Shutdown Report, sections 1.1, 3.1, 5.2.1.5.2, 6.2

12. **Programmatic/Organizational Issues:**

The design review for EPU implemented in Q2R16 did not adequately analyze and disposition the potential effect of equipment vibration on material condition of the plant. This is evident in several degraded conditions, including: steam dryer integrity (CR115510), ERV solenoid actuator (CR 186979), HPCI LLRT tap (CR 135466). The operating experience gained while operating the Quad Cities 1 and 2 has resulted in reduction of power to pre-EPU levels so that further analysis and disposition of EPU-induced vibrations can be understood and actions implemented to prevent further equipment degradation. People Work Practices (PWP) Lack of Knowledge (WPK) and Complex System (4CS) are the principal CR trend codes associated with analysis of vibration induced by the EPU modification.

A Corporate Investgative Team has been assembled to address EPU-related vibration issues at Quad Cities and Dresden.

13. **Previous Events:**

The table below summarizes the results of the OPEX / EPIX search that was performed on this issue. Based on this OPEX / EPIX search, the root cause identified in this report has not been identified previously for Target Rock 3-stage S/RVs in the Nuclear Industry. Based on this finding, further shaker table testing is scheduled to be performed in June, 2004 *(AT 220863-13, A8064MW-DR, 8/12/04)*

EPIX component failure search criteria and results: Manufacturer: Target Rock Corporation. Models: 67F, 67F-000, 67F000, 7467F, Target Rock 67F.

Station	Failure	Date	Failure Mode	Cause	Apply?
Duane Arnold 1	68	11/28/1999	premature opening	250VDC ground and open circuit on solenoid	No
Duane Arnold 1	244	11/20/2002	1. unavailable, not failed 2. found unavailable during nondemand observation	MS solenoid valve failed bench test	No
Duane Arnold 1	285	04/18/2003	failed to open on demand (stuck closed)	blown fuses in control circuit due to electrical short	No
Peach Bottom 3	21	11/28/1997	internal leakage when fully seated	High tailpipe temps, GE SIL 196 premature lift	No
Peach Bottom 3	107	10/20/1999	premature opening	the air plunger assembly failed to return to relaxed position after testing	No
Pilgrim 1	93	06/16/1999	found unavailable during nondemand observation	solenoid for rv-203-3A was found leaking	No
Monticello 1	177	05/25/2003	internal leakage when fully seated	main seat leakage attributed to inadequate rebuild procedure	No
Pilgrim 1	201	05/17/2001	found unavailable during nondemand observation	solenoid for rv-203-3D was found leaking	No
Quad Cities 2	288	01/22/2000	unavailable, not failed	ADS valves were taken OOS when required by TS	No
Vermont Yankee 1	197	10/22/1998	internal leakage when fully seated	bellows leakage	No
Peach Bottom 3	224	09/15/2003	failed to close on demand (stuck open)	pilot leakage due to FME in seat / disk	No
Peach Bottom 3	225	09/15/2003	failed to open on demand (stuck closed)	Diaphragm failure	No

OPEX review was conducted using a keyword search in the Nuclear Network Query Builder (http://www.inpo.org/NN04Asp/NNSearch.asp). The specific criteria utilized included keywords Target Rock, 3-stage, and vibration. The results are summarized below.

Station	Report	Date	Failure Mode	Cause	Apply?
Hatch 1	OE 4552	02/12/91	Setpoint drift on multiple S/RVs	Corrosion induced bonding in 2-stage S/RVs b/t pilot valve disc and seat	No
Hatch 2	OE11444	5/10/00	Leakage past the pilot disc seating surface	Unknown: 1. Mechanical shock during installation 2. Failure to reseat following actuation 3. Flow induced vibration in 2-stage valves	No
BWROG	RSEN 91-02		Setpoint drift on multiple S/RVs	Corrosion induced bonding in 2-stage S/RVs b/t pilot valve disc and seat	No
Monticello	LER 94-011-01	06/07/95	Increase in pilot stem to pilot disc abutment gap	Unknown	Yes

The Monticello LER, while not having the same cause factor, is worthy of discussion based on test results and actions to research the cause of the high lift. As-found

testing identified a 3-stage Target Rock S/RV to be +2.2% higher than nameplate, which was above the 1% Tech Spec tolerance, but within the 3% ASME Code tolerance. Testing and disassembly were performed to address the potential causes of pilot seat leakage, friction, sticking, wear, set spring adjusting nut movement, foreign matter, pilot disk and stem surface irregularities or connection looseness. While the investigation was thorough, it did not clearly identify a single cause for the high lift setpoint, but attributed it to a combination of factors. The Quad inspection approach addressed many of these same factors based on information known at the time of valve disassembly. The Monticello LER validates the areas considered by the Quad Cities Root Cause Investigation.

Quad data was reviewed for Target Rock S/RV setpoint tolerance deviations, and the following two items were identified. Neither report detailed a cause for the deviations other than setpoint drift.

PIF 97-2204/CR Q1997-02204 – Target Rock Safety Relief 1-0203-3A As-Found Setpoint Was Outside of T.S. 1% Limit and Also Exceeded the 3% ASME Limits. The as found setpoint was 1095 psig, which was 3.5% below the 1135 nameplate rating (SN 225, 4/25/97). Analyses by Nuclear Fuels determined that this setpoint did not cause a violation of any applicable accident scenarios. The available documentation surrounding this event did not describe any cause for the condition other than "setpoint drift."

PIF Q1999-01389 – Target Rock Safety/Relief Valve Fails As-Found Setpoint Tests. The as-found setpoint was 1174 psi, which was 3.4% above the 1135 nameplate rating (SN 120, 4/16/99). Analyses by Nuclear Fuels determined that this setpoint did not cause a violation of any applicable accident scenarios. The available documentation surrounding this event did not describe any cause for the condition other than "setpoint drift."

An SRV removed from Dresden Unit 2 after approximately 2 years at EPU operation failed the as-found pressure test in the low direction, at 1094 psig, or -3.6% of the 1135 nameplate set pressure. CR 223897 was initiated to document this condition, and will include an investigation into the cause for this low setpoint. Vendor support for the valve disassembly and testing is scheduled for mid-August, with the final Apparent Cause Evaluation due 9/17/04.

While the low setpoint may be the result of a feature or condition investigated on the Quad SN 172 valve (sticking, wear, FME, pilot disk to pilot stem interface, general damage, etc.), the specific damage identified on the Quad valve does not support a low setpoint scenario. Valve disassembly and inspection will be required to identify the apparent cause.

When the results of the Dresden valve disassembly are obtained, the data should be compared to the general findings of the Quad valve disassembly for any common issues. Should any specific common findings be identified, a supplemental CR should be generated.

- CA: ATI to A8430TP, 9/30/04 to review the data obtained from the Dresden SRV disassembly (reference CR 223897) for any similarity between Dresden and Quad valve conditions. Initiate a new CR should any adverse conditions parallel to the separate valves be identified.

14. **Other Issues:**

High Delay on First Lift

The first lift during the as-found testing of the S/N 172 valve had a higher delay time than the subsequent lifts. While it is possible that the corrosion identified in the first stage bonnet area contributed to this higher delay, this assumption cannot be verified. It should also be noted that the first lift often has additional delay time associated with the clearing of condensate that has gathered in the valve, where subsequent lifts have not allowed sufficient time for condensate to gather in the same quantity. A review of previous tests indicate that delay on the first lift of an as-found test is usually greater than subsequent lifts, as seen in the data below from a partial set of recent as-found tests. Delay time is a typical data point during safety valve testing.

Serial Number	Test Year	First Delay	Second Delay	Third Delay
225	1997	205	160	185
171	1997	190	185	405
198	1998	985	500	550
198	2001	440	440	375
171	2001	700	325	330
198	2002	200	150	180
225	2003	875	420	N/A
171	2003	650	325	N/A

Based on this data, the higher delay time is not being considered an issue in this investigation. The presence of the corrosion, while not ideal, may have contributed to increased delay, but does not impact pressure setpoint. Corrosion was noted by the vendor as being common in this area of the valve.

Because of the great amount of corrosion present, the potential sources of water intrusion into this area of the valve should be investigated, and appropriate corrective actions implemented. The most likely scenarios identified include, but are not limited to:

- Inadvertent use of fluid during IM calibration of Pressure Switch due to test equipment selection or residual fluid presence
- Lack of FME barrier in pressure port allows unidentified source of water to enter bonnet prior to valve installation

One theory previously considered, valve cool down coupled with break in instrument line in presence of humidity draws moisture into bonnet area, has been discounted. In the case of 3 stage valves installed at Limerick station, there is no pressure switch installed at this location, only a tell tale leak off line, open to atmosphere. The lack of historical evidence of severe corrosion in this area on the Limerick valves discounts this theory as being very probable. The source of the corrosion is still unknown at this time. An ATI is assigned to A8430TP to investigate and disposition corrosion found in Target Rock S/RV Bellows Cap area in Section 9 of this report.

15. **Evaluator and Reviewer signatures:**

Ryan Merema _____/_____
Evaluator Date

Greg Boerschig _____/_____
Approved by Dept. Manager Date

Chuck Alguire _____/_____
3rd Party Review Date

Steve Boline _____/_____
RCE Report Sponsor Date

Root Cause Report Quality Checklist

CONDITION REPORT NUMBER__215874__

A. Critical Content Attributes	YES	NO
1. Is the condition that requires resolution adequately and accurately identified?	X	
2. Are inappropriate actions and equipment failures (causal factors) identified?	X	
3. Are the causes accurately identified, including root causes and contributing causes?	X	
4. Are there corrective actions to prevent recurrence identified for each root cause and do they tie DIRECTLY to the root cause? AND, are there corrective actions for contributing cause and do they tie DIRECTLY to the contributing cause? Section 9.B, Section 9.C	X	
5. Have the root cause analysis techniques been appropriately used and documented? Section 9, Analysis Methodology	X	
6. Was an Event and Causal Factors Chart properly prepared? Attachment 1	X	
7. Have cause Codes been identified for equipment problems, human performance, and organizational weaknesses, and have all applicable trend codes been identified and documented in the report and entered into the Condition Report Trend/Cause Panel TIMA017 in Action Tracking? Section 9, ATI assigned to A8450CAP	*	
8. Does the report adequately and accurately address the extent of condition in accordance with the guidance provided in Attachment 3 of LS-AA-125-1003, Reference 4.3? Section 10	X	
9. Does the report adequately and accurately address plant specific risk consequences? Section 11	X	
10. Does the report adequately and accurately address programmatic and organizational issues? Section 12	X	
11. Have previous similar events been evaluated? Section 13. An OPEX / EPIX review was performed to gather industry experience – none was identified with this failure mode. Other events that happened at Quad Cities (but were not evaluated) include: PIF 97-2204/CR Q1997-02204, PIF Q1999-01389.	X	
B. Important Content Attributes		
1. Are all of the important facts included in the report?	X	
2. Does the report explain the logic used to arrive at the conclusions? Section 9, Attachment 12.	X	
3. If appropriate, does the report explain what root causes were considered, but eliminated from further consideration and the bases for their elimination from consideration? Section 9, Attachment 3	X	
4. Does the report identify contributing causes, if applicable? Section 12 addresses contributing cause for this event.	X	
5. Is it clear what conditions the corrective actions are intended to create? Section 9.B, Section 9.C	X	

6. Are there unnecessary corrective actions that do not address the root causes or contributing causes? No, all corrective actions associated with Root Cause or administrative requirements associated with this event (NER, LER, OPEX generation as appropriate).		X
7. Is the timing for completion of each corrective action commensurate with the importance or risk associated with the issue?	X	
C. Miscellaneous Items	**YES**	**NO**
1. Did an individual who is qualified in Root Cause Analysis prepare the report? Section 5, Report prepared by Ryan Merema under instruction of Chris VanDenburgh (RCI qualified).	X	
2. Does the Executive Summary adequately and accurately describe the significance of the event, the event sequence, root causes, corrective actions, reportability, and previous events? Section 6	X	
3. Do the corrective actions include an effectiveness review for corrective actions to prevent recurrence? Section 9.C	X	
4. Were **ALL** corrective actions entered and verified to be in Action Tracking? This will be completed after final acceptance at MRC on 6/9/04.	*	
5. Has an Operating Experience database search been performed to determine whether the problem was preventable if industry experience had been adequately implemented? Yes, Section 13	X	
6. Are the format, composition, and rhetoric acceptable (grammar, typographical errors, spelling, acronyms, etc.)?	X	

Ryan Merema
Root Cause Investigator / Date

Greg Boerschig
Department Head / Date

Chuck Alguire
3rd Party Review / Date

Steve Boline
Sponsor Manager / Date

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Attachment 3: QCGS System Engineering Trip Report for Valve Disassembly and Troubleshooting at Wyle Laboratories

Arrangements were made to have QCGS and Exelon corporate personnel present at Wyle Laboratories to supervise and document Target Rock S/RV disassembly and inspection, which took place between 5/11/2004 and 5/12/2004. Notes compiled by Quad Cities personnel during disassembly and inspection of the valve are as follows:

- Pilot assembly housing was removed from main valve housing and installed on a test stand.
- Nitrogen pressure testing was attempted to validate the as-found set pressure from the steam test. Seat leakage prevented this test from working. It was noted that approximately 10% of valves do not successfully test with nitrogen following the as-found steam testing, and this should not be considered a cause for failure.
- The bonnet cap was removed to allow access to the 1st stage pilot assembly. Some general corrosion was present in the areas visible at this point. It was also noted that the Cap assembly was not centered in the adjusting ring, but was tilted slightly toward the direction of the air operator.
    - This tilt, while typical of the valve assembly, is key to the Root Cause of the high set pressure. The design of the valve allows this tilt. Inspection of additional valves at Wyle Laboratories indicated that tilt allowed spring contact with cap in all sampled cases. The tilt did not cause degradation on other valves or previous Quad Cities valves (see notes below).
- The Adjusting Nut was removed, allowing the Cap and Spring to be removed from the bonnet. Upon removal, there was considerable corrosion, including loose corrosion, at the base of the cap, on the spring, and in the base of the pilot section. The vendor indicated that this section experiences some corrosion, but the level of corrosion found on this valve was more extensive than typically seen. This section of the valve is not pressurized. The bellows failure pressure switch is connected to a port on this chamber. At some plants, the port is open to atmosphere.
    - The corrosion likely lead to the high delay time during the first as-found lift, and may have contributed slightly to the high set pressure.
    - The source of water which lead to the corrosion should be investigated.
- Under the belief that an o-ring or seal weld at the bellows was leaking (although the plant did not receive any such alarm), the bellows was pressurized and snooped. No leakage was present. In order to further confirm that no leakage was present, the bonnet area was flooded with the bellows still pressurized. Again, no leakage was present.
- The Hold Down Nut for the Pre-Load Spacer was inspected, and the lockwire was still installed from the previous refurbishment. Dimensions from the flange to the Hold Down Nut were as expected, and the Nut was removed.

- The 1st stage Stem and Disk were removed. While some corrosion was present, it was typical of this valve section as described by the vendor. There was no binding in the yoke connection between the stem and disk components, all pins were present and tight, and the spring was present and not broken. No signs of vibration related damage was seen. The seat ring on the disk looked clean, although there was some corrosion on the nose of the disk. The seat appeared clean as well, with no obvious scoring, cutting, or corrosion on the seat ring. Abutment gap measurements were good, and disk to yoke gap measurements (after disassembly) were good.
- The pilot stage filter was inspected, and did not appear clogged. There were some specks on the filter, as expected, but nothing out of the ordinary (large pieces, large quantity of small pieces, etc.) was identified.
- Cap, Spring, Stem, Disk, Hold Down Nut, Adjusting Nut, and other parts were sent to decon for cleaning prior to further inspection.
- Following cleaning, there were clear indications of contact between the Cap and Spring which are tensioned under the Adjusting Nut for final pressure set point adjustment. There were two wear areas noted, with one caused by the uppermost partial coil (which is ground to provide a parallel seating surface) and one caused by the first full coil. The lower of the two, caused by the first full coil, had a depth of 0.008 inches. A small contact point was also noted on the interior of the cap, where it was riding at the base of the bellows; there was no apparent depth to the wear mark, see Attachments 14, 15, and 16.
- Inspections were made of another Quad Cities valve (removed during Q1R17), a Limerick valve, and a Peach Bottom valve. All three additional inspections indicated the presence of contact in this same area between the Cap and Spring, but only Quad Cities Target Rock serial #172 showed any depth in the area of the wear marks.
    - The Spring would be recessed into the groove on the cap. Additional force, provided by additional system pressure, would be required to cause the spring to "pop out" of the gap (Root Cause).
    - Consistent pressures were noted during all three test runs, with only the delay time changing between the first run and the subsequent two runs. The corrosion noted outside of the bellows in the first stage bonnet likely caused increased friction during the first test run, during which the corrosion products were cleared from the areas of contact. On the second and third run, the delay times were both normal, and consistent.
- The tilt of the Cap and Spring is a result of the design of the assembly, and includes friction that is considered normal by the Vendor. The Cap is supported in the center of the interior top surface on a ball bearing. This ball bearing rides in the top of the Pre-load spacer, and allows some tilt of the cap. The Spring is installed, and tensioned with the Adjustment nut. While the spring ends are ground to be generally parallel at full height, there may be some deviation during compression. Also, a portion of the end coil is ground flat, but there is not full 360 degree contact between this spring end and the cap ledge upon which it rides. For these reasons, the tilt identified is to be

expected, with the contact point being at the top of the spring on one side and the interior base of the wall on the opposite side. This is supported by the similar contact points seen on three other valves in the shop.

The Wyle personnel were interviewed for a clear understanding of the as-found and as-left steam tests performed on the valves. Two specific areas of interest included the possibility of Adjusting Nut motion following testing, or the accumulation of water in the first stage bonnet following testing. The following information was obtained, with the following reasoning that the testing is not considered a contributing factor to the high set pressure:

- The valve is received in the shop, and has a very limited amount of exterior decontamination performed.
- The first stage bonnet cover is removed in order to connect the Pilot LVDT measurement instrument to the valve.
    - While this does result in an anti-rotation pin between the bonnet cap and the adjusting nut becoming free, there is no rotation involved in removing the bonnet cap.
    - A full rotation of the adjusting cap provides approximately 60 to 75 psi set point change. In the case of this test, an adjustment of a more than a full rotation would be required to cause this error. This is not realistic.
    - In the case of as-left testing, Wyle understands that the anti rotation pin must align between the cap and the adjusting ring. For this reason, a tool has been created to guarantee proper alignment without the presence of the bonnet cap. Following testing, when the bonnet cap is reinstalled, no adjusting ring rotation takes place, or it is very minimal at most.
- The bonnet cover is not present during testing.
    - The valve is heated to normal operating temperature, and allowed to soak completely.
    - Any water present at the beginning of the test will be boiled off during the test.
    - Any leakage in this location would be identified during testing. No leakage was identified during certification testing.
- The bonnet cover is reinstalled following testing. An FME cover should be placed in this port.
- Wyle test documentation includes notation of any adjustment nut changes that are made. During as-left certification for this valve, a rotation of 5 holes counterclockwise was noted prior to the 3 final test runs.
- Wyle personnel showed a very clear understanding of this valve style, and the methods to adjust set pressure. The also showed a very clear understanding and expectation that no work which can impact set pressure will be performed prior to as-found testing.
- There were no findings from a review of Wyle test data, nor from interviews with Wyle personnel, to indicate that set pressure changes were a result of testing activities.

Attachment 4: Target Rock S/RV Testing Methodology

The Safety Setpoint of the S/RV must be as found tested within one year of removal from the plant in accordance with OM Code. For Quad Cities, this testing is performed at the S/RV Test Facilities at Wyle Laboratories in Huntsville, AL. Wyle Test Procedure number 1100 has been approved by Quad and Wyle to accomplish this testing.

The general intent of the procedure is to simulate as-installed conditions on the test stand in order to obtain accurate as-found set pressures, as well as to certify and set the valve following maintenance. General requirements include adequate valve heat up and soak time, adequate ambient temperatures and valve insulation, and adequate steam pressure, temperature, and supply volume.

Points monitored include steam supply temperature and pressure, valve body and bonnet temperatures, ambient temperatures, pilot disc position via an LVDT, and main disk movement via an accelerometer.

As found set point is tested as follows. After valve and test stand preparation, system heat up, and adequate soak time, inlet pressure is increased slightly above normal operating pressure in order to determine the presence of leakage. Pressure is then ramped up at a rate of 60 to 150 psi per second until the valve actuates, at which time the as-found set point is identified. Two additional runs are performed to test the safety set point of the valve. A final test run using the relief function (in this case, via air operator) is performed. Test data is collected in a data acquisition system and in test logs.

For post-rebuild certification, the sequence of events is very similar, but includes additional inspections for leakage and for setpoint adjustments.

A test report is generated to include the following data points:

- Customer, valve number, test number, test date, and tracking information
- Run date, time, and sequence number
- Set Pressure
- Inlet steam pressure ramp rate
- Delay time (time from initial opening of 1st stage pilot (or solenoid energized for air operator) to initial movement of main disk)
- Main disk stroke time (from initial movement to full open)
- Steam, valve body, and ambient temperatures
- Test curves of inlet pressure and pilot disk motion
- Instrument and calibration lists

Attachment 5: Target Rock 3-stage S/RV Description and Operation

The Target Rock Safety Relief Valve (S/RV) model 7467F is a 3 stage, pilot operated valve, which includes both a Safety function (lift on high pressure only) and a Relief function (lift on electrical control signal). The operation of the valve can be most easily understood by starting with the 3rd stage operation and working backwards to the first stage:

- The 3rd Stage, or the main disk, is pressure seated by a combination of a valve preload spring, system pressure over the main valve disk, and system pressure over a main valve piston. While in the closed, no flow conditions, the static pressures are equal between the valve body and the chamber above the main valve piston due to design leakage through gaps, drains, and an orifice.
- The 3rd stage disk is opened by venting the pressure from the chamber above the main valve piston. The system pressure below the main valve piston is then greater than the pressure above the piston, resulting in opening motion of the main disk. When the venting is no longer present, pressure entering the chamber through the orifice and through design leakage will provide sufficient force to re-close the main disk.
- As soon as the disk begins to open, the hydraulic seating force is reduced. The system pressure acting below the main valve piston without this seating force present results in the characteristic full opening, or popping, action.
- The 2nd Stage disk maintains the pressure boundary for the chamber above the main valve piston, and thus controls the position of the main disk. The 2nd stage area is vented to the discharge of the valve, and is not a pressurized area of the valve during normal operation. The second stage disk is held closed by a preload spring and by system pressure.
- The 2nd stage disk can be actuated via two different means. It can be manually opened (Relief Function) by using an air operator that depresses upon the 2nd stage pilot rod. It can be hydraulically opened (Safety Function) by allowing system pressure to be applied above the 2nd stage piston, resulting in sufficient force to open the 2nd stage disk.
- Hydraulic opening of the 2nd stage disk is controlled via the 1st stage pilot. The safety lift set point is established by adjustments to the 1st stage pilot.
- The 1st stage pilot assembly consists of a mechanical bellows, a pre-load spring, the first stage disk, the first stage stem, an adjusting nut, and a spring and cap. (Additional first stage components exist, but are not critical to the discussion of basic valve operation.) The mechanical bellows extends as system pressure is applied to the inside of the bellows; there is roughly 0.0035 inches of movement per 100 psi of pressure due to bellows and adjustment spring tension.
- During valve assembly, the bellows is mechanically extended a slight amount to provide a preload force on the 1st stage stem, which is transferred to the 1st stage disk to prevent leakage at low system pressures. This preload also provides a rough lift set point independent of the adjusting nut and spring.

- A cap and spring assembly rides on the mechanical bellows, and an adjusting nut is used to provide additional load and provide fine adjustment of the ultimate lift set point.
- In operation, the bellows extends as pressure increases. As pressure increases, the preload force is reduced to zero, at which time the 1st stage disk is being held closed only by internal pressure acting over the seat area.
- The 1st stage disk is coupled to the 1st stage stem and bellows through a yoke that has a specific clearance designated as an abutment gap. At low pressure, there was no gap present, and the bellows and springs were providing closing forces on the disk through the stem. At normal operating pressure, there is no preload pressure, and there is a gap present between the stem and disk (abutment gap).
- As pressure increases near the safety set point, the bellows pulls the stem to the point where the abutment gap decreases to zero, and opening forces are applied to the 1st stage disk.
- The set point is reached when the bellows develops sufficient force for the stem to pull the disk open against the system pressure seating the disk. As flow begins, the seating pressures further decrease, and the 1st stage fully opens with the characteristic popping of this style of valve. System pressure is now provided above the 2nd stage piston area, which actuates the second stage, which actuates the 3rd stage.

Attachment 17: Equipment Apparent Cause Evaluation Guide, LS-AA-125-1003

The Equipment Apparent Cause Evaluation checklist was reviewed for potential impact on the S/RV degradation. The review did not identify any obvious contributors associated with maintenance or quality issues. The design of the valve is identified as an issue only when coupled with increased vibration amplitudes; testing will take place to confirm this. The checklist topics are provided below.

Run to Failure Classification Check

The S/RV is not a Run To Failure component.

PM/PDM Review

The station PM for the main valve includes a 1 cycle replacement on a 24 month period. This valve is changed out each outage with a fully refurbished and recertified valve. The as-found setpoint is tested within 1 year of valve removal in accordance with the Code requirements.

The PM's for this valve are current and adequate.

Maintenance Performance Assessment

Maintenance is performed by the valve manufacturer, Target Rock, and is certified by a qualified vendor, Wyle Laboratories. Valves are inspected and maintained in accordance with the vendor procedures. No assembly issues (missing or loose fasteners, missing locking devices, etc.) were identified during pilot valve disassembly. The identified degradation was not a result of valve maintenance, but a result of vibration induced wear.

Performance Monitoring Assessment

Performance monitoring of the valve following certification only addresses seat leakage or bellows failure. During the previously installed cycle, no abnormalities were identified in either case.

The certification of the valve includes verification of set pressure and seat tightness. The certifications were reviewed and found to be acceptable.

Operating Experience Review

A detailed OPEX review is included in Section 13 of ATI 220863-20.

PCM Template Review

The PCM template for this style of valve was reviewed. The PCM template includes maintenance and setpoint verification on a 10-year frequency, but specifically notes that any Code or Regulatory requirements take precedence over the template. The Code requirements for a 1-cycle replacement are adequately addressed by the PM program.

Operational Performance Review
Operations previously tested this valve following maintenance by cycling the valve with the relief function at 300 psi reactor pressure (QCOS 0203-03). While the two times that this test was performed were both successful, the test does not verify the safety function of the valve. The certification test of the valve is the only opportunity to test this setpoint.

Maintenance Practice Review
A review of maintenance practices of Target Rock Field Service and Wyle Laboratories identifies only a historic issue of cleanliness. In some cases, very small pieces of debris from the refurbishment workscope would be captured on a seat, and lead to leakage. This situation has improved in recent years, and is a subject of ongoing scrutiny. Such FME concerns would not have lead to the type of degradation reviewed in this report.

Design Review
This style of valve is used on a number of GE BWRs in the United States with acceptable performance. The design of the valve does allow contact between the Cap and Spring as described in this report, and this fact was verified by inspecting the caps of three other valves that were in process at the test and refurbishment facility. The presence of contact has not, however, caused any measurable degradation on these other 3 sample valves. Thus, it appears that the design is acceptable, provided there are acceptable vibration levels in the plant.

Ongoing testing will determine critical frequencies and amplitudes associated with this assembly. Design material changes for the cap are being considered to eliminate the wear introduced by the identified component vibrations or resonance.

Manufacturer / Vendor Quality Check
No recent quality issues have been identified with Target Rock or Wyle Labs. No component quality issues are suspected associated with the degradation identified.

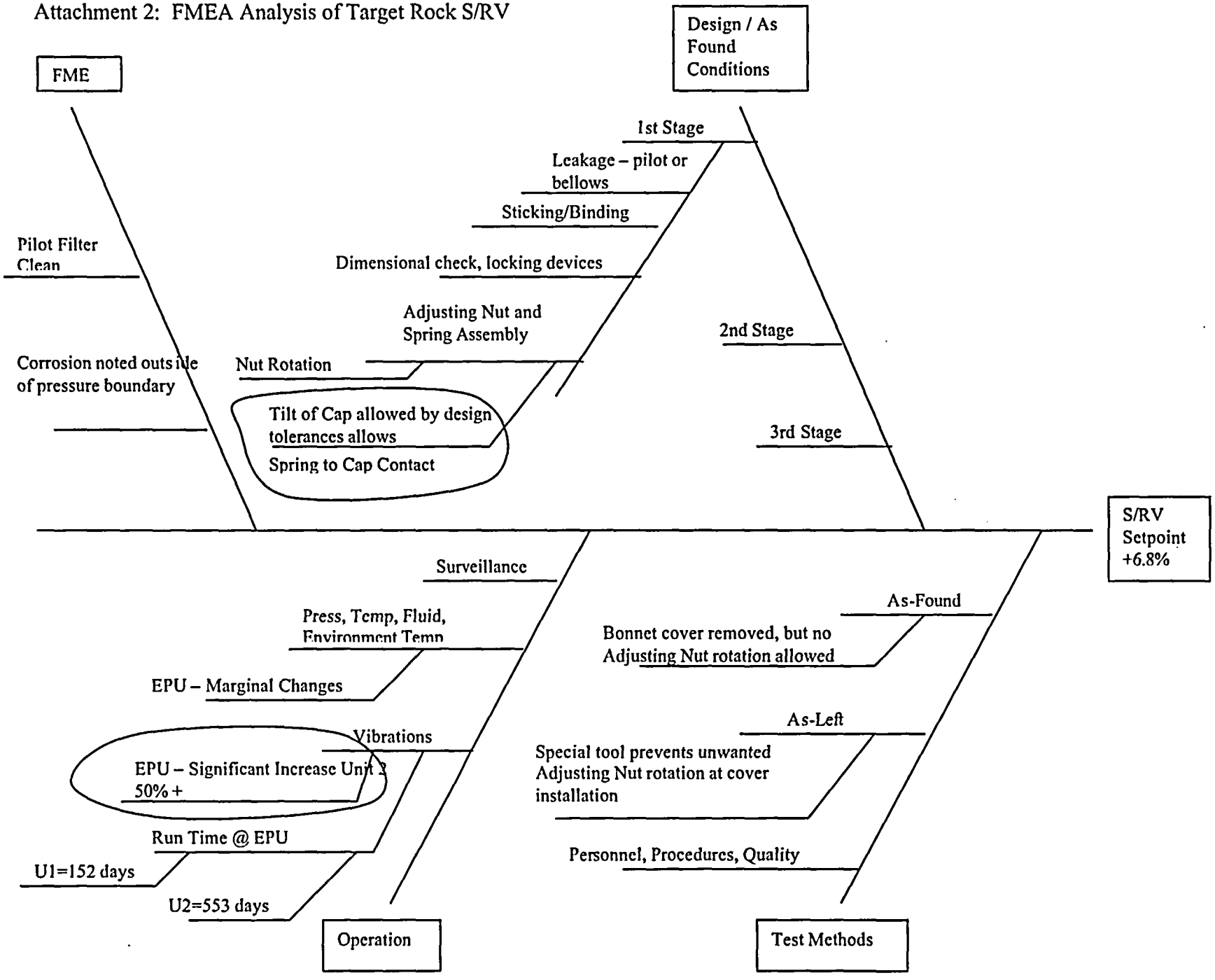
Problem / Issue Management Review
The presence of increased vibrations due to EPU has impacted Quad Cities in many areas, including vibration induced wear on other main steam relief valves. Original analysis of vibration impacts did not address subcomponents of specific valves or components, but such a review is currently in progress.

Unknown or Different Causal Factor
A specific unknown issue, that being critical frequencies for the S/RV subcomponents, will be clarified during upcoming tests on the S/RV or S/RV subassemblies. On other relief valves installed at Quad, it was discovered that vibrations in a certain frequency range were resulting in self-excitation of a spring

supported solenoid core. The identified frequency range was not previously considered a critical area for the specific component. The results of the S/RV testing will be compared to the known vibration spectrums on the main steam lines to determine appropriate corrective actions.

Other causal factors for the high setpoint (pressure leakage, FME in specific areas, locking devices or pins missing or loose, inadvertent setpoint nut adjustment, obvious damage other than cap and spring) were investigated and ruled out during valve disassembly.

Attachment 2: FMEA Analysis of Target Rock S/RV



ATTACHMENT 4B

Quad Cities Nuclear Power Station, Units 1 and 2
Operability Determination 220863-08, Revision 5
"1(2)-0203-3A Target Rock Safety Relief Valve"

1.0 ISSUE IDENTIFICATION:

1.1 CR #: 220863

1.2 Operability Determination #: 220863-08 Revision: 5

General Information:

1.3 Affected Station(s): Quad

1.4 Unit(s): 1 and 2

1.5 System: RX

1.6 Component(s) Affected: 1(2)-0203-3A Target Rock Safety Relief Valve

1.7 Detailed description of what SSC is degraded or the nonconforming condition and by what means and when first discovered: The 1st stage set pressure set point (Safety Function) of the Unit 1 Target Rock Safety Relief Valves (S/RV) 1(2)-0203-3A may be higher or lower than expected due to vibration induced wear between the Bellows Assembly Cap and the Pressure Adjustment Spring. This condition was identified on an S/RV removed from Quad Unit 2 at the end of Cycle 17 when the valve was tested at +6.8% of nameplate pressure, which exceeds both Tech Spec and Code requirements. The subcomponent degradation was identified on 5/12/04, during Exelon overview of the vendor (Target Rock) disassembly of the pilot stages of the valve. Later valves from Dresden Pre- and Post-EPU operation exhibited similar wear characteristics and experienced low lift pressures of –1.4% and –3.6%. These data points were not available at the time this report was originally generated.

The presence of such vibration induced wear may cause the Spring, which provides fine adjustment of the valve safety set point, to cause and be captured in a groove in the Bellows Assembly Cap. Additional force, in the form of additional main steam pressure, is then required to cause the spring to exit the gap and allow the 1st stage pilot to open. Likewise, alignment of the spring and cap positioning the spring at the upper edge of a groove as the setpoint is approached will be adding lifting forces at the edge of the groove, and will require less system pressure in order to open the seat . This condition only impacts the Safety function of the valve, and does not impact the Relief function (manual, pressure controller, or ADS operation of the Air Operator) of the valve. 9 of 9 Safety Valves, including 8 Main Steam Safety Valves (MSSV) as well as the Safety Function of the S/RV, are required to be operable in Plant Modes 1, 2, and 3.

The Unit 2 S/RV operability will also be addressed in this Operability Determination. The Shift Manager review of CR 220863 states that the Unit 2 S/RV is Operable because the unit power has been held to pre-EPU limits; this Operability Determination will support this position and define actions to maintain operability. The additional information provided by the Dresden S/RV's in IR 255880 indicates that the wear is not specifically EPU related, and further indicates that the impact on set pressure may be high or low. (Revision 5)

Attachment 1 is provided for a basic understanding of the valve components discussed in this report. It includes valve drawings and photos of the as-found component degradation from the unit 2 S/RV.

Revision 4 is a complete rewrite of the Operability Determination. While the conclusions and the corrective actions were basically unchanged, a core argument was revised based on data obtained since the original evaluation. The core argument in the original document was that EPU level vibrations specific to Quad Cities resulted in valve component degradation. Revision 1 added Appendix R impact to the evaluation scope. Revisions 2 and 3 revised due dates for corrective actions associated with testing.

The purpose of Revision 4 is to introduce the new data from the Dresden valve disassemblies as documented in IR 255880, as well as factors identified in shaker table testing to date. It has been found that the wear induced degradation is not EPU related, but is most likely a function of spring tolerances combined with typical main steam line vibrations. This effect may cause an increase or decrease in S/RV setpoint; however, based on reviews of historical as-found setpoints, the impact of this phenomenon is bounded by previous assessments and revised assessments presented below. Attachment 2 was added to document a revision to calculation MWMECH-04-002.

The purposes of Revision 5 is to make minor wording changes per MRC recommendations, to reinstate the EPU Power Level Restrictions as a conservative measure per MRC recommendation, and to adjust corrective action due dates to accommodate ongoing testing and valve improvement activities. The minor wording changes, and the re-instatement of the previous EPU power level restrictions, are intended to reflect the fact that the valve degradation is not specifically an EPU related phenomenon, but is indeed a function of vibration, which is known to have increased at EPU operating levels at Quad Cities that are generally higher than other fleet EPU sites. The due date changes for the corrective actions concerning testing and modification of the valves are being extended to allow for comprehensive documentation of testing activities and conclusion. The due dates established will still allow time for Quad valves to be upgraded or modified, and be certified prior to the spring Refuel and Planned Outages. (Revision 5)

2.0 EVALUATION

2.1 Describe the safety function(s) or safety support function(s) of the SSC. As a minimum the following should be addressed, as applicable, in describing the SSC safety or safety support function(s):

The Main Steam Safety Valves (1(2)-0203-4A through H) and the Safety function of the Target Rock Safety Relief Valve (S/RV) (1(2)-0203-3A) are installed to protect the reactor pressure vessel from overpressurization. Valves are removed and as-found tested during outages and replaced with tested and certified spares. The replacement valves are certified to have tolerance within +/- 1% of their setpoint. The ASME Code requires an as-found tolerance of +/- 3% although Technical Specifications requirements in section 3.4.3 requires +/- 1% tolerance. The MSSVs and S/RV tolerance is an input into the ASME overpressure and the ATWS overpressure analyses. The impact of these analyses is described in various sections of the Technical specifications and the UFSAR and sections are included below.
The ASME Boiler and Pressure Vessel code requires the reactor pressure vessel be protected from the consequences of pressure and temperature in excess of design conditions by self-actuating safety valves. As part of the nuclear pressure relief system, the size and number of safety valves are selected such that peak pressure in the nuclear boiler system will not exceed the ASME code limits for the reactor coolant pressure boundary.

Each unit is designed with nine safety valves, one of which also functions in the relief mode. This valve is a dual function Target Rock Safety/Relief Valve (S/RV).

The overpressure protection system must accommodate the most severe pressurization transient. Evaluations have determined that the most severe transient is the closure of all Main Steam Isolation Valves (MSIVs) followed by reactor scram on high neutron flux (reference UFSAR Section 5.2.2). In the existing ASME Overpressure analysis, all 9 safety valves are assumed to operate in the safety mode. The analysis results demonstrate that the design safety valve capacity is capable of maintaining reactor pressure below the ASME code limit of 110% of vessel design pressure (110% x 1250 psig = 1375 psig). The 1375 psig is the limit at the lowest elevation of the Reactor Pressure Vessel (RPV) and is equivalent to approximately 1345 psig as measured at the reactor steam

dome. The safety valves are credited to operate during the most severe pressurization transient to assure that the 1375 psig limit is not exceeded.

In addition, both the safety valves and relief valves are also credited in the ATWS overpressure analysis to keep peak vessel pressure less than 1500 psig.

- Does the SSC receive/initiate an RPS or ESF actuation signal?

    - Yes, the S/RV does receive an ESF actuation signal from the Automatic Depressurization System (ADS) to depressurize the reactor vessel to allow low pressure ECCS systems to inject coolant. However, this relief function is not impacted by the scope of this Operability Determination, because the safety function of the S/RV does not rely on any actuation signals. Only system pressure of sufficient magnitude is required to operate the safety function of the valve.

- Is the SSC in the main flow path of an ECCS or support system?

    - Yes, the S/RV Relief Function supports the ECCS system by using ADS to cause rapid depressurization of the vessel to allow low pressure ECCS system injection. The relief function, however, is not impacted by the scope of this Operability Determination.

- Is the SSC used to:

- Maintain reactor coolant pressure boundary integrity?

    - Yes, the S/RV body and seats, including the first stage bellows, provide a reactor coolant pressure boundary. The particular wear points addressed in this Operability Determination, the Adjusting Spring and Bellows Assembly Cap, do not, however, provide a coolant boundary.

- Shutdown the reactor?

    - No, the S/RV safety function does not act to shutdown the reactor.

- Maintain the reactor in a safe shutdown condition?

    - Yes, the relief function of the S/RV helps to maintain the reactor in a safe shutdown condition in some scenarios. The safety function, however, is not used to maintain a safe shutdown condition.

- Prevent or mitigate the consequences of an accident that could result in offsite exposures comparable to 10 CFR 50.34(a)(1) or 10 CFR 100.11 guidelines, as applicable.

    - Yes, the Main Steam Safety valves including the safety function of the S/RV provide over-pressurization protection for the reactor vessel during upset conditions. This function could prevent or mitigate the consequences of an accident that could result in offsite exposures comparable to 10 CFR 50.34(a)(1) or 10 CFR 100.11 guidelines by preventing the vessel from exceeding design pressure conditions. The S/RV relief function supports the ability to prevent or mitigate the consequences of an accident through their ADS function, although this Operability Determination does not impact the relief function.

- Does the SSC provide required support (i.e., cooling, lubrication, etc.) to a TS required SSC?

    - Yes, the S/RV relief function provides support to the low pressure ECCS system by reducing reactor pressure to allow injection. The safety function, however, does not provide support to other SSC's.

- Is the SSC used to provide isolation between safety trains, or between safety and non-safety ties?

- o No, the S/RV does not provide isolation between safety and non-safety trains. The S/RV safety function and Main Steam Safety Valves each work independently.

- Is the SSC required to be operated manually to mitigate a design basis event?

  - o Yes, the S/RV relief function has both manual and automatic actuation capabilities. However, this Operability Determination does not address the relief function. The safety function cannot be manually actuated.

- Have all safety functions described in TS been included?

  - o Yes, the Over-pressurization function of the S/RV safety function has been described above. The ADS relief function has been noted as appropriate, but is should be clearly noted that the relief function of the valve is unaffected by the condition being addressed in this Operability Determination.

- Have all safety functions described in UFSAR or pending revisions been included?

  - o Yes, the Over-pressurization function of the S/RV safety function is described below. The ADS relief function has been noted as appropriate, but is should be clearly noted that the relief function of the valve is unaffected by the condition being addressed in this Operability Determination. The Appendix R Safe Shutdown Report S/RV function is also described below.

- Have all safety functions of the SSC required during normal operation and potential accident conditions been included?

  - o Yes, the safety functions have been reviewed.

- Is the SSC used to assess conditions for Emergency Action Levels (EALs)?

  - o Yes, the relief function of the S/RV is used to assess the EAL. However, only the safety function of the S/RV is addressed in this Operability Evaluation.

Technical Specification Bases section 3.4.3 states the basis for the Safety Valves:

> "The ASME Boiler and Pressure Vessel Code requires the reactor pressure vessel be protected from overpressure during upset conditions by self-actuated safety valves. As part of the nuclear pressure relief system, the size and number of safety valves are selected such that peak pressure in the nuclear system will not exceed the ASME Code limits for the reactor coolant pressure boundary (RCPB). Each unit is designed with nine safety valves, one of which also functions in the relief mode. This valve is a dual function Target Rock safety/relief valve (S/RV)."

UFSAR Section 5.2.2.1 further states:

> "The safety valves are sized to protect the RPV against overpressure during a MSIV closure at full power, a failure of the reactor relief valves, a failure to scram from MSIV position switches, and a backup scram due to high neutron flux that shuts down the reactor (see Section 5.2.2.2.3 for further details). The ASME Code requires that each vessel designed to meet Section III be protected from the consequence of pressure and temperature in excess of design conditions. The USAS B 31.1 Code for Pressure Piping also requires overpressure protection."

In addition to the ASME code overpressure analysis, Anticipated Transients Without Scram can also cause an overpressure condition in the reactor pressure vessel. The ATWS scenario is described in UFSAR section 15.8:

> "This section covers the events, which result in an anticipated transient without scram (ATWS). Anticipated transient without scram events are beyond design basis accidents. Anticipated transients without scram are those low probability events in which an anticipated transient occurs and is not followed by an automatic reactor shutdown (scram) when required. The failure of the reactor to scram quickly during these transients can lead to unacceptable reactor coolant system pressures and to fuel damage. Mitigation of the lack of scram must involve insertion of negative reactivity into the reactor, thereby terminating the long-term aspects of the event.
>
> The occurrence of a common-mode failure, which completely disables the reactor scram function, is a very low probability event. Therefore, no significant risk to public safety is presented by the combination of an infrequent event and a common-mode failure, which prevents scram. Thus, attention is focused on those transient situations, which have a relatively high expected frequency of occurrence at a power condition at which serious plant disturbance might result.
>
> GE has performed a plant-unique ATWS analysis using Plant Design Licensing Basis (PDLB) approach for the Dresden and Quad Cities units. The PDLB approach establishes the analysis bases applicable to the four Dresden and Quad Cities units. The analysis was performed at the Quad Cities original licensed reactor power level (2511 MWt) and Extended Power Uprate (EPU at 2957 MWt). The original Quad Cities licensed power was chosen to provide the largest change in power and the maximum effect of EPU to the ATWS compliance criteria. The GE analysis shows that pressure regulator open to maximum demand (PRFO) [Note: Pressure Regulator Fail Open – essentially an Oversteam demand event that is terminated by MSL Low Pressure Group I isolation.] is the limiting event. The results confirm that the analysis meets the acceptance criteria of peak vessel pressure, peak clad temperature, peak clad oxidation, peak suppression pool temperature and peak containment pressure for GE14, ATRIUM-9B, and GE9/0 fuel types at 2957 MWt."

Although ATWS is not a design basis event, it is part of the Quad Cities Licensing Basis and thus is included in the discussion.

Appendix R requirements include the following:

> The Quad Cities Units 1 and 2 Fire Protection Reports, Volume 2, Appendix R Conformance (Sections III.G, III.J, and III.L) Safe Shutdown Report, was reviewed. This report provides descriptions of systems, structures, and components considered important for safe shutdown of the reactor during specific fire scenarios.
>
> The relief valves, including the SRV, are used for decay heat removal and pressure control. Immediate actions include placing the ADS (Automatic Depressurization System) Inhibit switch into INHIBIT to prevent major loss of reactor inventory. This prevents the ADS relief function of the SRV from actuating. The relief function of the SRV, however, is not addressed in this evaluation.

The basic analysis assumptions are included in 10 CFR 50.48, 10 CFR 50 Appendix R, and the emergency procedure operating guidelines. During an assumed fire, a reactor scram would occur, and a relief valve is assumed to be open for the first ten minutes of the event. After 10 minutes, it is assumed that the operators are able to close the open relief valve, and at this point, the reactor begins to repressurize. (As stated above, ADS is placed in INHIBIT to prevent the reliefs from automatically

actuating from an ADS signal, including the relief function of the SRV.) The reactor pressurizes up to the combined safety/relief (SRV) mechanical safety setpoint and the SRV relieves pressure down to the SRV reset pressure. The SRV will continue to cycle open and closed in this fashion as the SRV is the only means of decay heat energy removal from the reactor. The SRV cycling frequency will decrease as the decay heat energy decreases throughout the event. The reactor will undergo alternating periods of SRV cycling and reactor water makeup operation. This relief valve cycling is postulated to continue for 69 hours.

2.2 Describe the following, as applicable: (a) the effect of the degraded or nonconforming condition on the SSC safety function(s); (b) any requirements or commitments established for the SSC and any challenges to these; (c) the circumstances of the degraded/nonconforming condition, including the possible failure mechanism(s); (d) whether the potential failure is time dependent and whether the condition will continue to degrade and/or will the potential consequences increase; and (e) the safest plant configuration, including the effect of transitional action:

(a) The effect of the potential degradation would be an increase or decrease in the Safety function set pressure on the S/RV. This is based on the Unit 2 S/RV testing at +6.8% of nameplate pressure, and two Dresden S/RV's tested and disassembled at a later date, showing similar wear characteristics with set pressures at –1.4% and –3.6% of nameplate pressure. The S/RV Safety Function Setpoint is certified to be within +/- 1% tolerance when originally installed in the plant. One of the inputs into the ASME Overpressure and ATWS Overpressure analyses for the Reactor Pressure Vessel is safety valve tolerance, including the S/RV. This tolerance is currently specified at +1%, but the degradation to the Unit 2 S/RV caused that setpoint to increase. The effect of a higher valve setpoint may result in a higher peak pressure in the RPV during ASME or ATWS Overpressure events. The effect of a lower valve setpoint would be a decrease in the expected pressure peak in the same scenarios. As this is only one of 9 valves, the peak pressure would not be expected to increase or decrease more than marginally.

(b) The requirements for the Safety function of the S/RV are defined in Quad Cities Technical Specifications 3.4.3, which requires 9 of 9 Safety Valves. This includes eight Dresser Main Steam Safety Valves (1(2)-0203-4A through H) and the Target Rock S/RV (1(2)-0203-3A). The tolerance specified in Tech Specs is +/- 1%. While the availability of the valve is not challenged, the tolerance would be challenged by the potential degradation.

(c) The degradation appears to be related to dimensional tolerances on the spring, and does not appear to be specific to EPU level vibrations. While all springs checked to date have been within manufacturer's tolerances, some exhibit more skew than others, or have more "tilt." Shaker table testing to date indicates the following, which is supported by inspection of a random sampling of valves present at the test facility:

i. A spring with greater skew appears to cause wear at the upper end of the cap, primarily at the first full coil. This results in a higher force concentration, with most of this lateral force from greater skew being applied at a single spring coil location. Similar markings, with no discernable wear depth, were seen on a sample of valve caps that were currently available and disassembled at the test facility. The wear has been reproduced in more than one shaker table test as well.

ii. A spring with lesser skew appears to cause wear at locations on the lower end of the cap, spread across more than one spring coil. This spreads any lateral force present across multiple locations, resulting in lesser force concentration and lesser amounts of wear. This wear was produced during shaker table testing, and was also identified on cap samples of disassembled valves at the test facility.

iii. Rotation of the cap was identified during shaker table testing. This is allowed per the design of the valve. Different springs and different cap materials resulted in different amounts of rotation during test runs. The cap and spring do not necessarily rotate as a single unit, and therefore may produce differing amounts of wear, or may interact differently with each other depending upon orientation at any particular time.

iv. The failure mechanism appears to be additional force experienced as the spring coil is in contact with either the upper edge or lower edge of the grooved area. As the bellows is extending during

pressure increases, the bellows cap is moving upward in a linear motion. Should the spring be entering the upper edge of the groove, it will produce an upward force on the bellows cap that would not normally be present, resulting in a lower pressure lift of the pilot. Likewise, should the spring be within the groove and contacting the lower edge of the groove, it will produce a downward force on the cap that would not normally be present, resulting in a higher pressure lift on the pilot.

(d) The condition is time dependent, as wear will increase over time. The condition is not dependent upon operating at higher vibration levels due to EPU, as previously understood based on data available at the time, although engineering judgment would indicate that greater levels of vibration may contribute to increased wear. The strongest dependency appears to be on the spring tolerances, especially the straightness of the spring over its length. All springs used for this application fall within the required manufacturer's tolerances, but these tolerances for wire wound springs are broader than tolerances typically associated with machined valve components. This tolerance is not recorded during valve assembly or certification, so the condition of the installed valves cannot be estimated. Engineering judgment indicates that the consequences of the wear (that is, the setpoint of the valve) may increase or decrease at any time as the orientation between the spring and cap changes due to rotation. Should a spring with minimal skew be installed, no wear that can impact setpoint would be expected.

(e) The potential degradation appears to be a result of spring tolerance, and appears to be independent of EPU or Pre-EPU power or vibration levels. No specific plant operating configuration changes can impact the spring, and therefore no restrictions are recommended. Previous EPU power restrictions should be revoked. Ultimately, the safest plant condition will be to install a valve with the improved spring and cap.

	YES	NO
2.3 Is SSC operability supported? Explain basis (e.g., analysis, test, operating experience, engineering judgment, etc.):	[xxx]	[ ]

If 2.3 = NO, notify Operations Shift Management **immediately**.
If 2.3 = YES, clearly document the basis for the determination.

Yes, Operability of the Safety Function of the Target Rock S/RV is supported for both Units 1 and 2. There is currently no indication that the 1% Technical Specification requirement is not being met. Evaluations have shown that safety analyses are met for the current cycle assuming the +6.8% tolerance, which is a bounding value based on performance history. Details are provided in the discussion below, where a number of specific data points are individually introduced, and a conclusion is drawn from the individual pieces of data.

Test Results from Unit 2 S/RV and Dresden S/RV's:

The Unit 2 S/RV removed in Q2R17 had an as-found safety setpoint of 1213 psi, or +6.8% above the required setpoint. Vibration induced wear between the Pressure Adjusting Spring and the Bellows Assembly Cap resulted in a 0.008" groove in the cap in which a coil of the spring was captured, resulting in the high lift setpoint. This is the finding that is driving the Operability Determination for the Unit 1 and 2 S/RV's. More recent data from Dresden as documented in IR 255880 indicates that valves from Pre- and Post-EPU exhibited similar wear to the Quad valve, but had set pressures of –1.4 and –3.6%.

Historical S/RV data for the previous 4 cycles was also reviewed in EC 347434 (Quad) and Operability Determination 200174 (Dresden). Dresden's highest magnitude drift was +2.29%. Quad's highest magnitude drift was –3.52%, while the highest magnitude positive drift was +3.44%. Given these data sets, the +6.8% value is considered bounding for both stations.

Basic average values for all valves at Quad Cities, including S/RV and MSSV's for the previous 4 cycles, was determined to be less than 1% in EC 347434.

Historical Review and Vendor Interviews:

No previous S/RV refurbishments indicated significant wear or degradation to the Cap on pre-EPU valves from Quad Units 1 or 2. Target Rock Field Service personnel indicated that this is the first time that this type of Cap degradation has been identified on this general valve model.

Wear Characteristics Identified With New Data Points and Testing :

The table below summarizes wear characteristics identified on Quad and Dresden valves, as well as some data points from shaker table testing. The data was validated with the inspection of other valves available in a disassembled state at the test facility. This data further indicates that wear does not necessarily result in non-conservative drift.

UNIT	S/N	EPU	Groove Depth	Set Pressure or Test Comments
QDC 2	172	Yes	0.008"	+6.8
DRE 2	121	No	0.008"	-1.4
DRE 2	233	Yes	0.003"	-3.6
Test	172			Original Quad spring with greater skew produced groove at top of cap, first full spring coil. Duplicates Quad and Dresden wear, seen also in portion of random sampling of other valves inspected at test facility.
Test	172			Spring with lesser skew produced markings near bottom of cap, multiple lower spring coils. Duplicates portion of random sampling of other valves inspected at test facility.

Rotation of Cap and Spring During Vibration Testing :

Rotation has been seen during shaker table testing. The design allows for this rotation, and the rotation is acceptable during valve operation. As the rotation changes the orientation between the spring coils and any groove that may have evolved in the cap, the impact on setpoint can change. Contact near the top edge of the groove as the bellows is extending would produce upward force and a low setpoint. Contact near the lower edge of the groove as the bellows is extending would produce downward force and a high setpoint. If there is no interaction between the spring and groove, then the setpoint will not be impacted.

Impact on Appendix R Scenarios :

An Evaluation of the +6.8% S/RV setpoint impact on Appendix R Operator Action Time and Torus Temperature was performed under Risk Management Documentation Support Application SA-1333, which is in the review

process at the time of this writing. Multiple valve reset pressures were considered, and it was found that the high setpoint has insignificant impact on the time to Top of Active Fuel and the peak torus temperature. The report, upon final approval, will conclude that the high lift pressure should not affect the efficacy of Appendix R actions and the Appendix R analysis will remain valid. Setpoints lower than this are considered to be more conservative, as the valve will lift closer to, or prior to, the nameplate setpoint.

Impact on Vessel Overpressure Events :

Vessel overpressure analyses were performed for the previous Unit 2 cycle, in which the S/RV had a tolerance of +6.8%, and an additional MSSV had a tolerance of +2.3%. GE performed a Past Operation Justification under report GE-NE-0000-0028-R0, which is documented under AT 215874-04 under Nuclear Fuels task scoping document NF-B-201. The analysis determined that the ATWS and ASME Overpressure and Fuel Thermal Analyses limits were not exceeded for the previous Unit 2 cycle as a result of these tolerance issues.

ATWS is not impacted by the wear phenomenon in this section of the valve. For the ATWS event, the relief function of the S/RV will cause the valve to open independent of any setpoint drift. The setpoint is only associated with the safety function of the valve.

For the ASME Overpressure Event, analysis using +7% as a conservative S/RV setpoint (as compared to +6.8% for Quad) was evaluated against Dresden data. Dresden data for this analysis bounds Quad data because the Dresden 3 Cycle 18 was determined to be the most limiting with respect to ASME Overpressure margin for the four reactors at the two sites, per GE analysis for ongoing tolerance expansion (GE-NE-0000-0028-6561-R0). Also, the overall tolerance of the Dresden valves resulted in a larger statistical one-sigma value of 1.092% for MSSVs and 1.364% for the S/RV (Calc MWMECH-04-002), where the Quad valves had a one-sigma value of 0.945% for MSSVs and 1.959% for the S/RV (Calc MWMECH-04-003). The MSSVs, being more plentiful (8) and operating at higher pressures, have a much greater statistical and analytical impact than the single S/RV. Because of these factors, the following paragraph concerning Dresden analyses bounds both Quad units concerning ASME Overpressure.

Cantera Engineering performed a Monte Carlo analysis of the Dresden safety valve network including the S/RV at +7% tolerance, see Attachment 2. The analysis used the approved base calculation MWMECH-04-002, then revised the S/RV tolerance, and provided the results in an electronic communication to Dresden. The revised calculation determined that, taking credit for the random variation of tolerance around each valve's nominal setpoint statistically, it could be shown that for pressures sweeping through the range of setpoints, the total integrated flow across all the valves is 12% increased over the 95/95 case, i.e. the MSSVs set at +2.2% tolerance and an S/RV at +4.1% tolerance. A recent NRC submittal from Dresden demonstrated successful ASME overpressure results for a 95% probability with 95% confidence level for a S/RV setpoint tolerance of 4.1% above nominal and MSSV tolerance of 2.2% above nominal for D3C18 crediting the actual control rod patterns throughout the cycle. The Monte Carlo analysis shows that postulating the S/RV at +7% tolerance will have a greater integrated flow rate than the successful ASME overpressure case presented to the NRC.

An alternate method of describing how Quad operability is supported is to apply engineering judgment to alternate analyses completed for the current operating cycles. GE analyses (GE-NE-0000-0026-4682-R0 and GE-NE-0000-0027-0844-R0) in support of previous Operability Determinations showed that overpressure limits were maintained with all safety valve tolerances at +3%. Recent history would indicate that only a few valves will go outside of 1%, and some of these valves will also have tolerances in the negative direction. Furthermore, the S/RV is the lowest set safety valve, with the MSSV's ranging from 9.25% to 11% above the S/RV setpoint. Thus, even with an S/RV lifting at +6.8%, it will be providing flow prior to vessel pressure reaching the range of the MSSV's; with some safeties coming in early while others come in late, the maximum system flow is expected to be reached earlier than if all valves were at +3%. As stated previously, the average pressure of all S/RV's and MSSV's over the previous 4 cycles per EC 347434, is well within a 1% tolerance.

Conclusion:

The setpoint of the S/RVs on both Quad Units are currently subject to a wear phenomenon that may impact the lift setpoint in the high or low direction. The safety lift is fully expected to function despite the setpoint drift. New data and testing to date indicate that the degradation is not specifically related to EPU related vibrations, but is likely the result of spring skew combined with typical main steam line vibrations. (Revision 5) The ATWS event is not impacted because the relief function of the valve is credited, and is unaffected by this wear phenomenon. The ASME Overpressure event for the previous cycle was approved given the +6.8% tolerance for Quad Unit 2. Monte Carlo analyses of the safety valve network using a +7% S/RV value indicates that flows still exceed previously evaluated and approved flow scenarios associated with all valves at the 95/95 statistical setpoint in the positive direction. Evaluations have shown that there were no negative impacts on Appendix R for +6.8%, or on overpressure/fuels analyses with all valves at +3%. A review of S/RV data points for Quad and Dresden for the previous 4 cycles plus the most recent valves indicate that +6.8% is a bounding value.

The tolerance of the springs currently installed in the S/RVs is unknown, so there is no opportunity to estimate the amount of cap wear. However, there is currently no indication that the 1% Technical Specification requirement is not being met. It has been determined that the cap and spring rotate during shaker table testing. Engineering judgment indicates that the orientation of the spring to any potential cap groove may result in interaction that may increase or decrease the setpoint of the valve.

As noted in various locations through this evaluation, the Relief Functions (manual, ADS, or pressure controller actuations) of the S/RV were not impacted, and did not require a review for operability. This information is offered only to clarify that the balance of the S/RV functions remain operable.

	YES	NO
2.4 Are compensatory and/or corrective actions required?	[xxx]	[ ]

If 2.4 = YES, complete section 3.0 (if NO, N/A section 3.0).

Compensatory actions should limit reactor power to a pre-EPU level of 2511MWth. This limitation should be applied to both Unit 1 and Unit 2. (Revision 5)

Corrective actions should include test and/or evaluations into the natural frequency or frequencies associated with the Target Rock S/RV, especially the Pressure Adjusting Spring and Bellows Assembly Cap. Testing on similar equipment was completed utilizing shaker table testing at a seismic test facility. An additional corrective action is to specify, quote, and procure a Bellows Assembly Cap made of a material more resistant to wear than the existing cap. A material of Nitronic 60 has already been discussed with the Vendor. This material should be included in the shaker table testing. A final corrective action will be to submit a modification request as needed to utilize any upgraded parts in the Target Rock Valve replacement for Q1R18 and any future outages. It should be noted that testing to date, as required by the original corrective actions, has evolved from looking strictly at cap materials to investigation of spring tolerance refinement. Although the corrective actions as written do not specify this scope evolution, the current wording will not be revised. Action Tracking Items for these corrective actions will therefore not require revision.

2.5 Reference Documents:

2.5.1 Technical Specifications Section(s):

3.4.3 Safety and Relief Valves
B2.1.2 Reactor Coolant System (RCS) Pressure SL
3.4.10 Reactor Steam Dome Pressure

2.5.2 UFSAR Section(s):

3.1.2.4 Criterion 9 – Reactor Coolant Pressure Boundary
5.2 Overpressure Protection
Ch. 15 Accident and Transient Analyses
15.8 Anticipated Transient Without Scram

2.5.3 Other:

CR 215874, Target Rock S/RV As Found Lift Pressure High

RCI 215874, Target Rock S/RV High Lift Pressure (In Progress)
EC 346515, Evaluation Of Vibration Issues And Items Required For Full EPU Power Operation Of Quad Cities Unit 1, Revision 002
EC 348316, Evaluation Of MSL & Feedwater Component Vibration Effects...Quad Cites Unit 2, Revisions 000 And 001
Quality Receipt Inspection Package 131231, for Target Rock S/RV Serial Number 171, Including Wyle Labs Report 48744 and Target Rock Field Service Project 03Z037 (Test and Refurb of S/RV From Q1M16)
Wyle Labs Report 48744 of 4/19/04, As Found Testing of Serial Number 172 (Test of S/RV from Q2R17)
Interviews and Observations of Wyle and Target Rock Field Service During Serial Number 172 Disassembly, 5/11/04 to 5/12/04, Wyle Labs
Quad Cities Units 1 and 2 Target Rock Inlet Vibration Data, Summarized by Sharon Eldridge, Cantera, Electronic Copy
CR 200772, Main Steam Safety and Safety Relief Valve Tolerance
Point History review of Unit Gross Generation for Quad 1 and 2 to estimate number of days of operation at EPU power levels
Quad Cities Units 1 and 2 Fire Protection Reports, Volume 2, Appendix R Conformance (Sections III.G, III.J, and III.L) Safe Shutdown Report, 10 CFR 50.48, 10 CFR 50 Appendix R
Risk Management Document SA-1333, QC-2 Target-Rock SRV Lift Pressure High – MAAP Analysis of App. R Transient Scenario, in review process at time of writing
GE-NE-0000-0026-4682-R0, Licensing Analysis for Dresden and Quad Cities Safety and Relief Valve Setpoint Tolerance Increase (Phase 1A)
GE-NE-0000-0027-0844-R0, Licensing Analysis for Dresden and Quad Cities Safety and Relief Valve Setpoint Tolerance Increase (Phase 1B)
GE-NE-0000-0028-6556-R0, Evaluation of Out-of Tolerance Safety Valve Setpoints On Quad Cities 2 Cycle 1 Licensing Analysis
GE-NE-0000-0028-6561-R0, Licensing Analysis for Dresden and Quad Cities Safety and Releif Valve Setpoint Tolerance Increase (Phase 2A)
MWMECH-04-002, Monte Carlo Analysis of Dresden 2&3 MSSV Network
MWMECH-04-003, Monte Carlo Analysis of QC 1&2 MSSV Network
IR 255880 and Operability Determination, (Dresden) Target Rock Valve Degradation Found During Valve Rebuild
C0025 VETIP Binder, Target Rock Technical Manual, Safety/Relief Valve Model 67F
ATI 200174, (Dresden) Operability Evaluation, Main Steam Safety Valves...and Target Rock Safety/Relief Valve...on Each Unit
EC 347434, Quad Cities Statistical Analysis of Main Steam Safety Valve Setpoint Variations Based on As-Found Setpoint Data

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3.0 ACTION ITEM LIST:

If, through evaluating SSC operability, it is determined that the degraded or nonconforming SSC does <u>not</u> prevent accomplishment of the specified safety function(s) in the TS or UFSAR and the intention is to continue operating the plant in that condition, then record below, as appropriate, any required compensatory actions to support operability and/or corrective actions required to restore full qualification. For corrective actions, document when the actions should be completed (e.g., immediate, within next 13 week period, next outage, etc.) and the basis for timeliness of the action. Corrective action timeframes longer than the next refueling outage are to be explicitly justified as part of the OpEval or deficiency tracking documentation being used to perform the corrective action.

Compensatory Action #1: Establish controls to limit Unit 1 reactor power to 2511MWth for any significant duration. Prior to raising power on Unit 1 above 2511MWth, contact Engineering to determine any power level restrictions that may be needed, and revise this Operability Evaluation if required. Short duration power increases above 2511 MWth for data acquisition purposes is allowed (<=72hrs). (Revision 5)

Responsible Dept./Supv.: A8410OP/A Scott

Action Due: 11/16/04

Action Tracking #: 220863- 10

Compensatory Action #2: Establish controls to limit Unit 2 reactor power to 2511MWth. Prior to raising power on Unit 2 above 2511MWth, contact Engineering to determine any power level restrictions that may be needed, and revise this Operability Evaluation if required. Short duration power increases above 2511 MWth for data acquisition purposes is allowed (<=72hrs). (Revision 5)

Responsible Dept./Supv.: A8410OP/A Scott

Action Due: 11/16/04

Action Tracking #: 220863-11

Corrective Action #1: Specify, Submit RFQ, and Procure a Bellows Assembly Cap in wear resistant material Nitronic 60 or similar for use in shaker table test of Target Rock S/RV.

Responsible Dept./Supv.: A8430TP/Boline

Action Due: Complete

Basis for timeliness of action: Parts required for shaker table testing have a 5 week lead time. Initial estimates of shaker table testing are 60 days out (see CA #2). This order date supports the test dates.

Action Tracking #: 220863-12

Corrective Action #2: Perform shaker table test and any supporting structural or vibration analysis of the Target Rock S/RV. Testing should include identification of natural frequencies and determine correlation to existing frequencies at Quad Units 1 and 2. Aging runs at appropriate frequencies should be performed to determine wear characteristics of affected components within the valve, especially those associated with the adjustment spring and bellows cap assembly. If necessary, the testing may include only the pilot sections of the valve; this may be necessary due to radiological controls at the test facility.

Responsible Dept./Supv.: A8064MW-DR/Eldridge

Action Due: 12/03/04 (Revision 5)

Basis for timeliness of action: 60 days is the initial estimate for arranging testing and evaluation. An additional margin of approximately 14 days has been added to allow for analysis and documentation of test results. Extended due to testing durations and test facility availability. (Revision 2) Extended due to test facility availability windows, and testing of additional materials or configurations. (Revision 3) Extended to allow comprehensive documentation of test results to be completed. (Revision 5)

Action Tracking #: 220863-13

Corrective Action #3: Submit Modification Request for any S/RV parts upgrades determined necessary during the shaker table tests.

Responsible Dept./Supv.: A8430TP/Boline

Action Due: 12/09/04 (Revision 5)

Basis for timeliness of action: Allows time for test results to be examined to determine if modification is required. Allows time for scheduling of mod requests as needed. Extended due to testing durations and test facility availability. (Revision 2) Extended due to test facility availability windows, and testing of additional materials or configurations. (Revision 3) Extended to allow comprehensive documentation of test results to be completed. (Revision 5)

Action Tracking #: 220863-14

Corrective Action #4: Implement modification to S/RV in Unit 1 in Q1R18.

Responsible Dept./Supv.: A8430TP/Boline

Action Due: 4/5/05

Basis for timeliness of action: This outage is the next outage of sufficient duration in which the S/RV can be replaced..

Action Tracking #: 220863-15

Corrective Action #5: Implement modification to S/RV in Unit 2 in Q2R18

Responsible Dept./Supv.: A8430TP/Boline

Action Due: 4/15/06

Basis for timeliness of action: This outage is the next outage of sufficient duration in which the S/RV can be replaced.

Action Tracking #: 220863-16

4.0 SIGNATURES:

4.1 Preparer(s)_____ Date _____

_____ Date _____

4.2 Reviewer_____ Date _____
(10 CFR 50.59 screener qualified or active SRO license holder)

4.3 Sr. Manager Design Engg/Designee Concurrence _____ Date _____

4.4 Operations Shift Management Approval _____ Date _____

4.5 Ensure the completed form is forwarded to the OEPM for processing and Action Tracking entry as appropriate.

5.0 OPERABILITY EVALUATION CLOSURE:

5.1 Corrective actions are complete, as necessary, and the OpEval is ready for closure

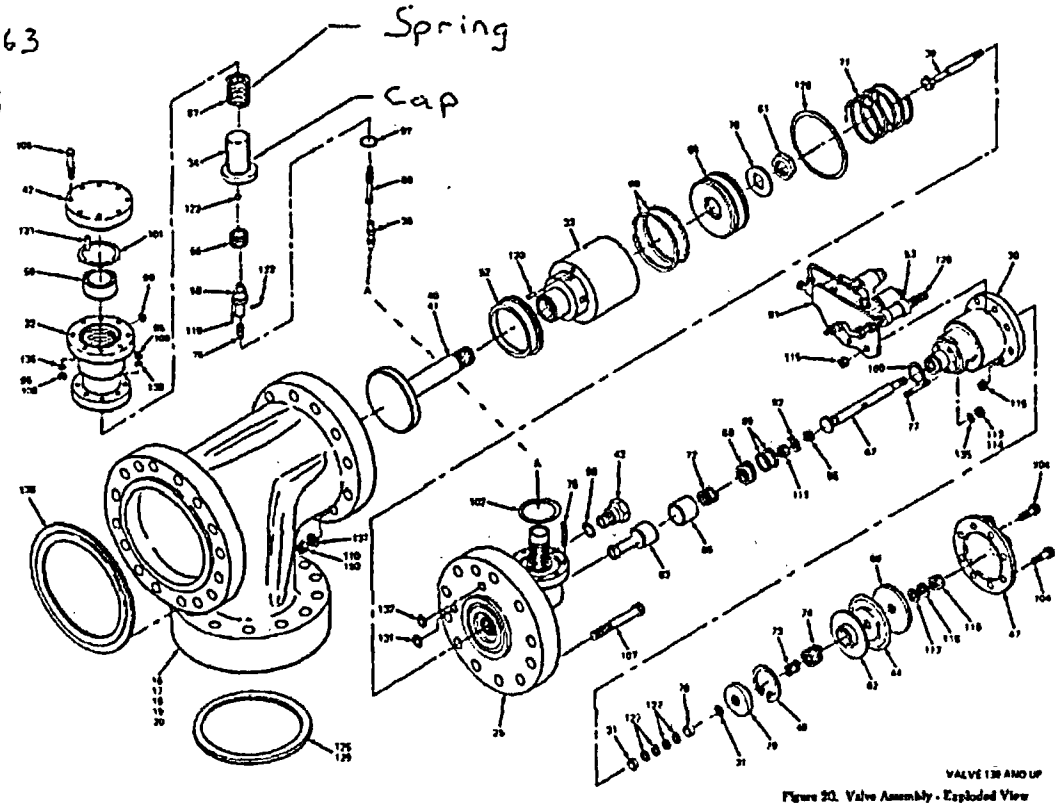
_____ Date _____
(OEPM)

5.2 Operations Shift Management Approval _____ Date _____

5.3 Ensure the completed form is forwarded to the OEPM for processing, Action Tracking entry, and cancellation of any open compensatory actions, as appropriate.

Op Det 220863
Att 1 Pg 1 of 3

SRV Exploded View



VALVE 130 AND UP

Figure 20. Valve Assembly - Exploded View (S/N 130 and up)

7-21/(7-22 blank)

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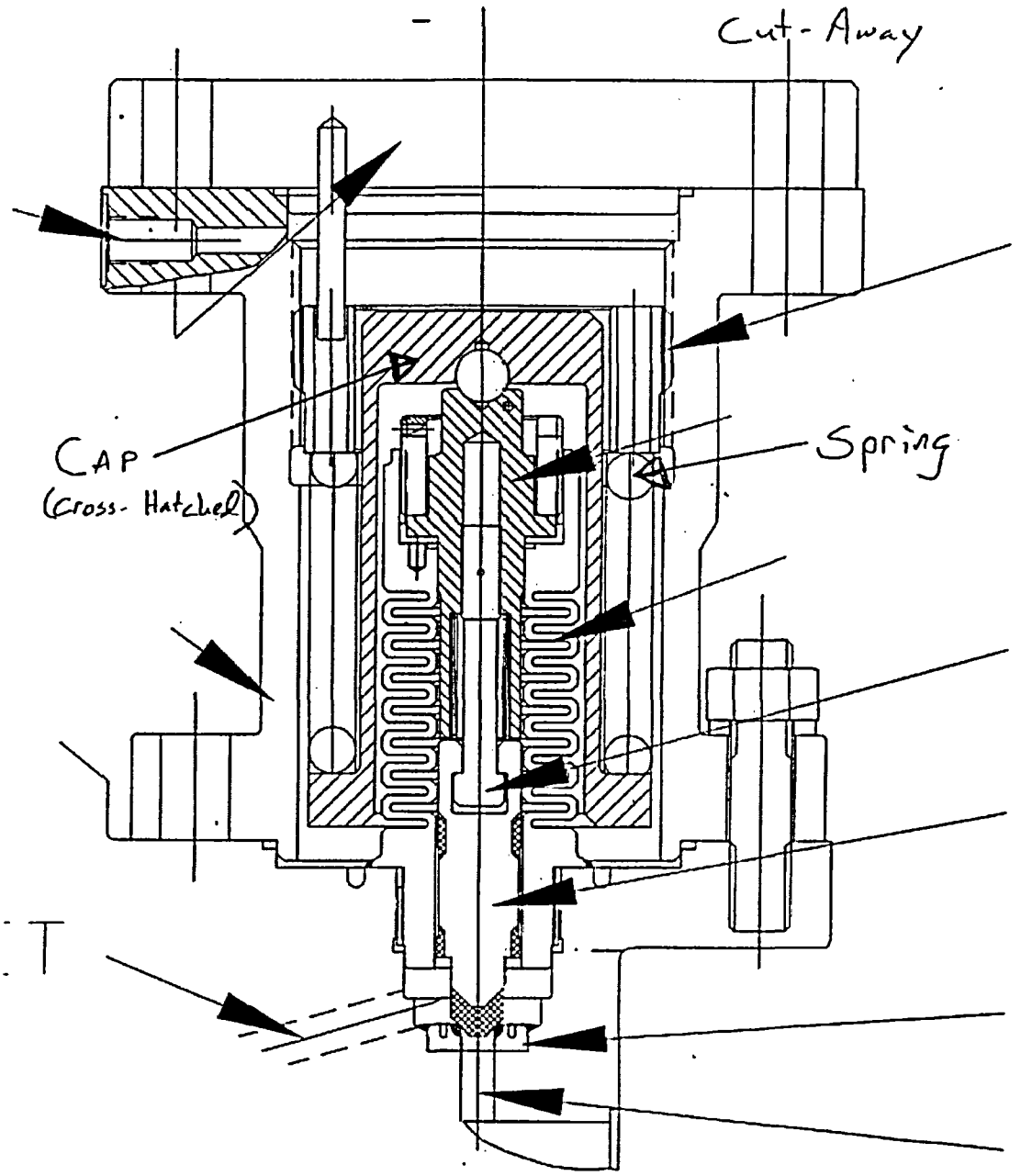
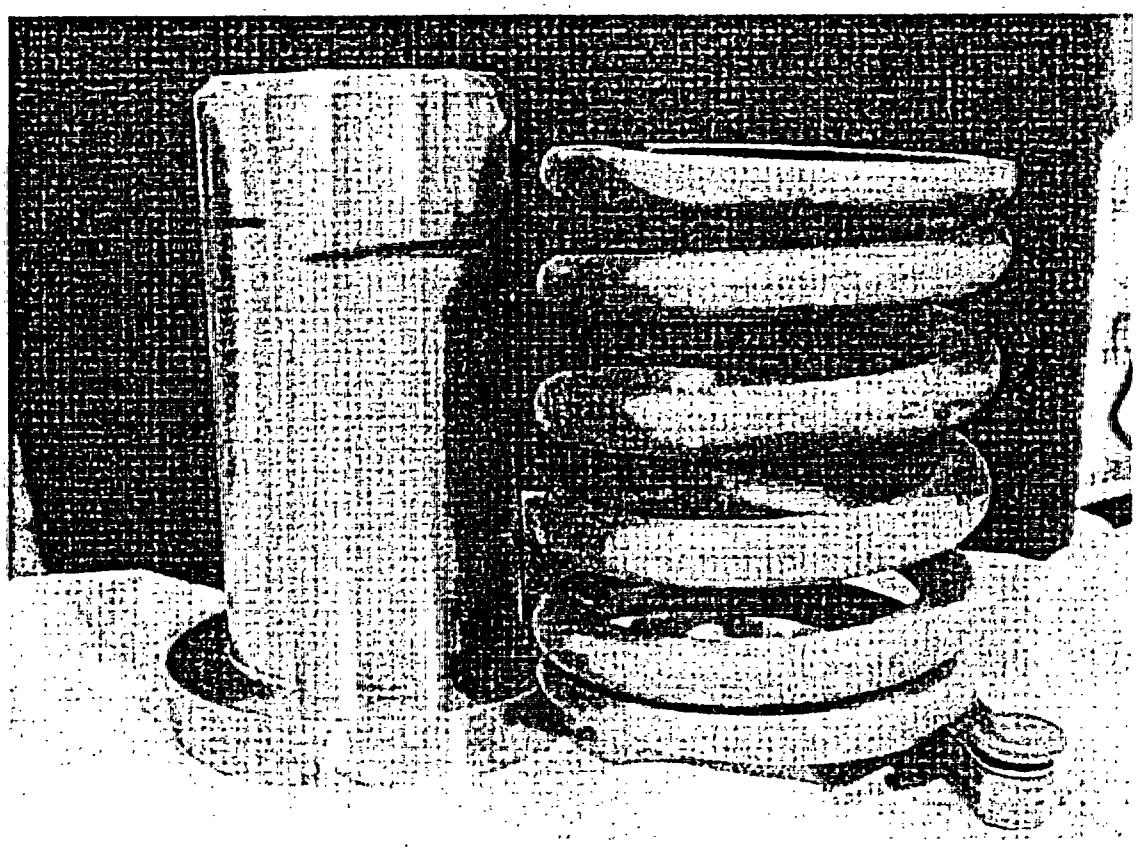
Att 1 Pg 2 of 3

1st Stage
SRV
Cut-Away



Op Det 220863.
Att 1 Pg 3 of 3



S/N 172
Cap and Spring
Unit 2, Removed Q2R17

Op Doc 220863
Att 2 Pg 1 of 8

Prediction of Safety Valve Performance

This worksheet develops a mathematical relationship for valve flow vs pressure to allow direct comparsion of the effects of setpoint changes on predicted network flow performance. This sheet is identical to Appendix C of MWMECH-04-002 with the exception that for the Monte Carlo summation, a 7% drift is mechanistically assumed for the Target Rock valve.

$p := 1150, 1150.5 .. 1375$ defines a pressure range variable

$Q_{mssv}(p) := 644500 \cdot \frac{p}{1255}$ main steam safety valve flow as a function of pressure and rated conditions, lbm/hr

$Q_{TR}(p) := 598000 \cdot \frac{p}{1095}$ Target Rock safety valve flow as a function of pressure and rated conditions, lbm/hr

the following define the valve opening setpoints for a 1% drift basis

$SP1 := 1161.5$ Target Rock valve

$SP2 := 1267.5$ Gr 1 MSSVs

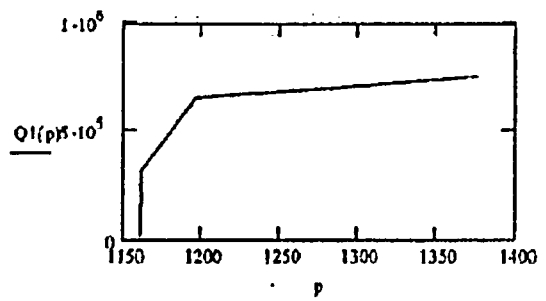
$SP3 := 1277.7$ Gr 2 MSSVs

$SP4 := 1287.8$ Gr 3 MSSVs

the following relation defines a stepwise continuous function representing the valve. It allows 50% flow at set pressure, and a linear ramp to full flow at 103% of set pressure.

$$Q1(p) := \begin{cases} 0 & \text{if } p < SP1 \\ \min\left[Q_{TR}(p) \cdot \frac{Q_{TR}(p)}{2} + \frac{Q_{TR}(p)}{2} \cdot \left(\frac{p - SP1}{.03 \cdot SP1}\right)\right] & \text{otherwise} \end{cases}$$

the following plot demonstrates the flow vs pressure being defined for the T/R valve



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the following perform the same definitions for the 3 groups of MSSVs

Gr1 MSSVs

$$Q2(p) := \begin{cases} 0 & \text{if } p < SP2 \\ \min\left[Q_{mssv}(p), \frac{Q_{mssv}(p)}{2} + \frac{Q_{mssv}(p)}{2}\cdot\left(\frac{p - SP2}{.03\cdot SP2}\right)\right] & \text{otherwise} \end{cases}$$

Gr2 MSSVs

$$Q3(p) := \begin{cases} 0 & \text{if } p < SP3 \\ \min\left[Q_{mssv}(p), \frac{Q_{mssv}(p)}{2} + \frac{Q_{mssv}(p)}{2}\cdot\left(\frac{p - SP3}{.03\cdot SP3}\right)\right] & \text{otherwise} \end{cases}$$

Gr3 MSSVs

$$Q4(p) := \begin{cases} 0 & \text{if } p < SP4 \\ \min\left[Q_{mssv}(p), \frac{Q_{mssv}(p)}{2} + \frac{Q_{mssv}(p)}{2}\cdot\left(\frac{p - SP4}{.03\cdot SP4}\right)\right] & \text{otherwise} \end{cases}$$

sum the total flow vs pressure. Note that the number of valves in each group is used as a multiplier

$$Q1tot(p) := Q1(p) + 2\cdot Q2(p) + 2\,Q3(p) + 4\,Q4(p)$$

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3% drift case
the identical approach is applied for the 3% drift case

$SP1 := 1184.5$

$SP2 := 1292.65$

$SP3 := 1302.95$

$SP4 := 1313.25$

$$Q13(p) := \begin{cases} 0 & \text{if } p < SP1 \\ \min\left[Q_{TR}(p), \frac{Q_{TR}(p)}{2} + \frac{Q_{TR}(p)}{2}\cdot\left(\frac{p - SP1}{.03\cdot SP1}\right)\right] & \text{otherwise} \end{cases}$$

$$Q23(p) := \begin{cases} 0 & \text{if } p < SP2 \\ \min\left[Q_{mssv}(p), \frac{Q_{mssv}(p)}{2} + \frac{Q_{mssv}(p)}{2}\cdot\left(\frac{p - SP2}{.03\cdot SP2}\right)\right] & \text{otherwise} \end{cases}$$

$$Q33(p) := \begin{cases} 0 & \text{if } p < SP3 \\ \min\left[Q_{mssv}(p), \frac{Q_{mssv}(p)}{2} + \frac{Q_{mssv}(p)}{2}\cdot\left(\frac{p - SP3}{.03\cdot SP3}\right)\right] & \text{otherwise} \end{cases}$$

$$Q43(p) := \begin{cases} 0 & \text{if } p < SP4 \\ \min\left[Q_{mssv}(p), \frac{Q_{mssv}(p)}{2} + \frac{Q_{mssv}(p)}{2}\cdot\left(\frac{p - SP4}{.03\cdot SP4}\right)\right] & \text{otherwise} \end{cases}$$

$$Q3tot(p) := Q13(p) + 2\cdot Q23(p) + 2\,Q33(p) + 4\,Q43(p)$$

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95-95 case

Perform a flow vs pressure calc for 95-95 setpoints. These are 4.03% for the T/R and 2.14% for the MSSVs

$SP195 := 1196.3$

$SP295 := 1281.8$

$SP395 := 1292.1$

$SP495 := 1302.3$

$$Q195(p) := \begin{cases} 0 & \text{if } p < SP195 \\ \min\left[Q_{TR}(p), \frac{Q_{TR}(p)}{2} + \frac{Q_{TR}(p)}{2}\cdot\left(\frac{p - SP195}{.03\cdot SP195}\right)\right] & \text{otherwise} \end{cases}$$

$$Q295(p) := \begin{cases} 0 & \text{if } p < SP295 \\ \min\left[Q_{mssv}(p), \frac{Q_{mssv}(p)}{2} + \frac{Q_{mssv}(p)}{2}\cdot\left(\frac{p - SP295}{.03\cdot SP295}\right)\right] & \text{otherwise} \end{cases}$$

$$Q395(p) := \begin{cases} 0 & \text{if } p < SP395 \\ \min\left[Q_{mssv}(p), \frac{Q_{mssv}(p)}{2} + \frac{Q_{mssv}(p)}{2}\cdot\left(\frac{p - SP395}{.03\cdot SP395}\right)\right] & \text{otherwise} \end{cases}$$

$$Q495(p) := \begin{cases} 0 & \text{if } p < SP495 \\ \min\left[Q_{mssv}(p), \frac{Q_{mssv}(p)}{2} + \frac{Q_{mssv}(p)}{2}\cdot\left(\frac{p - SP495}{.03\cdot SP495}\right)\right] & \text{otherwise} \end{cases}$$

$$Q95tot(p) := Q195(p) + 2\cdot Q295(p) + 2\cdot Q395(p) + 4\cdot Q495(p)$$

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Monte Carlo 95th percentile results

Perform a flow vs pressure calc for Monte Carlo based setpoints. Each valve is accounted for individually based on the 95th percentile setpoint calculated by PLANETS. The overall procedure is identical to that above.

$SP1mc := 1175.9$ T/R

$SP1mc := 1.07 \cdot 1150$ **Assume mechanistic 7% T/R drift**

$SP2mc := 1258.2$ 1st MSSV

$SP3mc := 1265$ 2nd MSSV

$SP4mc := 1270.6$ 3rd MSSV

$SP5mc := 1275.8$ 4th MSSV

$SP6mc := 1281$ 5th MSSV

$SP7mc := 1286.8$ 6th MSSV

$SP8mc := 1294.2$ 7th MSSV

$SP9mc := 1306.8$ 8th MSSV

$$Q1mc(p) := \begin{cases} 0 & \text{if } p < SP1mc \\ \min\left[Q_{TR}(p), \frac{Q_{TR}(p)}{2} + \frac{Q_{TR}(p)}{2} \cdot \left(\frac{p - SP1mc}{.03 \cdot SP1mc}\right)\right] & \text{otherwise} \end{cases}$$

$$Q2mc(p) := \begin{cases} 0 & \text{if } p < SP2mc \\ \min\left[Q_{mssv}(p), \frac{Q_{mssv}(p)}{2} + \frac{Q_{mssv}(p)}{2} \cdot \left(\frac{p - SP2mc}{.03 \cdot SP2mc}\right)\right] & \text{otherwise} \end{cases}$$

$$Q3mc(p) := \begin{cases} 0 & \text{if } p < SP3mc \\ \min\left[Q_{mssv}(p), \frac{Q_{mssv}(p)}{2} + \frac{Q_{mssv}(p)}{2} \cdot \left(\frac{p - SP3mc}{.03 \cdot SP3mc}\right)\right] & \text{otherwise} \end{cases}$$

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$$Q4mc(p) := \begin{cases} 0 & \text{if } p < SP4mc \\ \min\left[Q_{mssv}(p), \frac{Q_{mssv}(p)}{2} + \frac{Q_{mssv}(p)}{2}\cdot\left(\frac{p - SP4mc}{.03\cdot SP4mc}\right)\right] & \text{otherwise} \end{cases}$$

$$Q5mc(p) := \begin{cases} 0 & \text{if } p < SP5mc \\ \min\left[Q_{mssv}(p), \frac{Q_{mssv}(p)}{2} + \frac{Q_{mssv}(p)}{2}\cdot\left(\frac{p - SP5mc}{.03\cdot SP5mc}\right)\right] & \text{otherwise} \end{cases}$$

$$Q6mc(p) := \begin{cases} 0 & \text{if } p < SP6mc \\ \min\left[Q_{mssv}(p), \frac{Q_{mssv}(p)}{2} + \frac{Q_{mssv}(p)}{2}\cdot\left(\frac{p - SP6mc}{.03\cdot SP6mc}\right)\right] & \text{otherwise} \end{cases}$$

$$Q7mc(p) := \begin{cases} 0 & \text{if } p < SP7mc \\ \min\left[Q_{mssv}(p), \frac{Q_{mssv}(p)}{2} + \frac{Q_{mssv}(p)}{2}\cdot\left(\frac{p - SP7mc}{.03\cdot SP7mc}\right)\right] & \text{otherwise} \end{cases}$$

$$Q8mc(p) := \begin{cases} 0 & \text{if } p < SP8mc \\ \min\left[Q_{mssv}(p), \frac{Q_{mssv}(p)}{2} + \frac{Q_{mssv}(p)}{2}\cdot\left(\frac{p - SP8mc}{.03\cdot SP8mc}\right)\right] & \text{otherwise} \end{cases}$$

$$Q9mc(p) := \begin{cases} 0 & \text{if } p < SP9mc \\ \min\left[Q_{mssv}(p), \frac{Q_{mssv}(p)}{2} + \frac{Q_{mssv}(p)}{2}\cdot\left(\frac{p - SP9mc}{.03\cdot SP9mc}\right)\right] & \text{otherwise} \end{cases}$$

$$Qmctot(p) := (Q1mc(p) + Q2mc(p) + Q3mc(p) + Q4mc(p) + Q5mc(p) + Q6mc(p)) \ldots + Q7mc(p) + Q8mc(p) + Q9mc(p)$$

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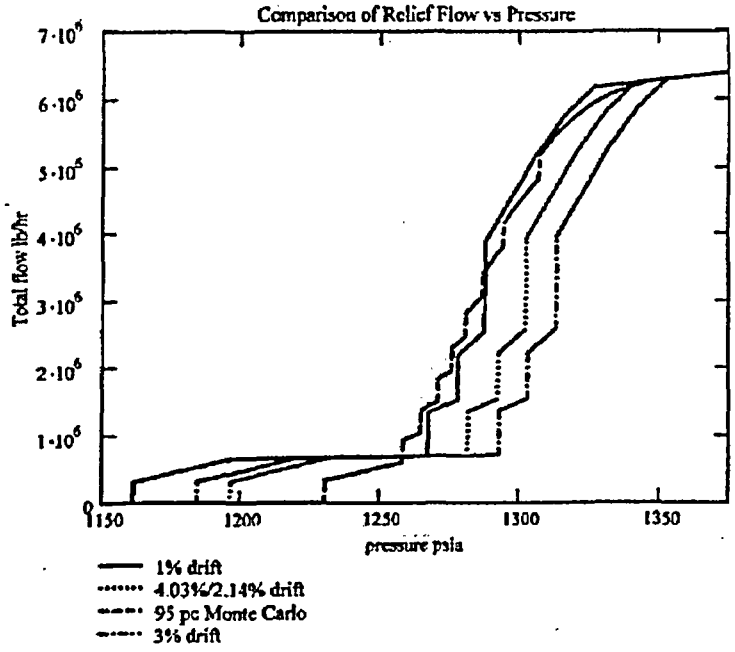

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Integrating the flow pressure curves

Integrating the flow vs pressure functions and converting to lb/sec yields

$$M_1 := \frac{\int_{1150}^{1350} Q1tot(p)\,dp}{3600} \qquad M_1 = 1.249 \times 10^5$$

$$M_{mc} := \frac{\int_{1150}^{1350} Qmctot(p)\,dp}{3600} \qquad M_{mc} = 1.127 \times 10^5$$

$$M_{9595} := \frac{\int_{1150}^{1350} Q95tot(p)\,dp}{3600} \qquad M_{9595} = 9.708 \times 10^4$$

$$M_3 := \frac{\int_{1150}^{1350} Q3tot(p)\,dp}{3600} \qquad M_3 = 8.286 \times 10^4$$

the ratios between the base 1% case and the others is generated

$\frac{M_{mc}}{M_1} = 0.902$ **Monte Carlo, with T/R at 7% assumed drift, this number compares to 98.3 % for the base calculation**

$\frac{M_{9595}}{M_1} = 0.777$ 95-95 setpoint

$\frac{M_3}{M_1} = 0.663$ 3% drift

ATTACHMENT 5

Feedwater Sample Probes – Root Cause Report and Operability Evaluation for Dresden Nuclear Power Station

5A) Dresden Nuclear Power Station, Units 2 and 3, Root Cause Report, "Feedwater Sample Probe Failure Resulting in Reactor Pressure Vessel Foreign Material Intrusion Due to Feedwater Sample Probe Design Deficiency"

5B) Dresden Nuclear Power Station, Units 2 and 3, Operability Evaluation 03-015, Revision 2, "Feedwater Sparger"

ATTACHMENT 5A

Dresden Nuclear Power Station, Units 2 and 3
Root Cause Report
"Feedwater Sample Probe Failure Resulting in Reactor Pressure Vessel Foreign Material Intrusion Due to Feedwater Sample Probe Design Deficiency"

ATTACHMENT 13
Root Cause Investigation Report Content and Format

ROOT CAUSE REPORT

1.0 **Title:** U2 and U3 Feedwater Sample Probe Failure Resulting in Reactor Pressure Vessel Foreign Material Intrusion due to Feedwater Sample Probe Design Deficiency

2.0 **Station/Unit:** Dresden / U2 and U3

3.0 **Event Date:** 10/29/03
Event Time: 1200 hours

4.0 **Action Tracking Item Number:** 183901-18
Report Date: 12/17/03

5.0 **Lead RCR Investigator:** L. Dyas - Programs Engineering

Sponsoring Managers: G. Dorsey - Chemistry Department Head
A. Shahkarami - Site Engineering Director

Team Members: R. Testin - Programs Engineering, RC Qualified
F. Polak - Plant Engineering, RC Qualified
A. McMartin - Programs Engineering
B. Geier - Programs Engineering
G. Baxa - Plant Engineering
D. Malauskas - Chemistry

6.0 Executive Summary

During the Dresden Unit 2 refueling outage D2R18 feedwater sparger inspection in October 2003, three holes were identified in the 240°, N4C, feedwater sparger. An inspection of the sparger revealed a feedwater isokinetic sample probe, resting in the sparger and appearing to have caused the sparger damage. The sample probe, assumed to be the original feedwater sample probe that had been reported missing and had been replaced in the most recent outage (D2R17) in 2001, was removed and the sparger was repaired.

As a result of the sparger damage and the discovery of the sample probe in Unit 2, plans were made to conduct external and internal inspections of all Unit 3 feedwater spargers at the next outage in order to locate a feedwater sample probe reported missing during the 2002 Unit 3 refueling outage (D3R17). During an emergent Unit 3 outage (D3M10) for steam dryer maintenance in December 2003, the planned sparger inspection was conducted. No sparger damage was noted during the external and internal inspections but two feedwater probes were discovered resting in the N4B, 150°, feedwater sparger. As a result of the sparger damage in Unit 2, and the discovery of failed sample probes in both Units, an investigation into the root cause of the sparger damage and probe failure was initiated. A summary of the results of this investigation is provided below along with a determination of the root cause.

In 1971, isokinetic sample probes, were installed in both units at the condensate pump discharge, the condensate demineralizer effluent and the feedwater sample points to support an expanded water chemistry program to evaluate demineralizer performance and feedwater corrosion product levels. The probes were designed and provided by GE. The design called for type 316 schedule XXS, however, the probes believed to be original were made from schedule 160.

In 1977, GE issued SIL 257, Improved Feedwater Sample Probe, after two Boiling Water Reactors (BWRs) reported feedwater probes broke off and lodged in downstream valves. The mechanism for failure was determined to be transgranular stress cracking corrosion (TGSCC) in the crevice between the collar and probe. The SIL recommended replacing the probe with a design that had a seal weld to protect the crevice. In 1990, GE issued SIL 518, Improved Recirculation Water Chemistry Sample Probe, to address a failure of a recirculation sample probe. The failure mechanism was identified as high cycle fatigue caused by flow induced vibration. There is no record of Dresden's disposition of, or action taken on, either SIL between 1977 and 2001.

A recent review of OPEX, and a poll of Exelon Stations and General Electric (GE), indicated between 1990 and 2001, sample probe failures occurred at four nuclear stations (Perry, Braidwood, Browns Ferry, and Grand Gulf). In each of these cases, the mode of failure was fatigue. In 1996 and 1997, Quad Cities replaced both feedwater probes in accordance with SIL 257, and recent e-mail correspondence indicates that both probes exhibited signs of failure near the pipe attachment when removed and inspected. Until the root cause investigation in 2003, Dresden was not aware of any of the operating experiences that resulted in fatigue failure.

In response to an evaluation of INPO SEN 204, Water Chemistry Induced Fuel Leaks, Dresden became aware of SIL 257, prompting the inspection of Unit 2 and Unit 3 feedwater probes in 2001 (D2R17) and 2002 (D3R17), respectively. In both cases, the feedwater probes were found missing and replaced with SIL 257 probes. Also, Nuclear Fuels Evaluations were completed and found the most likely location of the missing probe was a feedwater sparger. Each evaluation, NFM-MW:01-0351 and NFM-MW:02-041, concluded that there were no safety concerns associated with the operation of either unit with the feedwater probe missing.

Additionally, the condensate system probes were not considered for replacement because Dresden engineers and chemists were unaware of their existence.

In October 2003, in addition to the Unit 2 sparger event, the Unit 2 Condensate Demineralizer Effluent (CDE) probe was discovered in 2D Condensate Booster Pump casing. NFM Engineering Condition Report (EC) 345441 concluded the potential of missing parts of the CDE probe had a moderate risk of causing fuel fretting, and there were no fuel or control rod safety concerns associated with Unit 2 operation.

In December 2003, the Unit 3 spargers were internally inspected, discovering two probes. Also, ultra-sonic testing indicated that the Unit 2 feedwater probe installed in D2R17 was missing. Nuclear Fuels' analysis of the missing Unit 2 sample probe (NF document, ECN 345672, Dresden Unit 2 – Lost Parts Evaluation for Isokinetic Probe and Sparger Disks) and GE analysis (GE-NE-0000-0023-7311-RO, Lost Parts Analysis for Dresden Generating Station Units 2 and 3 Feedwater Sample Probes) are consistent and conclude that safe reactor operation will not be compromised with the presence of the potential of lost parts in the reactor vessel.

Exelon PowerLabs inspected all of the probes recovered and concluded the two sample probes recovered from the Unit 3 sparger both failed as a result of fatigue cracking, most likely due to high cycle/low stress loading. GE evaluated potential failure mechanisms for the two most recently installed feedwater probes and determined the most likely mode of failure was mechanical high cycle fatigue induced by flow vibrations. GE calculated the first natural frequency of both probes retrieved from the Unit 3 sparger under the conditions that they would have experienced during service. . The natural frequency of the SIL 257 probe was also evaluated at pre-EPU flowrates. The SIL 257 probe would not have been locked in to the Vortex Shedding Frequency at the pre-EPU flowrates. No analysis was performed to determine the affects of increased flow on the SIL 257 probes prior to installation.

The **Root Cause** of the U2 and U3 Feedwater sample probe failures and subsequent U2 Feedwater Sparger damage was attributed to a Feedwater sample probe design deficiency. The original U2 and U3 Feedwater sample probes installed per GE Drawing 921D233, Sample Probe for Feedwater Control System, Revision 1, dated 04/26/68, were susceptible to the failure mechanisms of Transgranular Stress Corrosion Cracking (TGSCC) and fatigue failure resulting from flow-induced vibration. The modified U2 and U3 Feedwater sample probes installed per the design specified in the GE Service Information Letter (SIL) 257 during D2R17 and D3R17 are susceptible to a fatigue failure mechanism resulting from flow-induced vibration.

The Corrective Actions to Prevent Recurrence (CAPRs) are to install sample probes in the U2 and U3 Feedwater systems tolerant of the TGSCC and fatigue failure mechanisms.

Contributing Causes were attributed to:

- Dresden's failure to effectively disposition GE SILs related to sample probe failures.
- A lack of ownership involving sample systems, specifically sample point locations, sample device drawings, and sampling methods.
- Historical inadequate design documentation for the Feedwater, Condensate Demineralizer Effluent and Condensate Pump Discharge sample probe type and location.
- A historical lack of communication between stations and ineffective use of the available OPEX information.

- Ineffective corrective actions associated with CRs 81081 and 127346, which were generated for the missing Feedwater probes identified in D2R17 and D3R17 respectively.
- A lack of analysis to address the affects of changing the system flow velocity while at the same time changing the effective probe length when the SIL 257 probes were installed in D2R17 and D2R18.

The **Extent of Condition** includes the six isokinetic probes originally installed in the U2 and U3 Condensate and Feedwater Systems due to their susceptibility to stress corrosion cracking and fatigue failure as identified in SIL 257 and SIL 518 respectively. Of these six probe locations, only two, the Unit 2 and Unit 3 Condensate Demineralizer Effluent (CDE) probes, remain installed. Both are recent installations of SIL 257 designs. Also, other sample probes in both units and common location were reviewed, determining that none were susceptible to the same failure mechanisms and only one was able to enter the reactor internals if it failed.

The **safety significance** of this event was minimal.

7.0 Condition Statement

This Root Cause Investigation will determine the causal factors associated with the Unit 2 Feedwater sample probe failure and subsequent Feedwater sparger damage. This will include the failure mechanism of the Feedwater sample probe and the failure mechanism of the damage to the Unit 2 Feedwater sparger. The investigation will also evaluate the Extent of Condition, initiate the Corrective Action to Prevent Recurrence (CAPR) and identify Corrective Actions (CAs). An extensive review of industry experience, OPEX and maintenance history will be performed as well. Emphasis will be placed on foreign material not retrieved and existing system sample probes.

In addition, the RCR addressed the issues identified in the following CRs:

- CR 187258, Unit 2 and Unit 3 Isokinetic Sample Probes Missing in Condensate and FW Systems
- CR 187492, Unit 2 Non-Conservative Analysis and Ineffective Corrective Actions
- CR 189800, Foreign Material in 150-Degree Unit 3 Feedwater Sparger
- CR 189992, Unit 3 Final Feedwater Isokinetic Probe Failed
- CR 190413, D2 FW Sample Probe Missing
- CR 190571, D2 FW Sample Probe Missing

8.0 Event Description

The following discussion provides a summary of the event milestones and analysis.

8.1 Event Summary - Historical Events

03/25/71: In response to a GE presentation on expanded water chemistry programs, Dresden installed three sample probes on each unit in the Condensate and Feedwater systems under work order FDCN D2-H-12 (2M 571). A total of six probes were installed, three on each unit at the condensate pump discharge, the condensate demineralizer effluent and the feedwater sample points, as shown in Figure 1. The probes purpose was to extract representative samples from the process fluid stream to evaluate condensate demineralizer performance and feedwater corrosion product levels.

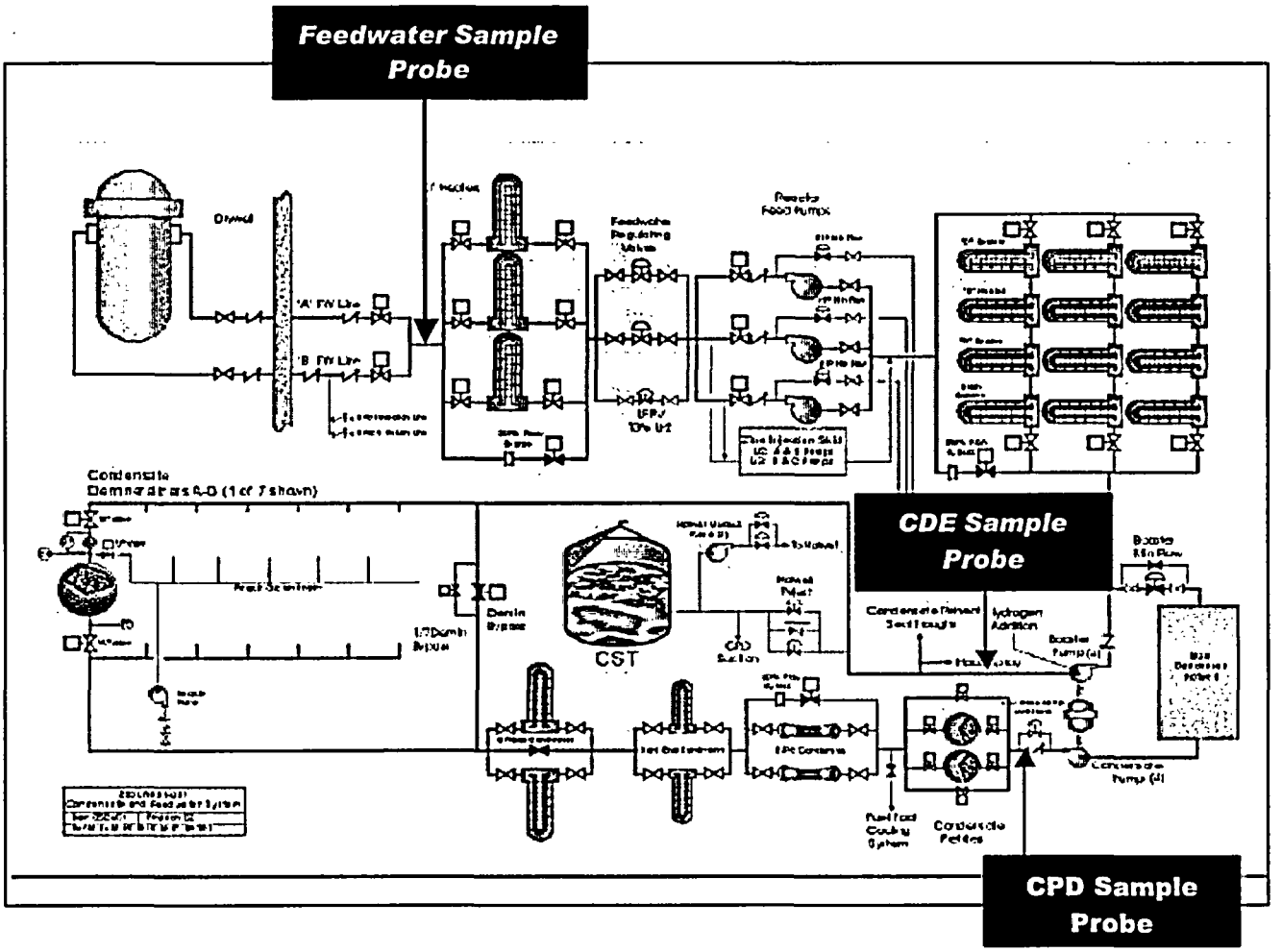

Figure 1: Condensate and Feedwater Sample Probe Locations

12/30/77: GE SIL 257, Improved Feedwater Sample Probe, was issued after two operating BWRs reported feedwater probes failed and lodged in downstream valves. SIL 257 stated that the most probable failure mechanism was Transgranular Stress Corrosion Cracking (TGSCC) associated with chloride or other halide contamination in the crevice between the collar and the probe. The SIL also recommended replacing susceptible probes with a modified probe design, which added a seal weld to dose the crevice. Dresden has no

record of the GE SIL 257 disposition. There is no database of how Dresden dispositioned SILs prior to 2001. An inactive, partial database concerning Dresden SIL 257 actions, obtained from GE records, recorded the status of SIL implementation as "do not intend to implement" for U3. The database did not have any record associated with U2. Thus, GE SIL 257 was either never dispositioned or was inadequately dispositioned at Dresden Station (Causal Factor 2).

1980 and 1981: The U2 and U3 Feedwater spargers were replaced to address design deficiencies in the original spargers. The original spargers had a 3" diameter hole at the T-box by design. No records were found indicating the removed spargers had been inspected for sample probe debris.

08/06/90: GE issued SIL 518, Improved Recirculation Water Chemistry Sample Probe, to address a failure of a recirculation sample probe during hot preoperational testing at a GE BWR 6 plant. The cause of the probe failure was identified as high cycle fatigue caused by flow induced vibration. GE SIL 518 was either never dispositioned or was inadequately dispositioned at Dresden station (Causal Factor 2).

04/30/92: Perry Nuclear Power Plant experienced a Feedwater probe failure causing feedwater sparger damage. The failed probe was of the pre-GE SIL 257 type. The Perry Root Cause Report rejected the transgranular stress cracking failure mode postulated in SIL 257 and determined the cause to be cyclic fatigue resulting from flow-induced vibrations. The probe was found in the feedwater sparger. A new probe, whose design was based on the GE SIL 518 recommendations, was installed. The feedwater sparger was repaired per the GE recommendations. Although GE had been provided with the information that this event had occurred, no record could be found to indicate that Dresden was aware of this event and no OPEX could be located on the INPO website.

1995: Quad Cities Station performed an NRC mandated SIL review which included SIL 257. As a result, Quad Cities Station inspected and replaced sample probes in accordance with SIL 257 recommendations. No record could be found to indicate that Dresden Station was made aware of, or took any actions, based on the Quad Cities experience (Causal Factor 5).

D2R14, 1995: Dresden Station performed an external visual inspection of the U2 feedwater spargers per the In-Vessel Visual Inspection (IVVI) Program. The inspection found no damage to the spargers. During D2R18, the tapes for those inspections were reviewed again to verify that there was no damage to the feedwater spargers.

09/30/99: INPO SEN 204, Water Chemistry Induced Fuel Leaks, was released as a result of a River Bend Station water chemistry incident. The SEN prompted the Dresden Chemistry Department to develop a plan of action to investigate the condition of the feedwater sample probes. Work Requests were generated to inspect the U2 and U3 feedwater sample probes during Refueling Outages D2R17 and D3R17 respectively. At this time, only the feedwater probes were scheduled for inspection because Chemistry and Engineering personnel were unaware similar probes were installed at two locations in the condensate system on each unit. This resulted from lack-of-ownership of the water sampling systems (Causal Factor 3).

D2R17: 10/31/01, the U2 feedwater probe was inspected per WO 99146160 and was missing. The probe was replaced per the GE SIL 257 recommendations. There were no actions taken at this time to attempt recovery of the failed probe.

Nuclear Fuels (NF) Evaluation NFM-MW:01-0351 was completed and concluded the most likely probe location was the U2 feedwater spargers. It also concluded it was highly unlikely the probe would be driven through the sparger nozzles. The evaluation also stated that, should the probe exit the spargers, it would most likely be carried to the lower vessel plenum, where the size and weight of the probe would not allow it to be swept by the fluid flow toward the inlet orifices of a fuel support casting. NF concluded there were no safety concerns associated with operation of U2 with the missing original feedwater sample probe.

The modified replacement U2 feedwater probe was constructed per the recommendations of GE SIL 257 to address the failure mechanism of TGSCC. CR 81081, titled Portion of U2 FW Sample Probe Missing Upon Inspection, was generated as a result of the U2 probe being missing. The Corrective Action (CA) was to replace the probe with a SIL 257 probe and two additional corrective actions were generated. CA 81081-04 assigned NFM to perform the lost parts analysis summarized above and CA 81081-05, which required Chemistry to review the event, poll industry experience and initiate service requests to establish the appropriate PM frequencies. Chemistry closed CA 81081-05with one phone call to Quad Cities to benchmark their PM frequency. No record exists of an OPEX or SIL review. If a review was performed, then GE SIL 518, which documents potential fatigue failure of sample probes, or the OPEX on sample probe failures at Browns Ferry and Grand Gulf may have been identified. The failure to perform an aggressive and expanded polling of industry experience and the lack of a questioning attitude represented a missed opportunity to identify other possible failure mechanisms (Causal Factor 6).

D3R17: The U3 inspection of the original feedwater probe, performed per WO 99146161, found the probe missing. This probe was also replaced per the SIL 257 recommendations. In letter NFM-MW:02-041, NF concluded the evaluation performed for the probe lost on U2 per NFM-MW:01-0351 was applicable to U3.

The modified replacement probe for U3 feedwater incorporated the recommendations of GE SIL 257 to address the failure mechanism of TGSCC but did not incorporate the recommendations of GE SIL 518 to address the high cycle fatigue failure mechanism caused by flow induced vibration. CR 127346, titled "Portion of U3 FW Sample Probe Missing Upon Inspection", was generated as a result of the Unit 3 probe being found missing. This CR was largely a "cut and paste" of CR 81081. The CAs included probe replacement with a SIL 257 probe, CA 127346-04 for NFM to conduct a lost parts analysis and CA 127346-04 for Chemistry to trend parameters to access the modified probe effectiveness. An OPEX review was not directed and the CR did not address why GE SIL 257 was not previously dispositioned or require an additional extent of condition SIL or OPEX review (Causal Factor 6).

The original feedwater probes found missing in D2R17 and D3R17 had the known susceptibility to TGSCC and the unknown susceptibility to flow-induced fatigue failure Causal Factor 1. CR 81081 and 127346 both stated that the probe was assumed to have failed in accordance with SIL 257 based on visual observation. A detailed failure mode analysis was not performed.

D2R17 and D3R17: Pre-filter modifications performed for U2 and U3 required the piping sections containing the Condensate Pump Discharge (CPD) probes to be replaced. Since the probes were incorrectly assumed to be wall taps, no investigation was performed to verify the probe status (Causal Factor 4). Therefore, either the probes were intact and removed with the piping, or the probes had previously failed. If the probes had failed before the pre-filter modification installation, then a high probability exists the failed probes are

located in Steam Jet Air Ejector (SJAE) Intercondenser water box or in piping between the condensate pre-filter and the SJAE Intercondenser water box. It is not believed that the probe inside of the SJAE Intercondenser water box will affect efficiency or cause significant damage, however, ACIT07 will evaluate this condition.

8.2 Event Summary - Recent Events

10/09/03: During a PM inspection of the 2D Condensate Booster Pump per WO 560652, a sample probe was found in the pump casing. The probe caused damage to the pump impeller, pump casing, and shaft. The subsequent investigation identified the probe as the Condensate Demineralizer Effluent (CDE) probe, which was installed in 1971, and was not known to exist by Chemistry or Engineering personnel (Causal Factor 3). The P&ID indicated a wall tap instead of the probe (Causal Factor 4). Exelon PowerLabs assisted in determining the possible time and failure mechanism of the probe and subsequent pump damage. The pieces missing from the probe, pump casing and impeller were analyzed by NFM in EC 345441. This evaluation concluded while these pieces have a moderate risk of causing fuel fretting, there are no fuel or control rod drive safety concerns associated with U2 operation from the missing probe pieces.

Further investigation into this incident yielded the discovery of three additional sample probes susceptible to the failure mechanism described in SIL 257. These probes include the U3 CDE probe, U2 Condensate Pump Discharge (CPD) probe and U3 CPD probe.

Further analysis of the damaged CDE probe by PowerLabs (Attachment 3) Probe ID 0, during D3M10 indicated that it was constructed of schedule 160 pipe (wall thickness 0.219) as opposed to the schedule 316 XXS pipe (wall thickness 0.308) required per design document GE Drawing 921D233, titled Sample Probe for Feedwater Control System, for both the pre-SIL 257 and post-SIL 257 probes. No failure mechanism was discovered.

10/29/03: During feedwater sparger external inspections per (IVVI 4 refuel PM, damage was discovered to two of the nozzles on the N4C sparger located at 240°. Three holes were discovered, two in the second nozzle from the T-box and one in the third nozzle from the T-box.

11/02/03: A boroscopic inspection was performed inside of the sparger in the area of the damage. The inspection found an isokinetic probe inside the N4C sparger. The probe was removed from the sparger and the damage was repaired per GE recommendations. The sparger internal inspection was discontinued when the probe was found.

The removed probe was found to be stainless steel and 10 ¾ " in length, 1.05" OD and 0.434" ID. Both ends of the probe exhibited wear. There were earmarks along the length of the probe that were consistent with the holes created in the nozzles of the N4C Feedwater sparger.

PowerLabs performed a further analysis of the probe (Attachments 3 and 4; Probe ID 1); the probe was constructed of type 316 stainless steel, schedule XXS pipe per the design documents. However, this probe had two 1/16th inch holes and one 1/8th inch hole instead of three 1/8th inch holes as required by the design document. A failure mechanism could not be determined due to the failed end being damaged.

11/25/03: The U3 CDE probe was UT inspected and found to be intact.

D3M10, 12/6/03 – Present: In response to the Feedwater sparger damage found during refueling outage D2R18, an inspection plan was developed to find the Feedwater probe previously identified as missing in D3R17. The plan expected to find one probe in the Feedwater spargers. External inspections of all Feedwater spargers and the top of the moisture separator reveled no damage or missing sample probe. Boroscope internal inspections were then performed on all of the Unit 3 Feedwater spargers. This inspection identified two probes and a washer in the N4B sparger. No sparger damage was detected. One of the two probes was believed to be the feedwater probe, which was previously identified as missing in D3R17. The other probe is believed to be the GE SIL 257 probe that was installed during D3R17. The origin of the washer was indeterminate.

The two Feedwater probes found in the Unit 3 sparger were not damaged at the failure surface. Analysis of these probes (Exelon Power Labs Field Report DRE-90509) identified the mode of failure for both as fatigue cracking, most likely due to high cycle/low stress loading.

12/11/03: U2 entered forced outage D2F40 due to high generator stator cooling temperatures. During the outage, a UT inspection indicated that the modified U2 Feedwater probe that was installed in D2R17 per GE SIL 257 recommendations was missing. A summary of the status of all sample probes that were installed in the feedwater and condensate system is included in Table 1.

Table 1: Feedwater and Condensate Sample Probe Failure Summary

Sample Probe	Install Date	Design Spec	Status	Recovered	Additional Actions Required
Unit 2					
Original Feedwater Probe	1971	Per GE Drawing 921D233	Failed	Note 1	Note 1
Modified Feedwater Probe	2001	Per GE SIL 257	Failed	Note 1	Note 1, CAPR 2 for installation of redesigned probe in D2R19
Original Condensate Demineralizer Effluent (CDE)	1971	Per GE Drawing 921D233	Failed	Yes	None
Modified Condensate Demineralizer Effluent (CDE)	2003	Per GE SIL 257	In-Place	Not Applicable	Engineering to review probe susceptibility to fatigue failure
Original Condensate Pump Discharge (CPD)	1971	Per GE Drawing 921D233	Note 2	Note 2	Note 2
Modified Condensate Pump Discharge (CPD)	2001	Per Condensate Prefilter Mod	In-Place	Note 3	None
Unit 3					
Original Feedwater Probe	1971	Per GE Drawing 921D233	Failed	Yes	None
Modified Feedwater Probe	2002	Per GE SIL 257	Failed	Yes	CAPR 1 for installation of redesigned probe in D3R18
Original Condensate Demineralizer Effluent (CDE)	1971	Per GE Drawing 921D233	Removed Intact	Not Applicable	None
Modified Condensate Demineralizer Effluent (CDE)	2003	Per GE SIL 257	In-Place	Not Applicable	Engineering to review probe susceptibility to fatigue failure
Original Condensate Pump Discharge (CPD)	1971	Per GE Drawing 921D233	Note 2	Note 2	Note 2
U3 Modified Condensate Pump Discharge (CPD)	2001	Per Condensate Prefilter Mod	In-Place	Note 3	None

Note 1: A Feedwater sample probe was removed from the U2 RPV feedwater sparger in D2R18. However, the subsequent inspection could not determine if the recovered probe was the original or modified feedwater sample probes. One probe was recovered and one probe is assumed to remain in the U2 RPV feedwater sparger. Recovery of the missing probe is planned during D2R19 per WR 124348.

Note 2: The U2 and U3 Original Condensate Pump Discharge (CPD) sample probes either failed before the condensate pre-filter modifications were installed in 2001, or the probes were unknowingly removed with the condensate pipe replacement during the modification implementation. If the CDE probes failed prior to the condensate pre-filter modification installation, then the probes are assumed to be located in the respective Unit Steam Jet Air Ejector (SJAE) A or B Intercondenser water box. WRs 121350 and 121351 were generated to inspect the U2 SJAE A and B Intercondenser water boxes respectively. WRs 121352 and 121353 were generated to inspect the U3 SJAE A and B Intercondenser water boxes, respectively.

Note 3: The U2 and U3 Original CPD sample probes were installed by the condensate pre-filter modification in 2001. The probes are not susceptible to the TGSCC and fatigue failure mechanisms identified in this RCR. Therefore no additional actions are required.

12/12/03: General Electric conducted a qualitative assessment of the impact of Feedwater sampling and analysis and issued a draft report. This assessment supported the continued operation of Dresden Unit 2 and Unit 3 for the remainder of the fuel cycle in both units without installation of new Feedwater probes. The report determined the soluble impurity analysis would be unaffected while insoluble impurity analysis will be no more than 20% low. This condition was also addressed in CR190571, written to document the potential affects of the missing probe on reactor water chemistry.

The evaluation of several potential failure mechanisms determined that the most likely cause of the failures in both Units 2 and 3 was mechanical high cycle fatigue induced by flow-induced vibration. The driving force for the failure mechanism was explained as the extended length of the probe protruding into the flow stream. The report predicted that, once the length was lost, the driving force would no longer exist and no further parts would be lost.

In the report, GE calculates that natural frequency of the Feedwater sample probe as 136 Hz and a Feedwater flow induced vortex shedding frequency at about 140 HZ. The report states that under these conditions the probe natural frequency locks in with the vortex shedding frequency and the probe responds in a resonant condition, which leads to high cycle fatigue failure of the sample probe. In the range of Reynolds Numbers in the Feedwater line, lock-in was predicted to occur over a wide range of flow rates. (GE-NE-0000-0024-2731). A Dresden review of the GE calculations indicated that the length used on this calculation was incorrect. When corrected the natural frequency was approximately 166 Hz but still within the vortex shedding lock-in range.

Operability Determination, 03-015 was revised to determine the operability of U2 with a feedwater sample probe missing and potentially in the reactor pressure vessel. This evaluation reviewed the potential effects of the missing probe and potential feedwater flow impingement on the RPV cladding. The missing probe was postulated to have created a condition similar to what was seen on U2 during D2R18; this condition was analyzed and it was concluded that there is no fuel or control rod drive safety concerns. Similarly, feedwater impingement on the vessel wall due to potential holes in the sparger is not expected to occur. Additionally, Local Leak Rate Tests last performed on the Feedwater Containment Isolation Check Valves confirmed that the probe did not become lodged in the seating area

of a containment isolation valve. The probe would not be expected to lodge in a containment isolation valve based on size and geometry. Thus, the containment, reactor vessel, feedwater sparger and feedwater system are operable.

12/13/03: The sample probes and washer found in the Unit 3 sparger were removed. Exelon PowerLabs conducted analysis of the two sample probes removed from Unit 3 sparger, the sample probe removed from the Unit 2 sparger and a recently removed, intact and original Unit 3 CDE probe. See Attachment 2 for a Summary of Recovered Probes

Exelon PowerLabs field report, DRE-90509, concluded that both probes retrieved from the Unit 3 feedwater sparger failed due to fatigue cracking, most likely due to high cycle/low stress loading. The failure mechanism for the probe from the Unit 2 sparger could not be determined due to severe mechanical damage and wear of the fracture surface.

Of the two failed probes that were removed from the Unit 3 feedwater sparger, some manufacturing differences were noted at the welded end cap and the pipe sizes were of a different schedule, but the fracture surfaces of both probes showed benchmark patterns consistent with fatigue failure. The fracture surface of the probe removed from the Unit 2 feedwater sparger during D2R18 had been polished smooth by wear. This wear was caused as the probe wore against the nozzles on the N4C sparger during the previous operating cycle.

The intact probe, removed from the Condensate Demineralizer Effluent Sample point on Dresden 3, was of the original design described in GE SIL 257, without the SIL recommended improvements. This probe was the same type and came from the same location in the system as the Dresden 2 probe recovered from the 2D Condensate Booster Pump Casing on 10/9/03. The Unit 2 CDE probe was damaged sufficiently by interaction with pump impeller to preclude any useful analysis.

Based on the dimensions of the five probes that power labs conducted analysis on, Dresden Engineering determined that the two CDE probes that were considered to be from the original 1971 installation and one of the sample probes from the Unit 3 sparger were made from schedule 160 pipe, which has 2/3 the wall thickness that the original design called for. One of the probes from the Unit 3 sparger and the probe retrieved from the Unit 2 sparger were made from schedule 316 XXS piping as specified in the design drawings and the D2R17 and D3R17 work packages.

The PowerLabs analysis, GE analysis, industry experience and the relatively rapid failure of the feedwater probes installed in D2R17 and D3R18, support the conclusion that the failure mechanism of the two probes that were extracted from the Unit 3 sparger is flow induced fatigue failure. Since one of these probes can be assumed to be a pre-SIL 257 design installed in 1971 and the other a post-SIL 257 design installed in 2002, this is indication that both designs and both schedules of pipe used in the probes are subject to fatigue failure at operating conditions that have been experienced both before and after 2001. (Causal Factor 1)

12/16/03 GE conducted an analysis of the actual Unit 3 feedwater probes recovered during D3M10 to determine the natural frequency based on actual length of the probe, and Vortex Shedding Frequency for the flow conditions that each probe experienced. A similar analysis for the currently installed Condensate Demineralizer Effluent probes was also conducted from Work Package dimensions and current operating conditions. Finally, the analysis determined the susceptibility of the SIL 257 probe to flow induced fatigue failure at flow velocities seen prior to D2R17 and D3R17. The results, summarized in Table 2, indicate

that both the original 1971 probe and the SIL 257 modified probe, installed in 2002, are susceptible to Vortex Shedding lock on and therefore were susceptible to flow induced vibration at their respective operating conditions. This analysis is consistent with industry operational experience and flow induced fatigue failure for both of the probes. The analysis further determined that the SIL 257 probes would not have been locked in to Vortex Shedding frequency at flow velocities seen prior D2R17 and D3R17. The flowrates for both the condensate and feedwater lines increased after D2R17 and D3R17. However, the combination of changing the effective length of the probe and the flow rate at the same time was not analyzed prior to installation (Causal Factor 7).

Table 2: GE Vibration Analysis Summary

Probe Design	Length of probe from end cap to seal weld	Schedule	Flow Rate (fps)	Natural Frequency (Hz)	Vortex Shedding Frequency (Hz)	Ratio of VSF to Natural Frequency
Case 2: U3 FFW Probe (pre-SIL 257)	14 ¾"	160	21.67	134	116	0.87
Case 3: U3 FFW probe (post SIL 257)	13 $^1/_8$"	XXS	26.1	163	140	0.86
Case 4: U2 CDE probe (post SIL 257)	16 $^1/_8$"	XXS	11.28	109	32	0.3
Case 5: As Design (pre SIL 257)	14 ¾"	XXS	21.67	129	116	0.9
Case 6: Post SIL (pre 2001)	13 $^1/_8$"	XXS	21.67	163	116	0.71

All probes are ¾" nominal pipe size.

SIL 257 recommended an inner and outer seal weld to sample probes. With one seal weld, some uncertainty is introduced into the natural frequency calculations because of possible dynamic effects of the external piping and valves, about which information is typically not readily available. Consideration of these external piping / valve / supports effects in an engineering calculation would require extensive finite element analysis, and is beyond the scope of this evaluation. It is noted, however, that these effects may cause some variance in the calculated structural natural frequencies in this calculation for those cases, and these differences are not expected to be very large. Those cases with one seal weld (Pre-SIL 257) are reported as cases 2 and 5. These are the original design.

For those probes with two seal welds, greater isolation is provided from the external piping / valve / supports, and so this variance would be smaller in those cases. Those cases with two seal welds (Post-SIL 257) are reported as cases 3, 4 and 6. These are the modified design Based on the ASME Boiler and Pressure Vessel Code, Appendix N titled "Dynamic Analysis Methods", paragraph N-1324, values of the ratio of vortex shedding frequency to probe natural frequency, provided in the last column, below 0.76 are considered acceptable. However, other documents provide more stringent criteria. The closer the ratio of vortex shedding frequency to probe natural frequency is to 1, the more likely a resonance lock-on condition is to occur.

The chart above suggest that both cases 2 and 3 should have failed very rapidly due to the Maximum Vortex shedding frequency to natural frequency ratio. However, when you take into account the greater isolation seen by case 3, it is understandable that it would fail more rapidly. There is a variance to the ratio seen by case 2 that could in fact explain the extended life of the original probes.

9.0 Evaluation

Attachment 1 contains the Event and Causal Factor Chart, developed to document the event timeline and identify the causal factors. Attachment 2, Barrier Analysis, was performed to identify which Barriers apparently failed and allowed the event to progress. TapRooT© was used for the disciplined cause and effect analysis to identify the problem and evaluate the Causal Factors.

A. Causal Factor 1:

*The **Root Cause** of the U2 and U3 Feedwater sample probe failures and subsequent U2 Feedwater Sparger damage was attributed to a Feedwater sample probe design deficiency. The original U2 and U3 Feedwater sample probes installed per GE Drawing 921D233, titled Sample Probe for Feedwater Control System, Revision 1, dated 04/26/68, were susceptible to the failure mechanisms of Transgranular Stress Corrosion Cracking (TSGCC) and fatigue failure resulting from flow-induced vibration. The modified U2 and U3 Feedwater sample probes installed per the design specified in the GE Service Information Letter (SIL) 257 during D2R17 and D3R17 are susceptible to a fatigue failure mechanism resulting from flow-induced vibration.*

a. *Error Precursor: The industry standard design and GE recommended re-design per SIL 257 were not challenged and a Dresden site-specific review was not performed.*

b. Cause: This was the root cause for these events.

c. Bases: Industry experience identified the probes are susceptible to TGSCC and fatigue failure resulting from flow-induced vibration. The Exelon PowerLabs testing of the retrieved U3 original and modified Feedwater sample probes identified the failure mechanism as a fatigue failure. The probe retrieved from the U2 Feedwater sparger was too worn to ascertain the failure mechanism.

d. Corrective Actions to Prevent Recurrence: Install sample probes in the U2 and U3 Feedwater systems which are tolerant to the failure mechanisms of TGSCC and fatigue failure resulting from flow-induced vibration.

Problem and Causal Factor Coding			Process and Organization Codes			
Trend 1	Trend 2	Trend 3	Process Code	Process Name	ORG Code	ORG Name
EQM	MM	3S	CC22	Design Adequacy	VD02	Design Engineering

Causal Factor 2:

A **Contributing Cause** was attributed to Dresden's failure to effectively disposition GE SILs related to sample probe failures. On 12/30/77, GE issued SIL 257, titled Improved Feedwater Sample Probe, to address sample probe failures attributed to TGSCC. On 08/06/90, GE issued SIL 518, Improved Recirculation Water Chemistry Sample Probe, to address sample probe failures attributed to fatigue failure resulting from flow-induced vibration. Dresden implemented the recommendations of SIL 257 for the modified Feedwater sample probes in Refueling Outages D2R17 and D3R17. However, the recommendations in SIL 518 were not performed.

a. *Error Precursor: Inadequate internal and external communication. Ineffective process for industry experience review and disposition.*

b. Cause: This was a contributing cause for these events.

c. Bases: Failure to implement the recommendations of GE SIL 257 is the probable cause of the U2 CDE probe failure. The historical failure to implement the recommendations of GE SIL 518 is considered to be the cause of the original and modified U3 Feedwater probe failures. Dresden does not have an effective database of SILs and OPEX dispositioned prior to 2001.

d. Corrective Actions:
   - Regulatory Assurance to review the Dresden process for dispositioning GE SILs and Technical Information Letters (TILs). Initiate additional actions as appropriate
   - *Regulatory Assurance to develop a database of GE SILs and which documents the Dresden disposition and coordinate a review of open SILs per LS-AA-115. Initiate additional actions as appropriate*

Problem and Causal Factor Coding			Process and Organization Codes			
Trend 1	Trend 2	Trend 3	Process Code	Process Name	ORG Code	ORG Name
PWP – People Work Practices	WPK - Lack of Communication	5NM – Communication System Need Improvement	LS06	OPEX/NON	RA	Regulatory Assurance

Causal Factor 3:

A **Contributing Cause** was attributed to a lack of ownership involving sample systems, specifically sample point locations, sample device drawings, and sampling methods.

a. *Error Precursor: Lack of communication, lack of knowledge.*

b. Cause: This was a contributing cause for these events.

c. Bases: Neither Engineering nor Chemistry demonstrated ownership regarding details about sample point locations, sampling devices, or sampling methods. No clear lines of ownership currently exist.

d. Corrective Actions:

- Establish memorandum of agreement between Chemistry and Engineering to define the boundaries, roles and responsibilities for Chemistry sample systems
- Review and document the disposition of Training Request 03-2071 to perform training on sample system theory and design training for Chemistry Technicians and Managers. Initiate additional actions as appropriate

Problem and Causal Factor Coding			Process and Organization Codes			
Trend 1	Trend 2	Trend 3	Process Code	Process Name	ORG Code	ORG Name
PWP – People Work Practices	WPC - Lack of Communication	4NC No communication or not timely	CY50	System Chemistry	ENSE/CH	Engineering – System Engineering/Chemistry
PWP – People Work Practices	WPK – Lack of Knowledge	5DL Learning Objective	TQ53	Engineering Training	ENSE	Engineering-System Engineering
PWP – People Work Practices	WPK – Lack of Knowledge	5DL Learning Objective	TQ56	HP and Chemistry Training	CH	Chemistry

Causal Factor 4:

A **Contributing Cause** was attributed to historical inadequate design documentation for the Feedwater, Condensate Demineralizer Effluent and Condensate Pump Discharge sample probe type and location.

a. *Error Precursor: Lack of knowledge and ownership of the sampling system within Chemistry and Engineering.*

b. Cause: This was a contributing cause for the CDE and CPD probe failures because Chemistry and Engineering personnel did not know that the probes existed when responding to SIL 257 for Feedwater probes.

c. Bases: Original Piping and Instrument Diagrams (P&IDs) M-15 titled U2 Condensate Piping and M-348 titled U3 Condensate Piping did not identify the CDE and CPD probe installations. Therefore these probes were not included in the corrective actions upon the identification of SIL 257.

d. Corrective Actions:

- Review prefilter mod and sample probe work packages and initiate appropriate changes to drawings
- Review the U2 and U3 Feedwater and Condensate design document sample probe information and revise as appropriate. The review shall encompass P&IDs, Condensate Prefilter modification and probe detail drawings

Problem and Causal Factor Coding			Process and Organization Codes			
Trend 1	Trend 2	Trend 3	Process Code	Process Name	ORG Code	ORG Name
PRD – Process Document	PRDQ – Document Quality	4FI – Wrong	CC10	Configuration Control Process	CH	Chemistry

Causal Factor 5:

A **Contributing Cause** was attributed to a historical lack of communication and implementation of corrective actions from events that have occurred outside of Dresden Station, both internal and external to the Exelon fleet.

a. *Error Precursor: Lack of communication. Lack of knowledge.*

b. *Cause: This was a contributing cause for these events.*

c. Bases: In 1995 Quad Cities removed and replaced all original sample probes as a result of an NRC mandated SIL review. During the SIL review Quad Cities identified their failure to previously implement the recommendations of GE SIL 257. No communication with Dresden was documented. Note: All Quad Cities sample probes were removed in varying states of failure, but none had migrated downstream.

   Dresden was not aware of OPEX outlining the failure of sample probes, in particular, events at Browns Ferry and Grand Gulf.

   Discussions with Engineers who have completed initial training expressed a knowledge deficiency in the retrieval of web based industry experience.

d. Corrective Actions:

   - Review and document the disposition of Training Requests 03-2069 and 03-2070 to perform training on the retrieval of web based industry experience to Maintenance Work Planning and Engineering personnel respectively. Initiate additional actions as appropriate.
   - Conduct a Check-in Self Assessment on the Site OPEX program per LS-AA-126-1005.

Problem and Causal Factor Coding			*Process and Organization Codes*			
Trend 1	*Trend 2*	*Trend 3*	*Process Code*	*Process Name*	*ORG Code*	*ORG Name*
PWP – People Work Practices	*WPK - Lack of Communication*	*5NM – Communication System Need Improvement*	*LS06*	*OPEX/NON*	*RA*	*Regulatory Assurance*
PWP – People Work Practices	*WPK - Lack of Knowledge*	*5LP-Lesson Plan Needs Improvement*	*TQ20*	*Training Material*	*TG*	*Training*

Causal Factor 6:

A **Contributing Cause** was attributed to ineffective corrective actions associated with CRs 81081 and 127346, which were generated for the missing Feedwater probes, identified in D2R17 and D3R17 respectively.

a. *Error Precursor: Mindset, Lack of Questioning Attitude Assertiveness.*

b. Cause: This was a contributing cause for these events.

c. Bases: In September 1999, SEN 204 titled Water Chemistry Induced Fuel Leaks,

prompted work requests to perform material condition inspections to be performed on the U2 and U3 feedwater sample probes during D2R17 and D3R17 respectively. These inspections identified both probes were missing. The associated CRs addressed the failure mechanism as, "As discussed in GE SIL 257, the apparent cause is transgranular stress corrosion cracking" and the corrective action as "Replace the probe in conjunction with the guidelines of GE SIL 257. When reinstalled per the SIL, one of the mechanisms for corrosion cracking is eliminated. No further evaluation is required as the cause is discussed and understood via the GE SIL." The Corrective Action Program (CAP) did not address why SIL 257 was not previously dispositioned or require an Extent of Condition review of SILs and OPEX.

CA 81081-05 was assigned to Chemistry to poll industry experience for sample probe failures. However, CA 81081-05 was closed after one phone call to Quad Cities. This is representative of a mindset and lack of a questioning attitude.

d. Corrective Actions: Regulatory Assurance to generate an information package for Department Heads and Branch Managers related to the OPEX review process and resources.

Problem and Causal Factor Coding			***Process and Organization Codes***			
Trend 1	***Trend 2***	***Trend 3***	***Process Code***	***Process Name***	***ORG Code***	***ORG Name***
PWP-People Work Practices	*WPWS – Use of Work Standards*	*4CA – Corrective Action*	*LS02*	*Corrective Action Program*	*CH*	*Chemistry*
PWP-People Work Practices	*WPWS – Use of Work Standards*	*4CA – Corrective Action*	*LS02*	*Corrective Action Program*	*RA*	*Regulatory Assurance*
HPH – Human Nature	*5 - Mindset*	*J – Questioning Attitude*	*LS02*	*Corrective Action Program*	*CH*	*Chemistry*

Causal Factor 7:

A **Contributing Cause** was attributed to changing the effective probe length when the SIL 257 probes were installed while at the same time changing the system flow velocity during D2R17 and D2R18 without performing an engineering analysis of the new conditions.

a. *Error Precursor: Lack of knowledge and ownership of the sampling system within Chemistry, Engineering and Project Management.*

b. Cause: This was a contributing cause for this event.

c. Bases: During D2R17 and D3R17, the SIL 257 recommended probes were installed in the Unit 2 and Unit 3 Feedwater systems in place of the original probes. The installation of the SIL 257 probe changed the effective length of the probe due to the location of an additional weld. An analysis of the feedwater SIL 257 probes performed during this root cause found the SIL 257 probe was not susceptible to Vortex Shedding lock under the flow conditions seen prior to D2R17 and D3R17. This analysis further discovered the probes were susceptible to Vortex Shedding lock on at the flow velocities seen after D2R17 and D3R17 and were susceptible to flow induced vibration.

d. Corrective Actions:

- Install sample probes in the U2 and U3 Feedwater systems, which are tolerant to fatigue failure resulting from flow-induced vibration.
- Perform a review of the Condensate and Feedwater thermowells and other instrumentation that protrudes into the flow stream for vulnerability to a flow induced vibration failure mechanism. Initiate additional actions as appropriate.

Problem and Causal Factor Coding			*Process and Organization Codes*			
Trend 1	*Trend 2*	*Trend 3*	*Process Code*	*Process Name*	*ORG Code*	*ORG Name*
EQM	*MM*	*3S*	*CC22*	*Design Adequacy*	*ENDE*	*Design Engineering*

B. Corrective Actions

1. Immediate and Interim Corrective Actions

a. Immediate Corrective Actions (ImCA):

ImCA1:

Action: WO 560652-01 was completed to repair the 2D Condensate Booster Pump per ECR 362066 in D2R18.

Assignee: Ludwig
Status: Complete

ImCA2:

Action: WO 625592-01 was completed to inspect and repair the U2 CDE sample probe in D2R18.

Assignee: Ludwig
Status: Complete

ImCA3:

Action: WO 641422-01 was completed to inspect the U3 Feedwater Sparger and retrieved the original and modified Feedwater sample probes in D3M10.

Assignee: Ludwig
Status: Complete

ImCA4:

Action: WO 632536 -01 was completed to repair the U2 FW Sparger.

Assignee: Baron
Status: Complete

b. Interim Corrective Actions (InCA):

Action: Obtain GE Analysis on Chemistry results to allow continued operation of Unit 2 and Unit 3 without sample probes.

Assignee: Dorsey
Status: Complete

2. Corrective Action to Prevent Recurrence (CAPR):

CAPR1: Install modified U3 Feedwater sample probe.
Assignee: K. Ludwig
Alert Group: A8322MM
Due Date: 12/17/04, Schedule D3R18
ATI: CAPR 183901-28

CAPR2: Install modified U2 Feedwater sample probe.
Assignee: K. Ludwig
Alert Group: A8322MM
Due Date: 12/02/05, Schedule D2R19
ATI: CAPR 183901-29

3. Corrective Action (CA):

CA01:
Action: Issue a modification for the installation of a U2 Feedwater sample probe. Initiate additional actions as appropriate.
Assignee: Loch
Alert Group: A8352NESDM
Due Date: 02/27/04
ATI: CA 183901-30

CA02:
Action: Issue a modification for the installation of a U3 Feedwater sample probe. Initiate additional actions as appropriate.
Assignee: Loch
Alert Group: A8352NESDM
Due Date: 02/27/04
ATI: CA 183901-31

CA03:
Action: Schedule the U2 and U3 Feedwater sample probe modifications in the respective refueling outages and forced outages of sufficient duration. Initiate additional actions as appropriate.

Assignee: Bockholdt
Alert Group: A8340OUT
Due Date: 03/26/04
ATI: CA 183901-32

CA04:

Action: Review and document the disposition of Training Requests 03-2069 and 03-2070 to perform training on the retrieval of web based industry experience to Maintenance Work Planning and Engineering personnel respectively. Initiate additional actions as appropriate.
Assignee: Garrison
Alert Group: A8361TR
Due Date: 02/20/04
ATI: CA 183901-33

CA05:

Action: Review the U2 and U3 Feedwater and Condensate design document sample probe information and revise as appropriate. The review shall encompass P&IDs, Condensate Prefilter modification and probe detail drawings.
Assignee: Guerrero
Alert Group: A8352NESDM
Due Date: 02/19/04
ATI: CA 183901-34

CA06:

Action: Regulatory Assurance to review the Dresden process for dispositioning GE SILs and TILs. Initiate additional actions as appropriate.
Assignee: Hansen
Alert Group: A8301RAPR
Due Date: 04/23/04
ATI: CA 183901-35

CA07:

Action: Regulatory Assurance to generate information package for Department Heads and Branch Managers related to the OPEX review process and resources
Assignee: Hansen
Alert Group: A8301RAPR
Due Date: 04/23/04
ATI: CA 183901-36

CA08:

Action: Regulatory Assurance to develop a database of GE SILs and which documents the Dresden disposition and coordinate a review of all GE SILs per LS-AA-115. Initiate additional actions as appropriate.
Assignee: Hansen
Alert Group: A8301RAPR
Due Date: 06/02/04
ATI: CA 183901-37

CA09:

Action: Regulatory Assurance to conduct a Check-in Self Assessment on the Site OPEX program per LS-AA-126-1005.

Assignee: Phalen
Alert Group: A8301RAPR
Due Date: 03/26/04
ATI: CA 183901-38

CA10:

Action: Establish memorandum of agreement between Chemistry and Engineering to define the boundaries, roles and responsibilities for Chemistry sample systems.

Assignee: Dorsey
Alert Group: A8332CHEM
Due Date: 03/05/04
ATI: CA 183901-39

CA11:

Action: Review the U2 and U3 Feedwater and Condensate design document sample probe information and revise as appropriate. The review shall encompass P&IDs, Condensate Prefilter modification and probe detail drawings.

Assignee: Bezouska
Alert Group: A8330NESTB
Due Date: 03/07/04
ATI: CA 183901-40

CA12:

Action: Review and document the disposition of Training Request 03-2071 to perform training on sample system theory and design training for Chemistry Technicians and Managers. Initiate additional actions as appropriate.

Assignee: Garrison
Alert Group: A8361TR
Due Date: 02/20/04
ATI: ACIT 183901-41

CA13:

Action: Review the susceptibility of the U2 and U3 Condensate Demineralizer Effluent sample probes for potential fatigue failure resulting from flow-induced vibration. Initiate additional actions as appropriate.

Assignee: Loch
Alert Group: A8352NESDM
Due Date: 03/25/04
ATI: CA 183901-42

CA14:

Action: Perform a review of the Condensate and Feedwater thermowells and other instrumentation that protrudes into the flow stream for

vulnerability to a flow induced vibration failure mechanism. Initiate additional actions as appropriate.

Assignee: Bezouska
Alert Group: A8330NESTB
Due Date: 04/15/04
ATI: ACIT 183901-43

4. Additional Actions:

ACIT01:

Action: Document the D3R18 inspection results for the U3 CPD sample probe search per WRs 121352 and 121353 for the 3A and 3B SJAE Intercondenser Water Boxes respectively. Initiate additional actions as appropriate.

Assignee: Bonomo
Alert Group: A8330NESTB
Due Date: 12/09/04, Code D3R18
ATI: ACIT 183901-44

ACIT02:

Action: Document the D2R18 inspection results for the U2 CPD sample probe search per WRs 121350 and 121351 for the 2A and 2B SJAE Intercondenser Water Boxes respectively. Initiate additional actions as appropriate.

Assignee: Bonomo
Alert Group: A8330NESTB
Due Date: 12/09/05, Code D2R18
ATI: ACIT 183901-45

ACIT03:

Action: Inspect the 3A and 3B SJAE Intercondenser Water Boxes to perform a search for the U3 CPD sample probe per WRs 121352 and 121353.

Assignee: Bonomo
Alert Group: A8330NESTB
Due Date: 01/28/04
ATI: ACIT 183901-46

ACIT04:

Action: Perform a D2R19 scope addition to perform a search for the U2 CPD sample probe per WRs 121350 and 121351 for the 2A and 2B SJAE Intercondenser Water Boxes respectively.

Assignee: Baxa
Alert Group: A8330NESTB
Due Date: 01/28/04
ATI: ACIT 183901-47

ACIT05:

Action: Review the periodic inspection frequency for U2 and U3 Feedwater sample probes per PMID 167081-01 and PMID 167083-01 respectively, and initiate a Service Request to revise

the PMID frequency as appropriate. Develop the inspection frequencies for U2(3) CDE and U2 (3) CPD sample probes and generate a Service Request to create the new PMIDs. Initiate additional actions as appropriate.

Assignee: Loch
Alert Group: A8352NESDM
Due Date: 04/23/04
ATI: ACIT 183901-48

ACIT06:

Action: Review all Lost Parts Analysis performed by Plant Engineering prior to January 2001 to determine further extent of condition and ensure that all missing foreign material is appropriately dispositioned.

Assignee: Flick
Alert Group: A8330NESTB
Due Date: 10/29/04
ATI: ACIT 183901-49

ACIT07:

Action: Perform loose parts analysis for the missing Condensate Pump Discharge isokinetic sample probe that is potentially inside of the SJAE Intercondenser water box.

Assignee: Bonomo
Alert Group: A8330NESTB
Due Date: 03/26/04
ATI: ACIT 183901-50

ACIT08:

Action: Revise the model work order for the replacement of sample probes to include a step to quarantine the old probe and welded collar for power labs analysis.

Assignee: Osmonson
Alert Group: A8330NESTB
Due Date: 03/26/04
ATI: ACIT 183901-51

NER1:

Action: Review approved RCR to determine need for Yellow NER.

Assignee: Driehaus
Alert Group: NCS A8002OPEX
Due Date: 01/16/04
ATI: NER 183901-26

NNOE1:

Action: Prepare and obtain approval for an OE.

Assignee: Dyas
Alert Group: A8351NESPR

Due Date: **01/16/04**
ATI: NNOE 183901-52

NNOE2:
Action: OPEX Coordinator: Issue OE.
Assignee: Phalen
Alert Group: A8301OPEX
Due Date: **01/23/04**
ATI: OPXR 183901-53

EFR1:
Action: Perform an Individual Effectiveness Review of CAPR 1.
Assignee: Dyas
Alert Group: A8351NESPR
Due Date: **08/19/05**
ATI: EFR 183901-54

EFR2:
Action: Perform an Individual and Collective Effectiveness Review of CAPRs 1 and 2.
Assignee: Dyas
Alert Group: A8351NESPR
Due Date: **12/22/06**
ATI: EFR 183901-55

10.0 Extent of Condition

Sample Probes

Currently, there are 23 probes installed on each Dresden unit and 5 common unit probes for a total of 51 sampling probes. Many of the probes were installed during initial plant construction. All probes were evaluated for applicability to the failure mechanisms identified in this RCR.

Evaluation criteria used:

- Probe design: material, probe length, protrusion length into the process stream;
- Affected by SIL 257 or 518;
- Ability to enter the reactor pressure vessel or primary systems;
- Other downstream components

The following probe locations have been analyzed per the above criteria:

Table 3: Dresden Sample Probes

System/Unit	Unit 2	Unit 3	Unit 2/3
Condensate Demineralizer Outlet	7	7	—
Waste Sample Tanks	—	—	3
Floor Drain Sample Tanks	—	—	2
Reactor Water Clean-Up Filter Inlet	1	1	—
Reactor Water Clean-Up Filter Outlet	1	1	—
Reactor Water Clean-Up Demineralizer Outlet	3	3	—
Fuel Pool Filter Inlet	1	1	—
Fuel Pool Demineralizer Inlet	1	1	—
Hotwell	2	2	—
Condensate Pump Discharge	1	1	—
Condensate Demineralizer Effluent	1	1	—
Feedwater	1	1	—
Feedwater Pump Discharge	3	3	—
Recirculation Loop	1	1	—
Total	**23**	**23**	**5**

Probe Design

Dresden currently utilizes three general sample probe designs. The first, built into the systems as part of original construction, are shown in Figure 2. These probes are made from the same material as the process piping and not susceptible to TGSCC as identified GE SIL 257. Due to the decreased length, the probes with this design have the least exposure to the process stream fluid force. The probe length is a function of the size of the process piping; the probes of this design are no more than 2 inches in length and extend no more than 25% into the process stream. By comparison, the Isokinetic probes are up to 16 inches in length and extend approximately 50% into the process stream by design.

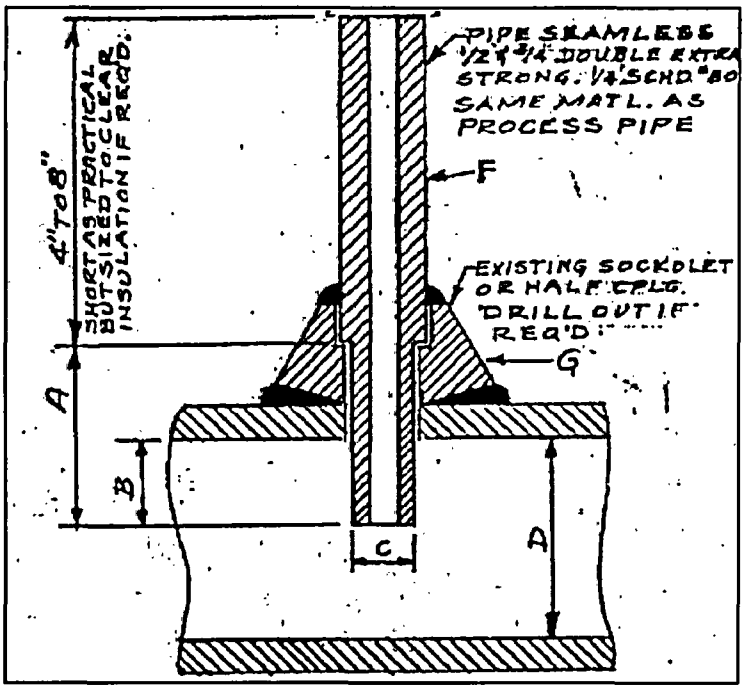

Figure 2: Basic Sampling Probe

In 1971, isokinetic probes as depicted in Figure 3 below, were installed in the U2 and U3 Condensate and Feedwater systems.

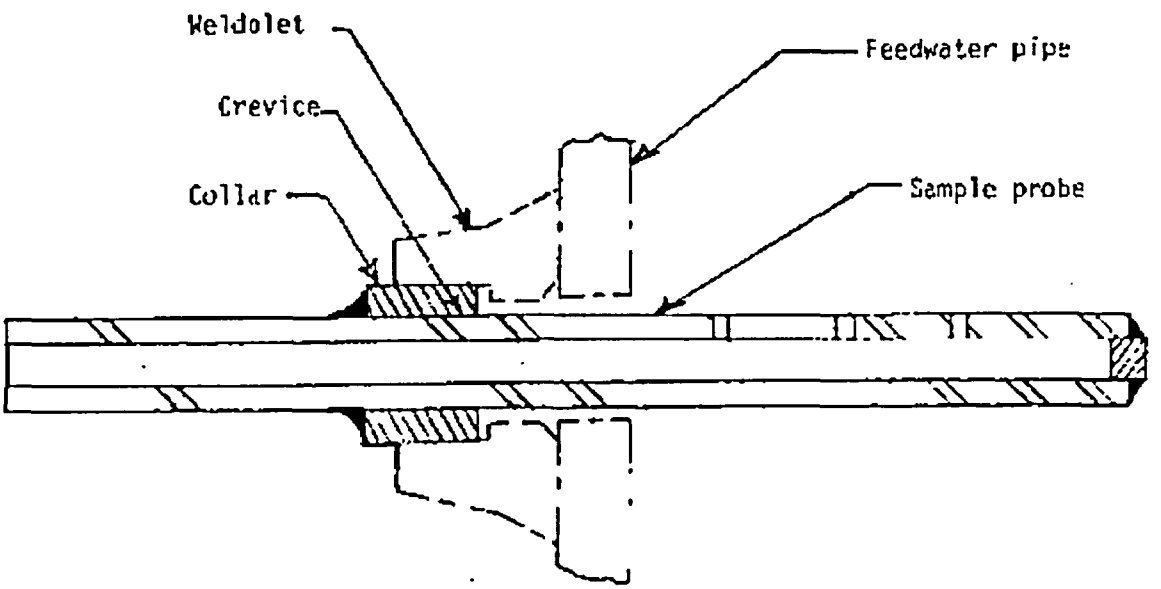

Figure 3: Original Isokinetic Probe Design

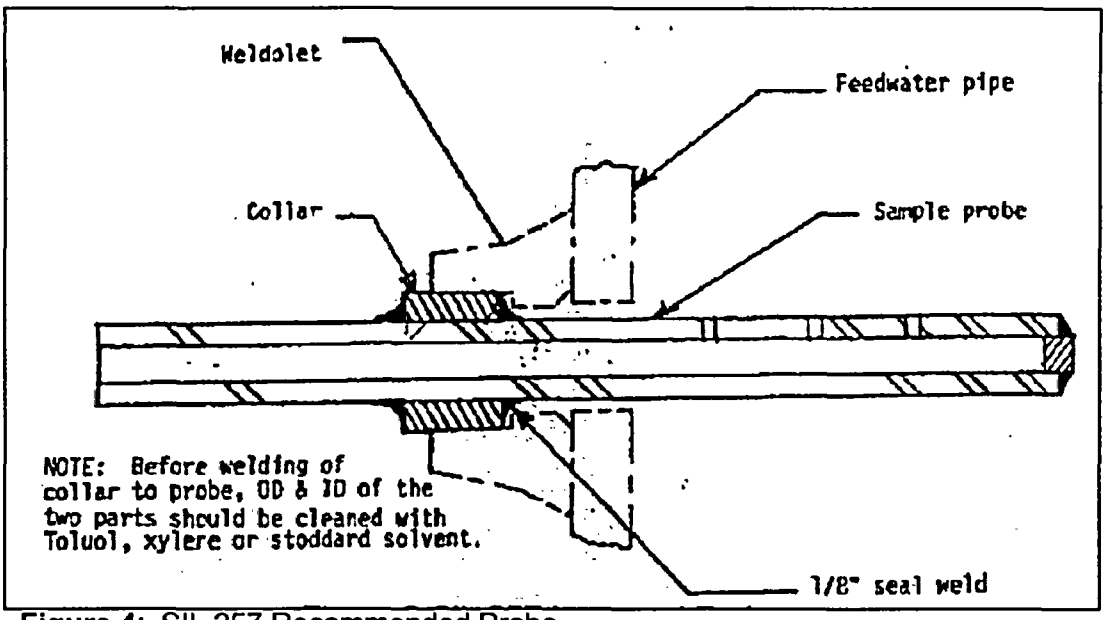

Figure 4: SIL 257 Recommended Probe

In 1977, GE SIL 257 addressed failure of the Condensate and Feedwater probes due to TGSCC. The addition of a seal weld was recommended to prevent TGSCC as outlined in SIL 257 and depicted in Figure 4. Of the six originally installed isokinetic probes, the Feedwater and CDE remain on each Dresden unit. Industry and Dresden experience has shown that in addition to TGSSC, this probe design is susceptible to fatigue failure from flow-induced vibration. Perry Nuclear Power Plant had a study completed which demonstrated that this design is able to fail due to an induced vibration called Karmen Vortex Street, where flow induced eddy vortices on the probe create vibrations. This vibration for this design is close enough in magnitude to the natural frequency to cause failure.

GE SIL 257 and 518

As stated above, GE SIL 257 only addressed the Dresden Feedwater, CDE and CPD probes. The CPD probes have been replaced with a different design as part of the Condensate Prefilter Modification. The installed design is similar to the designs suggested by General Electric in the BWR Water Sampling Requirements per GE Document 22A2748AA dated 1982. Information from GE SIL 518 could be applied to the feedwater probe, but the SIL was written to address Recirculation Loop Probes at BWR 6 plants. Dresden does not have this design of probe in our recirculation system, nor is the installed probe in the same relative location as the addressed probes.

Ability to Enter the Reactor Pressure Vessel

There are two probe locations that may be able to enter the reactor vessel in the event of probe failure. These are the feedwater and the Fuel Pool Filter Inlet. The feedwater probe will be caught in the Feedwater Spargers but may cause damage as seen at Dresden 2 and Perry. The Fuel Pool Filter Inlet probe may enter the reactor through the reactor well during refuel operations, or into the Spent Fuel pool during normal operations. Table 4 summarizes the sample probes at Dresden and evaluates their ability to enter the reactor vessel.

Table 4: Possibly Effected Downstream Components

Probe	Process Line	Downstream Components	Active / Passive	Reference
Waste Sample Tanks	2/3-2067(A-C)-4"-H	Respective WST	Passive	M-45 Sht 3
Floor Drain Sample Tank A	2/3-2032B-3"-LX	A FDST	Passive	M-45 Sht 3
Floor Drain Sample Tank B	2/3-2018A-3"-LX	B FDST	Passive	M-45 Sht 3
RWCU Filter Inlet	2-1207-8"-H	Valves, Flow Element, RWCU Dem	Passive	M-30
RWCU Filter Outlet	2-1214-8"-H	RWCU Demineralizer	Passive	M-30
RWCU Demineralizer A-C Outlet	2-1215C(A, B)-6"-H	Demineralizer Post-Strainer	Passive	M-30
Fuel Pool Demineralizer Inlet	2-1909-6"-L	FPC Demineralizer	Passive	M-50
Fuel Pool Filter Inlet	2-1906-6"-L	Fuel Pool or Reactor Well	Passive	M-50 and M-31
Condensate Demineralizer A-G Outlet	2-3305(A-G)-12"-L	Demineralizer Post-Strainer	Passive	M-17
Hotwell	2-3301A(B)-42"-L	Condensate Pumps	Active	M-15 Sh1, M-3350
Condensate Pump Discharge	2-3302-30"-L	Condensate Prefilter	Passive	M-15 Sh1, M-119 Sht 3
Original Condensate Pump Discharge	2-3302-30"-L	SJAE Intercondenser	Passive	M-15 Sht 1 Rev A
Condensate Demineralizer Effluent	2-3309-30"-L	Condensate Booster Pumps	Active	M-15 Sht 1
Final Feedwater	2-3204-24"-C	Feedwater Sparger, PCIVs	Passive	M-14
Feedwater Pump Discharge A-C	2-3201(A-C)-18"-C	HP Heaters, FW Regulating Valve	Passive	M-14 and M-117 Sht 3
U2 Recirculation Loop B	2-0201B-22"	Jet Pumps, Reactor	Passive	M-26 GE DWG 148F770
U3 Recirculation Loop A	3-0201B-22"	Jet Pumps, Reactor	Passive	M-3700

Other Downstream Components

Table 4 illustrates the downstream components that could be impacted by a sample probe failure. Each component or group of components was labeled as either active such as pumps, or passive such as valves, heat exchangers, demineralizers, and etcetera. Active component damage should be discovered through monitor of vibrations and would occur within a relatively short period of time. Passive component damage would be discovered through performance trending over longer periods of time.

Of the probes analyzed, the isokinetic probes are the only probes known to have failed. The feedwater probes have been recovered from the Feedwater Spargers and the Unit 2 CDE probe was removed from the casing of the 2D Condensate Booster Pump. One of the feedwater

probes has yet to be recovered from Unit 2. The two CPD probes that were installed in 1971 may still be located in the Condensate system. Their final locations are time dependent. If the probes failed prior the Condensate Prefilter modification, the probes may be in the Steam Jet Air Ejector Intercondenser Inlet waterboxes. The probes may cause fretting damage to the tube sheet, but are too large to pass through the tubes. The damage to this tube sheet would be difficult to detect with the trending tools available at this time, therefore visual inspections of the waterboxes will have to be performed. While there is no safety significance to this damage, the decline in performance will affect unit production capability in summer months. If the CPD probes were not carried into the SJAE Intercondenser or not removed with the piping for the Prefilter modification, the probes may have entered the Prefilter during Condensate system restart during D2R17 and D3R17. Any damage to the Prefilter will be determined during the next media replacement on each unit. If the probes had yet to fail when the prefilters were installed, they were removed from the system with the process piping and are no longer a concern.

Thermowells And Instrumentation Wells

Thermowells for resistance temperature detectors and probes for other instruments could be susceptible to the same flow induced fatigue mechanism that has been seen with the current sample probe design. There are three thermowells located in the same area of Feedwater piping as the Feedwater sample probes with the same access to the sparger area. There have been no thermowell failures in the Dresden Condensate and Feedwater Systems and a review of OPEX found no examples of thermowell failures in the Feedwater and condensate system. Several examples of cracking were found in other BWR OPEX for other systems. It would be considered prudent, however, to perform a review of the Condensate and Feedwater thermowells and other instrumentation that protrudes into the process stream for vulnerability to a flow induced vibration failure mechanism. This review shall be performed per ACIT 05.

Fuel Analysis

Nuclear Fuels (NF) and GE have analyzed fuel safety in November and December of 2003 respectively. The GE document, GE-NE-0000-0023-7311-RO, Lost Parts Analysis for Dresden Generating Station Units 2 and 3 Feedwater Sample Probes, includes detailed discussion on the potential locations of the probe under the conservative assumption the probe has exited the sparger. The NF document, ECN 345672, Dresden Unit 2- Lost Parts Evaluation for Isokinetic Probe and Sparger Disks, includes the conservative assumption that the pieces missing from the sparger, due to wear and rubbing by the probe, came off in three 1-inch diameter pieces. These evaluations address the safety and operational concerns for plant operation without recovery of the lost parts from the reactor vessel.

The evaluations are consistent and conclude that safe reactor operation will not be compromised with the presence of the potential lost parts in the reactor vessel. There is no safety concern for flow blockage to the fuel bundles, interference with the scram function, corrosion or adverse chemical reaction with other reactor materials, interference with Nuclear Boiler or Neutron Monitoring Instrumentation, or interference with RWCU or SDC isolation valves. There is moderate operational concern related to an increased probability of damage to one SDC pump, partial bottom head drain plugging and recirculation system performance, all of which can be detected and mitigated through existing online monitoring and procedures. NF also discusses the low to moderate risk for fuel fretting, which would be detected through the Fuel Reliability Indicators that are monitored weekly.

11.0 Risk Assessment

The consequences of this event had minimal impact on reactor safety. The feedwater sparger is not classified as safety-related. It also is not under the jurisdiction of ASME Boiler and Pressure Vessel Code. However, damage to the feedwater sparger may result in:

- Feedwater flow impinging directly on the vessel surface, eventually leading to thermal fatigue in the vessel cladding.
- Any damage caused to the feedwater sparger may generate lost parts.
- Nuclear Fuel Management performed a lost parts evaluation for U2 and U3 per NFM-MW:01-0351 and NFM-MW:02-0401 respectively and concluded there were no safety concerns associated with the unit operation.
- General Electric performed a lost parts analysis as documented in GE-NE-0000-0023-7311RO, titled Lost Parts Analysis for Dresden Generating Station Units 2 and 3 Feedwater Sample Probes, and dated 12/03. The analysis concluded that safe reactor operation would not be compromised with the presence of the potential lost part in the reactor vessel.

Therefore the safety significance of these events was considered minimal.

12.0 Programmatic/Organizational Issues:

A knowledge deficiency was identified in the retrieval of industry experience within Engineering.

Sample system boundaries, roles and responsibilities require clear establishment between Chemistry and Engineering.

13.0 Previous Events:

Table 5: OPEX Events Regarding Failed Sample Probes

Plant, Unit(s), Event Date	OPEX Title	Description
Two Unidentified BWRs 12/30/77	GE Nuclear SIL 257, Improved Feedwater Sample Probe	Two operating BWRs reported Feedwater sample probes broke off on the process side during reactor plant operation and became lodged in downstream valves. The **probable failure mechanism was determined to be transgranular stress corrosion cracking.** SIL recommended improved design.
An unidentified GE BWR/6 located in the United States	GE Nuclear SIL 518, Improved Recirculation Water Chemistry Sample Probe.	A Recirculation sample probe broke off during hot preoperational testing and was recovered from inside the reactor vessel. **The cause of the failure was determined to be a high cycle fatigue caused by flow-induced vibration.** SIL 518 recommended shorter probe design less susceptible to high cycle fatigue.
Braidwood Nuclear Station, Unit 2, Spring 1990	No OPEX discovered. Braidwood provided information on the event including a Sargent and Lundy letter dated April 24, 1990.	Sample probe 2AX-PS084 was discovered resting against the cage of one of the Feedwater Regulation valves during an inspection performed in the first refueling outage, A2R01, March 2003. **The failure mechanism was identified as flow induced mechanical fatigue based on the clean break of the sample probe and high line velocity.** Engineering recommended not reinstalling these probes and that the additional probes be inspected/modified at the next outage to reduce the chance of fatigue failure.
Browns Ferry 3 07/23/92	EN 296-920723-1 Broken Feedwater Sample Probe Found During Shutdown	Broken Feedwater sample probe was discovered broken off just past the weld at the inside junction of the probe and collar. **Probable failure mechanism appeared to be fatigue.**
Perry Nuclear Power Plant 04/30/92	No OPEX discovered. GE provided information that event occurred after D2R18 and Dresden obtained copy of Root Cause from PNPP.	Feedwater isokinetic sample probe found protruding through first nozzle of a feedwater sparger. A second hole also discovered in the fourth nozzle. Probe was ¾" O.D. schedule XXS SS pipe 15 ¼" in length. **Perry rejected the SIL 257 probable cause of transgranular stress corrosion cracking and determined the Root Cause for failure was high cycle fatigue based in flow induced vibration fatigue failure described for a recirculation system probe in SIL 518.** Probe replaced with design based on the GE SIL-518 recommended shorter probe design.
LaSalle Nuclear Station Unit 1 4/12/94	No OPEX discovered. LaSalle provided this information to assist in the root cause investigation.	*Sample probe in a reactor recirculation loop broke off, traveled through a jet pump,* and was found in the reactor vessel. The jet pump incurred no damage. The failure mechanism was high cycle bending fatigue. A new probe was designed and installed per GE SIL 518 recommendations.
Quad Cities Nuclear Station, Units 1and2 1996-1997	No OPEX written. QC provided this information to assist in the root cause investigation.	Feedwater probes were inspected in 1996 and 1997 as part of a SIL review that identified GE SIL 257. After 22 years of operating at 9.8 Mlbs/hr feedwater flow, the probe on U1 was found broken at the pipe attachment but was 100% retrieved from the pipe. The U2 probe was found intact, but crumbled into pieces as it was removed from the pipe. Probes were replaced per GE SIL 257 and a 10-year replacement PMID was assigned.
Grand Gulf 02/06/01	EN 416-010206-1 Heater Drain Pump Sample Line Failure	Leak discovered at a line break involving a sample probe on the heater drain system. The probe was a ¾" SS, schedule XXS pipe welded to a SS valve on one end and to a ¾" 6000-pound carbon steel half coupling on the other end. The break was on the valve side of the pipe at the toe of the bi-metallic weld. **Root Cause of failure was determined as high cycle low stress fatigue failure caused by operating conditions that induced vibration levels creating stresses sufficient to initiate a crack and cause it to propagate to failure.** An engineering report was issued to install a shortened probe as recommended by GE SIL 518
Braidwood Nuclear Station, Unit 1 04/25/03	OE16411 and OE17453 - The 1B Feedwater Pump Suction Strainer was Found Damaged During Inspection	1B FW pump suction strainer was inspected during the 2003 Spring refueling outage as part of an investigation regarding material discovered in the steam generators (SG). Lodged in the inspection port and strainer screen was a 17"-long, ¾"-diameter, extra heavy wall, 10 lb_m section of pipe. This pipe was believed to be part of an upstream sample probe that fractured at the weld connection. The Unit 2 probe installed in the FW pump suction header was discovered at the suction strainer of 2B FW pump when the pump was being installed as a follow-up corrective action after the above event. The damage to the strainer was not as severe as seen on Unit 1.

The industry event review identified multiple events that could have prevented some, or possibly all, of the Dresden probe failures. A review of the available OPEX during D2R17 and D3R17, when the original feedwater probes were discovered missing, would have challenged the mode of failure and the SIL 257 design. These missed opportunities have been identified as Causal Factor 2.

14.0 Other Issues:

The following CRs were subsequently added to this RCR and are dispositioned as follows:

CR 187258, titled U2 and U3 Isokinetic Sample Probes Missing in Condensate and FW Systems, was generated to document three isokinetic probes are missing within the Condensate and FW systems. One probe was assumed to be located in the U3 Feedwater sparger and the other two probes are assumed to be located in the units' respective SJAE Intercondenser water box. There is also a probe in the U3 Condensate Demineralizer Effluent that has likely failed; however status at this time is unknown.

Therefore no additional actions are required.

CR 189800, titled Foreign Material in 150-Degree U3 Feedwater Sparger, was generated to document the identification of a second sample probe being identified in the U3 Feedwater sparger.

Both sample probes were recovered from the U3 RPV Feedwater Sparger in D3M10 per WO 641422-01.

Therefore no additional actions are required.

CR 187492, titled U2 Non-Conservative Analysis and Ineffective Corrective Actions, was generated due to the Nuclear Fuel Management document NFM-MW:01-0351, titled Dresden Unit 2 Lost Parts Evaluation for Isokinetic Sample Probe, dated 11/02/01, did not address vessel component damage. In 2001, the evaluation led to the conclusion that the Feedwater Sparger would not be affected as a result of the missing sample probe intrusion. The Feedwater Sparger damage identified in D2R18 proved the evaluation was inadequate and the conclusion incorrect.

General Electric performed a lost parts analysis as documented in GE-NE-0000-0023-7311RO, titled Lost Parts Analysis for Dresden Generating Station Units 2 and 3 Feedwater Sample Probes, and dated 12/03. The analysis addressed potential damage to reactor vessel components and potential impairment of the Recirculation system performance. The analysis concluded that safe reactor operation would not be compromised with the presence of the potential lost part in the reactor vessel.

Therefore no additional actions are required.

CR 189992, titled U3 Final Feedwater Isokinetic Probe Failed, was generated to document the Ultrasonic Testing of the Feedwater line 3-3204-24"-C at the sample probe location confirmed the Feedwater sample probe installed during D3R17 had failed.

The sample probe was recovered from the U3 RPV Feedwater Sparger in D3M10 per WO 641422-01.

Therefore no additional actions are required.

CR 190413, titled D2 FW Sample Probe Missing, was generated to document the Ultrasonic Testing of the Feedwater line 2-3204-24"-C at the sample probe location confirmed the Feedwater sample probe installed during D2R17 had failed.

Operability Determination 03-015, Revision 2 was completed. Recovery is planned during D2R19 per WR 124348.

Therefore no additional actions are required.

CR 190571, titled D2 FW Sample Probe Missing, was generated to document the Ultrasonic Testing of the Feedwater line 2-3204-24"-C at the sample probe location confirmed the Feedwater sample probe installed during D2R17 had failed.

This CR documents the impact of the missing sample probe on chemical analysis. A GE analysis was conducted and the report of the Dresden Unit 2 and 3 Broken Feedwater Probe Justification for Continued Operation indicates that both plants can continue to operate for the balance of the fuel cycle without installation of new feedwater probes. A shorter probe design is recommended for a replacement for the missing probes. The consequence with respect to Chemistry is soluble impurity sample results will not be affected. Insoluble impurity results can be expected to be systematically 10-20% low during normal operations and higher during transient operations. GE assesses that, based on historical values of feedwater iron; a 20% bias for these measurements would have no impact on fuel performance.

Therefore no additional actions are required.

CR 191618, titled P&IDs Not Revised to ID Sample Probe after Prefilter Mod, was generated to document the failure to properly revise all drawings associated with the modification process. The affected Piping and Instrument Diagrams were not revised to reflect Modified Condensate Pump Discharge (CPD) sample probe during the U2 and U3 Condensate Prefilter Modifications. This CR is currently awaiting supervisory review.

Attachments:

Attachment 1	Event and Causal Factors Chart
Attachment 2	Barrier Analysis
Attachment 3	Summary of Recovered Probes
Attachment 4	Exelon PowerLabs Field Report on the Evaluation of Chemistry Sampling Probes from Dresden Station
Attachment 5	Root Cause Investigation Charter, Rev. 1
Attachment 6	LS-AA-125-1001, Attachment 14, Root Cause Report Quality Checklist

References:

1. GE SIL 257, Improved Feedwater Sample Probe, issued 12/30/77.
2. GE SIL 518, Improved Recirculation Water Chemistry Sample Probe, issued 08/06/90.
3. GE Drawing 921D233, titled Sample Probe for Feedwater Control System, Revision 1, dated 04/26/68
4. SEN 204, Water Chemistry Induced Fuel Leaks, issued 09/20/99
5. Exelon PowerLabs Field Report on the Evaluation of Chemistry Sampling Probes from Dresden Station, Report Number DRE-90509.
6. General Electric document GE-NE-0000-0023-7311RO, titled Lost Parts Analysis for Dresden Generating Station Units 2 and 3 Feedwater Sample Probes, dated 12/03
7. Dresden Operability Determination 03-015 Performed to analyze the effects of the missing feedwater probe for the Unit 2 Start Up following D2F40 outage.

Attachment 2
Barrier Analysis

Failed or Ineffective Barrier	How Barrier Failed	Why Barrier Failed	Corrective Action to Restore Barrier to Effectiveness
SIL/OPEX Review and Response Process	SIL 257 and 518 not dispositioned. Perry, Brown's Ferry and Grand Gulf fatigue failures not considered.	-General lack of knowledge of OPEX search methodology. -Historical failure to properly disposition SIL and OPEX	- Review and document the disposition of Training Request 03-2069 and 03-2070 to perform training on the retrieval of web based industry experience to Maintenance Work Planning and Engineering personnel. Initiate additional actions as appropriate. - Regulatory Assurance to develop a database of GE SILs and which documents the Dresden disposition and coordinate a review of open SILs per LS-AA-115. Initiate additional actions as appropriate -Regulatory Assurance to conduct a Check-in Self Assessment on the Site OPEX program per LS-AA-126-1005.
Corrective Action Program Management Review Effectiveness	CR's 81081 and 127346 did not address why SIL 257 was not dispositioned previously or require additional extent of condition SIL or OPEX review.	-Weak wording of OPEX action tracking item allowed closure to one phone call. -No investigation into why the SIL was not implemented.	- Regulatory Assurance to generate information package for Department Heads and Branch Managers related to the OPEX review process and resources.
System and component ownership for sample probes.	There was no individual responsible for the maintenance of sample probes.	-Failure to designate and owner for sample probes issues. -General lack of knowledge of sample probes and sample systems between the system and the sample panel.	- Establish memorandum of agreement between Chemistry and Engineering to define the boundaries, roles and responsibilities for Chemistry sample systems. -TR 03-2071 was generated develop sample system theory and design training for Chemistry Technicians and Managers.
Design Documentation System associated with the Feedwater and Condensate systems was not up-to-date.	-Sample probe drawing was not updated after SIL 257 installation. -All Condensate drawings were not properly updated after 1971 original probe installation or after pre-filter mod.	-Did not properly identify drawings that should be changed in the design modification process.	Review the U2 and U3 Feedwater and Condensate design document sample probe information and revise as appropriate. The review shall encompass P&IDs, Condensate Prefilter modification and probe detail drawings.
Design Specifications	-Some FW and CDE probes were constructed of schedule 160 versus the design specification of XXS. -Measurements of retrieved broken probes indicate that original probes were too long.	Original probes provided for installation in 1971 were not constructed to specifications and this was not discovered before installation.	This was a historical barrier that failed. The current Maintenance Planning process has the required steps to prevent recurrence.

Attachment 3
Summary of Recovered Probes

Probe ID	Probe Location	Schedule	Wall Thickness/ ID (in)	Length to Collar per Design (in)	Length As found (in)	Comments
0	U2 CDE	160	0.219/0.612	16	Indeterminate	Removed from 2D Condensate Booster pump. Probe was wrapped around impeller shaft and mangled.
1	U2 Feedwater	XXS	0.308/0.434	13	10 ¾	Removed from the N4C 240° Feedwater sparger, capped end facing end of sparger. Caused damage wearing 3 holes through sparger. The sample entry holes drilled in the probe were 1/16, 1/16 and 1/8 inches in diameter instead of the three 1/8th inch holes called for by design.
2	U3 CDE	160	0.219/0.612	16	16 $^3/_8$	Removed intact during D3M10. Had 3/8 inch beveled cap.
3	U3 Feedwater	XXS	0.308/0.434	13	~ 13	Removed from N4B 150° Feedwater sparger. No damage observed. Open end facing end of sparger.
4	U3 Feedwater	160	0.219/0.612	13	14 ¾	Removed from N4B 150° Feedwater sparger. No damage observed. Open end facing end of sparger. 3/8 inch beveled cap.

There are some dimensional differences with respect to the length, size of beveled caps, hole separation distances and material of the probes thought to be the original 1971 probes. In addition the sizes of the holes drilled on sample number 2 varied from $^1/_{16}$ to $^1/_8$ inch. All other identifiable holes in the recovered probes were $^1/_8$ inch.

It is believed from the information above that at least some of the original probes were improperly manufactured from schedule 160 stainless steel instead verses the design specification of XXS. Probe 2 was removed intact from the condensate system and is believed to be the original probe installed in 1971.

Attachment 4

PowerLabs Field Report on the Evaluation of Chemistry Sampling Probes from Dresden Station

Field inspections were performed on four stainless steel chemistry sampling probes at Dresden Station on 12/13/2003. The samples had the following identifications.

Sample 1 – Failed probe removed from the Unit 2 sparger during D2R18
Sample 2 - Intact probe removed from Unit 3 CB pump suction piping in D3M10
Sample 3 – Failed probe removed from Unit 3 sparger during D3M10
Sample 4 – Failed probe removed from Unit 3 sparger during D3M10

The purpose of the inspections was to determine the cause of the failures.

1. CONCLUSIONS

Samples #3 and #4 failed by fatigue cracking, most likely due to high cycle/low stress loading.

The failure mechanism for Sample #1 could not be determined due to severe mechanical damage and wear of the fracture surface.

Samples #1 and #3 were fabricated from pipe with a smaller inner diameter (~$^{7}/_{16}$") than samples #2 and #4 (~$^{5}/_{8}$"). The outer diameter on all the samples was ~1". The end caps on samples #1 and #3 also had smaller bevels ~$^{1}/_{8}$" than the $^{3}/_{8}$" bevels on samples #2 and #4.

The two holes that were farthest from the cap on Sample #1 had a ~$^{1}/_{16}$" diameter. All other holes were ~$^{1}/_{8}$" diameter.

2. RESULTS AND DISCUSSION

The overall length of sample #1 measured ~10 ¾". Both ends and several areas along the length of the sample exhibited severe wear. The worn regions were smooth and had a polished appearance. The fracture surface had been completely removed by the wear.

Sample #2 was intact. The external surface of the stainless tube was covered with a light layer of orange deposits. The length from the probe tip to the carbon steel collar measured ~16 $^{3}/_{8}$". The pipe to collar crevice was open (i.e., was not sealed by a fillet weld).

Sample #3 measured ~13" long from the cap tip to the fracture surface. No weld metal was observed on the sample. The fracture surface was relatively flat and exhibited a faint beach mark pattern. Beach marks are indicative of a progressive cracking

mechanism, such as fatigue. Ratchet marks suggested there were several initiations on one side of the pipe.

A metallurgical sample through the Sample #3 fracture was prepared and examined using a field microscope. The fracture propagated in a non-branching, transgranular manner that was typical of fatigue. No weld metal was observed in the examined section.

Sample #4 measured ~14 ¾" long. No weld metal was observed around the fracture edge. The fracture was relatively flat and faint beach marks were observed. The fracture texture suggested a single initiation site.

The outer diameter on all four probes measured ~1". The inner diameter on samples #1 and #3 measured ~$^{7}/_{16}$". The inner diameter on Samples #2 and #4 measured ~$^{5}/_{8}$".

The two chemistry sampling holes that were farthest from the end cap on Sample #1 measured ~$^{1}/_{16}$" diameter. The remaining hole on Sample #1 and all the holes on the other probes measured ~$^{1}/_{8}$" diameter.

The end cap bevels on Samples #2 and #4 measured approximately $^{3}/_{8}$" long. The end cap bevels on Samples #1 and #3 measured ~$^{1}/_{8}$" long.

Reported by: Jim Chynoweth, 12/13/2003
Senior Metallurgical Engineer
Tech Services West

Attachment 5
Root Cause Investigation Charter
Revision 1

CR# 183901
Subject: U2 Feedwater Sparger Damage and U3 Foreign Material Intrusion

Sponsoring Manager: Gordon Dorsey
Team Leader: Linda Dyas

Team Investigator(s):

RCR Lead Investigator: Linda Dyas, Full Time - Days

Team Members:
Frank P. Polak - Full Time Days (RC Qualified)
Bob Testin, Full Time - Nights (RC Qualified)
Amy McMartin, Full Time - Nights
Dan Malauskas, Full Time - Days
George Baxa, Full Time - Nights
Jim Chynoweth, Exelon PowerLabs, Full Time - Days
Jeff Rund, Full Time - Nights
Harold Herzog, GENE, Full Time - Days
Bob Geier, Part-time - Days

Scope:

During the D2R18 inspection of the feed water sparger three holes in two of the nozzles were discovered. Initially it was assumed that the missing FW sample probe that was discovered during D2R17 was the possible foreign material that caused the damage to the sparger. Foreign material was retrieved from the sparger and was identified as this FW Chemistry Sample Probe. CR 189800 was generated to document the identification of FME in the U3 Feedwater Sparger during D3M10.

This Root Cause Investigation will determine the cause of the nozzle damage discovered during the feed water sparger inspections. This will include the failure mechanism of the FW Sample Probes and the failure mechanism of the damage done by the probe to the U2 FW sparger. Additionally, the investigation will evaluate the extent of condition and initiate the Corrective Action to Prevent Recurrence (CAPR). An extensive review of industry experience, OPEX and maintenance history will be performed. Emphasis will be placed on foreign material not retrieved and existing sample probes in the system.

The RCR will address the issues identified in CRs titled:

CR 190413, Modified U2 Sample Probe Missing
CR 189800, Foreign Material in 150 Degree U3 Feedwater Sparger
CR 189992, U3 Final Feedwater Isokinetic Probe Failed
CR 187492, U2 Non-Conservative Analysis and Ineffective Corrective Actions

CR 187258, U2 and U3 Isokinetic Sample Probes Missing in Condensate and FW Systems

An NER and OE will be generated to communicate this event.

Interim Corrective Actions:

- Partial inspection of the U2 FW Sparger has been performed one FW probe was removed.
- EVT-1 inspection was performed on the vessel wall, no recordable indications noted.
- Repairs to the U2 sparger were completed.
- The probe will be retrieved from the U3 FW sparger.
- Failure analysis of the U3 probes will be performed by Exelon PowerLabs

Root Cause Report Milestones:

1. Event Date 10/14/03
2. MRC Screening Date 11/03/03
3. Completion of Charter Revision 0 11/05/03
   Completion of Charter Revision 1 12/13/03
4. MRC Update 11/19/03
5. Report provided to Sponsoring Senior Manager for Approval 12/15/03
6. Report provided to Site CAPCO for distribution to SVP, Plant Manager, RA Manager 12/15/03
7. Final Root Cause Investigation MRC Presentation Date 12/16/03

Team meeting location: Room 224, Ext 3333

Prepared By: Frank P. Polak, 12/13/03

Reviewed by DCAPCO: Frank P, Polak, 12/13/03

Approved By: Gordon Dorsey, 12/13/03
(Sponsoring Senior Manager/Director) Date

GE Proprietary Information

Attachment 6
Supplemental Information to
"Dresden 2 and 3 Broken Feedwater Probe – Justification for Continued Operation" dated 12/15/2003

This supplement to the GE report "Dresden 2 3 Broken Feedwater Probe – Justification for Continued Operation" contains results of calculations for other sample probes, as requested by Exelon, reference 1, on 12/17/2003.
The calculation methods used in this supplement were the same as those used in GE-NE-0000-0024-2731.

3. SAMPLE PROBE DATA

Case	Schedule	Length from End to Inboard Weld, Inches	Water Velocity, FPS	Probe Natural Frequency, Hertz, fn	Maximum Vortex Shedding Frequency, Hertz, fs	3.1. fn
2	160	14-3/4	21.67	134	116	.87
3	XXS	13-1/8	26.1	163	140	.86
4	XXS	16-1/8	11.28	109	32	.3
5	XXS	14-3/4	21.67	129	116	.9
6	XXS	13-1/8	21.67	163	116	.71

3.1.1. General Notes

All probes are ¾" nominal pipe size.
SIL 257 recommends an inner and outer seal weld to sample probes. With one seal weld, some uncertainty is introduced into the natural frequency calculations because of possible dynamic effects of the external piping and valves, about which information is typically not readily available. Consideration of these external piping / valve / supports effects in an engineering calculation would require extensive finite element analysis, and is beyond the scope of this evaluation. It is noted, however, that these effects may cause some variance in the calculated structural natural frequencies in this calculation for those cases, and these differences are not expected to be very large. Those cases with one seal weld (Pre-SIL 257) are reported in the DIR as 2 and 5.

For those probes with two seal welds, greater isolation is provided from the external

Based on the ASME Boiler and Pressure Vessel Code, Appendix N titled "Dynamic Analysis Methods", paragraph N-1324, values of the ratio of vortex shedding frequency to probe natural frequency, provided in the last column, below 0.76 are considered acceptable. However, other documents provide more stringent criteria. The closer the ratio of vortex shedding frequency to probe natural frequency is to 1, the more likely a resonance lock-on condition is to occur.

3.1.2. References

1 Task Design Input Request, Missing Feedwater Probe Analysis D2 and D3, 12/17/2003

ATTACHMENT 14
Root Cause Report Quality Checklist
Page 1 of 2

Condition Report Number: 183901-18

A. Critical Content Attributes	YES	NO
1. Is the condition that requires resolution adequately and accurately identified?	X	
2. Are inappropriate actions and equipment failures (causal factors) identified?	X	
3. Are the causes accurately identified, including root causes and contributing causes?	X	
4. Are there corrective actions to prevent recurrence identified for each root cause and do they tie DIRECTLY to the root cause? AND, are there corrective actions for contributing cause and do they tie DIRECTLY to the contributing cause?	X	
5. Have the root cause analysis techniques been appropriately used and documented?	X	
6. Was an Event and Causal Factors Chart properly prepared?	X	
7. Have cause Codes been identified for equipment problems, human performance, and organizational weaknesses, and have all applicable trend codes been identified and documented in the report and entered into the Condition Report Trend/Cause Panel TIMA017 in Action Tracking?	X	
8. Does the report adequately and accurately address the extent of condition in accordance with the guidance provided in Attachment 3 of LS-AA-125-1003 Reference 4.3?	X	
9. Does the report adequately and accurately address plant specific risk consequences?	X	
10. Does the report adequately and accurately address programmatic and organizational issues?	X	
11. Have previous similar events been evaluated?	X	

ATTACHMENT 14
Root Cause Report Quality Checklist
Page 2 of 2

B. Important Content Attributes	**YES**	**NO**
1. Are all of the important facts included in the report?	X	
2. Does the report explain the logic used to arrive at the conclusions?	X	
3. If appropriate, does the report explain what root causes were considered, but eliminated from further consideration and the bases for their elimination from consideration?	X	
4. Does the report identify contributing causes, if applicable?	X	
5. Is it clear what conditions the corrective actions are intended to create?	X	
6. Are there unnecessary corrective actions that do not address the root causes or contributing causes?	X	
7. Is the timing for completion of each corrective action commensurate with the importance or risk associated with the issue?	X	
C. Miscellaneous Items		
1. Did an individual who is qualified in Root Cause Analysis prepare the report?	X	
2. Does the Executive Summary adequately and accurately describe the significance of the event, the event sequence, root causes, corrective actions, reportability, and previous events?	X	
3. Do the corrective actions include an effectiveness review for corrective actions to prevent recurrence?	X	
4. Has an Operating Experience database search been performed to determine whether the problem was preventable if industry experience had been adequately implemented?	X	
5. Are the format, composition, and rhetoric acceptable (grammar, typographical errors, spelling, acronyms, etc.)?	X	

Linda L. Dyas
Lead RCR Investigator

Bob Testin
Evaluator / Root Cause Investigator

Amir Shahkarami
Site Engineering Director

Frank P. Polak
Department CAPCO

Gordon Dorsey
Mohammad Molaei
Sponsoring Managers

George Baxa
Amy McMartin
RCR Investigators

ATTACHMENT 5B

Dresden Nuclear Power Station, Units 2 and 3
Operability Evaluation 03-015, Revision 2
"Feedwater Sparger"

1.0 ISSUE IDENTIFICATION:

1.1 CR #: 190413

1.2 OpEval #: 03-015 Revision: 2

General Information:

1.3 Affected Station(s): Dresden

1.4 Unit(s): Two

1.5 System: RX/0202

1.6 Component(s) Affected: Feedwater Sparger

1.7 Detailed description of what SSC is degraded or the nonconforming condition and by what means and when first discovered:

This operability determination is being performed to address the potential consequences of a lost feedwater sample probe in the Unit 2 Feedwater system, after discovery of a missing probe on 12/11/03.

BACKGROUND:

In response to a GE recommendation in March 25, 1971 on expanded water chemistry programs, Dresden installed three sample probes on each unit in the condensate and feedwater system, for a total of six probes at the Station. These probes were installed as an enhancement to the Water Chemistry Program. The purpose of these probes was to introduce a limited feedwater sampling and analysis program to determine condensate demineralizer performance and feedwater corrosion product levels.

In 1977, GE issued SIL 257 regarding failure mechanism of these isokinetic probes installed in the feedwater and condensate systems. This SIL stated that the failure mechanism was stress corrosion cracking at the exposed dissimilar metal weld. At that time several plants had identified these probes to be missing.

In September of 1999, INPO SEN 204 was published as a result of a River Bend water chemistry incident. The Chemistry Department at Dresden station initiated a work request in response to SEN 204 to inspect the Final Feedwater isokinetic sample probe on each unit during the next available outage, D2R17 and D3R17 respectively. Inspections during D2R17 & D3R17 found both the Unit 2 & Unit 3 Final Feedwater probes to be missing. They were replaced per GE SIL 257. However, the failed probes were not recovered. The conclusion was reached that the final barrier to prevent the probe from entering the reactor would be the feedwater sparger. Nuclear Fuels evaluation documented in NFM-MW:01-0351 indicated that this probe would not enter the reactor vessel through the sparger. The evaluation also stated that if the probe was to enter the reactor pressure vessel it would be carried to the lower vessel plenum, where the size and weight of the probe would not allow it to be swept by the flow toward the inlet orifices of a fuel support casting. Nuclear Fuels (NF) concluded that there were no fuel or control rod drive safety concerns associated with operation of Unit 2 with the missing final feedwater isokinetic sample probe.

DESCRIBE THE PERRY EVENT BRIEFLY

UNIT 2 DAMAGE:

On October 14, 2003, during Dresden Unit 2 Feedwater Sparger inspections, damage was discovered on two of the nozzles on the N4C Sparger located at 240°. Upon further boroscopic inspection, an isokinetic probe was located inside of the sparger and

removed. After inspection of the probe, it was identified as the feedwater sample probe that was discovered to be missing during D2R17. The probe is stainless steel and approximately 11 ½" in length, 1" O.D. and ½ " I.D. There were marks along the length of the probe that were consistent with the holes in the nozzles of the N4C Feedwater Sparger. The failure mechanism of the sparger is most likely erosion, wear, and rubbing that caused the probe to fret a hole in the nozzle located third from the T-box. The probe then began to rub against the second nozzle from the T-box causing two holes, one on either side of the second nozzle. The holes caused by the probe in the sparger were evaluated by GE to be inconsequential to the function of the feedwater system.

CORRECTIVE ACTIONS:
The damage to the N4C Feedwater Sparger was repaired/modified in accordance with GE recommendations (FDDR EB2-0094 Feedwater Sparger Nozzle Repair). An EVT-1 inspection of the reactor wall near the damaged sparger proved that no degradation had occurred due to this incident. An Engineering Evaluation (EC # 345690) has been performed to address the impact of the missing sparger pieces on Systems connected to the Reactor Recirc system.

D3M10:
The Unit 3 Final Feedwater probe was discovered to be missing during D3R17 inspection. As part of the periodic in-vessel visual inspections, an external visual examination of the Unit 3 Feedwater Spargers was performed during D3R17 and no indications were noted. Internal boroscopic inspection of the Unit 3 spargers during D3M10 revealed two probes and a washer inside of the 150° feedwater sparger – no internal damage was seen. The final feedwater probe installation point was inspected using UT and the probe installed during D3R17 (2002) was discovered to be missing. It is believed to be the additional probe that was located inside of the U3 150° feedwater sparger.

UNIT 2 EXTENT OF CONDITION:
Thursday December 11, 2003, at approximately 2335, Unit 2 was shutdown due to the Stator Cooling Water System temperature unexpectedly increasing. During this unexpected unit shutdown, the NDE group was able to perform a UT inspection of the U2 final feedwater probe installation point. It was determined that the probe installed during D2R17 (2001) was missing. This probe is likely to reside in a Unit 2 feedwater sparger.

As part of the periodic in-vessel visual inspections, an external visual examination of the Unit 2 Feedwater Spargers was performed during D2R18, damage was discovered on the 240° (N4C) sparger. This sparger was examined internally in the area of the damage and a final feedwater probe was located. A final feedwater probe was removed from the sparger and repairs were made per the GE plan. It has since been discovered that the newly installed (2001) Unit 2 Feedwater Probe is missing. It is possible that the missing probe has been in a feedwater sparger for more than one operating cycle since a full extent of condition review was not performed during D2R18. This Operability Determination will examine the consequences of a failed probe in the Unit 2 feedwater system.

Note that both the evaluations, and the industry experience such as Perry and Dresden, show that the most likely place for this probe to rest is the feedwater spargers. Prior successful results of the Local Leak Rate Tests for the Containment Isolation Valves between the missing probe location and the reactor vessel showed that the probe was not wedged in these valves. The feedwater check valves are a tilting disc (Chapman model 973) style valve that is streamlined. The removable trim is recessed into the valve body and will not create an edge into which the probe could lodge.

Therefore, for the purpose of this evaluation, it will be assumed that the probe is inside the sparger and will cause damage similar to that discovered at Perry Nuclear Power Plant and during D2R18.

The possibility exists that damage will be found in Unit 2 similar to that found during D2R18. Based on Unit 3 and industry experience, the probe is equally likely to be carried to the end of the sparger to a low flow area and rest in the sparger. (In 4 instances of industry experience where a probe got into a sparger, 2 rested in the sparger and 2 caused similar damage but were captured within the sparger.) In the two known cases where probes have become entrained in the feedwater flow and damaged spargers, flow has lifted the probe toward the flow nozzles and formed leak paths in an upward direction. As a result, the cool feedwater has been adequately mixed and not impinged directly on the reactor vessel wall.

While this damage can be costly and time consuming to repair, it does not affect the safety function of the spargers. Nuclear Fuels has also evaluated the possibility of the probe entering the vessel and concludes that there are no fuel or control rod drive safety concerns associated with the operation of the Unit.

The primary function of the feedwater sparger is to distribute the feedwater uniformly within the reactor so that it will form a homogeneous mixture with the saturated separator return flow and the recirculation pump flow. If the probe were to bore a hole in the Unit 2 Sparger, this potential damage that could occur and the subsequent repair would be bounded within the GE evaluation completed for the Unit 2 feedwater sparger repairs made during D2R18. There is concern with the potential damage to the sparger and the subsequent direction of flow following the damage. There could be a change in the direction of the feedwater flow causing impingement onto the vessel surface; this could lead to thermal fatigue in the vessel cladding. Industry experience from D2R18 and Perry indicate that the sample probe would be lifted in the flow stream towards the outlet nozzles causing damage to the top surfaces of the sparger. This is not expected to direct flow onto the vessel wall.

EVALUATION:

2.1 Describe the safety function(s) or safety support function(s) of the SSC. As a minimum the following should be addressed, as applicable, in describing the SSC safety or safety support function(s):

- It is not believed that the probe will exit the sparger based on D2R18, D3M10 and Perry experience. Based upon engineering judgment due to size and geometry of the probe, if it were to exit the sparger, it would not become located in the steam path (during normal or accident conditions) of the reactor vessel, the steam lines, and associated systems but rather enter the Recirculation System or the lower plenum. Therefore, this probe does not impact the safety functions of the Main Steam Isolation Valves (MSIVs), Automatic Depressurization System (ADS), High Pressure Coolant Injection (HPCI), and Iso. Condenser. The Core Spray System has its own injection header and is not affected by any potential migration path of the lost part(s). The safety functions that could be impacted by the probe are limited to the Reactor Vessel lower head area (including the reactor pressure boundary), the fuel core, the Standby Liquid Control System (SBLC) injection sparger and associated instrumentation, and the Recirculation System (including the systems taking a suction from the Recirculation System: Reactor Water Cleanup (RWCU) and Shutdown Cooling (SDC). Other safety functions being evaluated are:

  1) Fuel bundle flow blockage and fuel damage due to overheating of the fuel cladding,

2) Control rod operation,
3) Corrosion or adverse chemical reactions with other reactor materials,
4) Interference with the RWCU System and
5) Interference with the Nuclear Boiler or Neutron Monitoring Instrumentation.

In addition, there are operational concerns with:

1) Potential fuel fretting,
2) Potential blockage of the reactor vessel bottom head drain, and
3) Potential for impairment of the Recirculation System performance.

GE Report GE-NE-0000-0023-5200-R0 for QC1 evaluated these aspects and determined that the presence of the lost part(s) will not compromise safe reactor operation. This report was reviewed and determined to be bounding for the case of the probe exiting the Feedwater Sparger. Provide some further discussion of geometry of the probe and why bounded by geometry discussed in the GE report – same materials, etc, . . .

- Does the SSC receive/initiate an RPS or Engineering Safety Features, ESF, actuation signal?

  No, assuming the probe does not exit the feedwater sparger then the affected component in this operability determination is the feedwater spargers, which are passive devices that evenly distribute feedwater flow entering the reactor. There are no RPS or ESF functions associated with the feedwater system. The system, with the exception of the containment isolation valves and the associated piping, is classified as Non-Safety related and it is assumed to perform no safety function during a postulated accident.

  In the unlikely event that the probe were to exit the sparger, the probe has the potential to affect several SSC's that receive a RPS or ESF actuation signal. These include:

  1) Containment isolation valves, Shutdown Cooling, LPCI and RWCU;
  2) CRD System, preventing rod insertion;

- Is the SSC in the main flow path of an ECCS or support system?

  Yes, a portion of the feedwater piping, including the sparger, is in the return flow path for the HPCI injection. As non-safety related devices, the feedwater spargers play no active role in this safety function. HPCI injection would be performed despite any damage to the sparger.

  If the probe were to exit the feedwater sparger, the Low Pressure Coolant Injection (LPCI) System could be impacted if the Recirculation Pump discharge valve could fail to close. The SDC System has not been impacted, because of the geometry of the SDC lines and the fact that the system would be isolated during power operation. This conclusion is further addressed in section 2.2.

- Is the SSC used to:

  - Maintain reactor coolant pressure boundary integrity?

    As a Non-Safety related component, feedwater spargers do not constitute a barrier to the reactor coolant system and do not form a pressure boundary. The spargers are effectively open pipes distributing Feedwater flow within the reactor vessel.

If the probe were to exit the sparger and arrive in the lower plenum, it has the potential to fret reactor pressure boundary components. This fretting potential is further evaluated in Section 2.2 and found to not be significant.

- Shutdown the reactor?

  No, negative reactivity to shutdown the reactor is provided by the control rods and their CRD systems. Alternately, SBLC can inject boron to shutdown the reactor if the control rods are not effective. The feedwater spargers have no physical interaction with those systems.

  If the probe were to exit the sparger, it would have the capability of affecting the control rods and their associated drive systems. The evaluation in Section 2.2 and the GE Lost Parts Analysis show that there is no effect on the CRD System from the lost part(s).

- Maintain the reactor in a safe shutdown condition?

  No. While the feedwater spargers are in the flow path for HPCI, as stated previously, function of the HPCI system will not be hindered as a result of damage to the sparger.

  The probe has no impact on reactivity. The probe has no effect on the Standby Liquid Control System (SBLC) and its capability. The SBLC sparger is constructed of stainless steel and impingement by the probe, which is also stainless steel, would be of minimal to no consequence due to the relatively small mass of the probe and the relatively lower velocities in this region. Therefore, the instrumentation functions of the SBLC sparger would also not be affected.

- Prevent or mitigate the consequences of an accident that could result in offsite exposures comparable to 10 CFR 50.34(a)(1) or 10 CFR 100.11 guidelines, as applicable.

  No. After the probe fails and is transported through the feedwater system it will pass through the feedwater check valves due to the geometry of the valves. Therefore, the performance of these containment isolation valves will not be affected.

  The feedwater spargers do not form a fission product barrier for the design basis dose analyses. As non-safety related components, their only function is the distribution of the feedwater inside the vessel.

  If the probe were to exit the sparger, it would not change any accident or accident response. The potential to impact ECCS are evaluated in section 2.2 below. The conclusion is that the capacity or reliability of those systems is not impaired.

- Does the SSC provide required support (i.e., cooling, lubrication, etc.) to a TS required SSC?

  No. The spargers do not perform any function directly or indirectly required by Technical Specifications.

  If the probe were to exit the sparger, it could affect Tech Spec required SSC's. Where that potential exists, it has been evaluated in section 2.2 and these systems will be able to perform their design function.

- Is the SSC used to provide isolation between safety trains, or between safety and non-safety ties?

  No, the feedwater spargers do not form a pressure boundary and have no isolation functions.

If the probe were to exit the sparger, the potential impact on containment isolation valves, such as RWCU is addressed in section 2.2.

- Is the SSC required to be operated manually to mitigate a design basis event?

  No, there are no manual controls for the feedwater sparger; it is a passive component within the feedwater process flow distributing feedwater as it enters the reactor.

  If the probe were to exit the sparger, it would not cause or prevent any SSC from being operated manually. The probe would not require new manual actions where automatic actions would have functioned previously.

- Have all safety functions described in TS been included?

  Yes. The Technical Specifications do not address the feedwater spargers. The spargers are not an ASME Code Class 1, 2 or 3 component and therefore Technical Requirements Manual (TRM) 3.4.a for Structural Integrity does not apply. However, the Technical Specifications implicitly assume that the feedwater spargers will not prevent any equipment important to safety from performing its design function.

  If the probe were to exit the sparger, the safety functions that could be impacted by the lost probe have been included in this evaluation. These safety functions include LPCI, fuel integrity, control rod operations, chemical reactions with other materials, nuclear boiler or nuclear instrument monitoring systems and reactor internals. Section 2.5.1 of this evaluation lists the Technical Specification sections that were referenced.

- Have all safety functions described in the UFSAR or pending revisions been included?

  Yes. Since there are no safety functions associated with the feedwater sparger, the UFSAR does not provide any descriptions related to the sparger performing a safety functions.

  If the probe were to exit the sparger, the safety functions described in the UFSAR that could be impacted by the probe are included in this evaluation. Section 2.5.2 of this evaluation lists the sections of the UFSAR that were referenced.

- Have all safety functions of the SSC required during normal operation and potential accident conditions been included?

  Yes. There are no safety related functions for the feedwater spargers. All other safety functions are included in section 2.2

- Is the SSC used to assess conditions for Emergency Action Levels (EALs)?

  No. While Feedwater system is used as a source of make up during emergency operating conditions, no credit has been taken for the feedwater system as a whole to mitigate consequences of an accident. Loss of feedwater system is analyzed in the UFSAR as one of the transients in Chapter 15.

  The lost probe does not prevent any condition assessment for EAL determination. The potential for the lost probe to impact nuclear instrumentation or instrument lines used for the measurement of reactor pressure, level, or other safety significant instrumentation is evaluated in section 2.2.

2.2 Describe the following, as applicable: (a) the effect of the degraded or nonconforming condition on the SSC safety function(s); (b) any requirements or commitments established for the SSC and any challenges to these; (c) the circumstances of the degraded/nonconforming condition, including the possible failure mechanism(s); (d) whether the potential failure is time dependent and whether the condition will continue to degrade and/or will the potential consequences increase; and (e) the safest plant configuration, including the effect of transitional action:

Feedwater Sparger

The feedwater spargers do not perform a safety function. The spargers are designed to withstand both normal operational loads as well as design basis events without generating any loose parts.

The lost parts analysis performed for the sample probe and sparger disks evaluated the effects of the loose parts on fuel and control rod drives, concluding no fuel or control rod drive safety concerns associated with the operation of Unit 2 (NF-MW: 01-0351, NFM-MW:01-0351, EC# 345441, 345690 and GE-NE-0000-0023-5200-R0).

There are no requirements or commitments established for the spargers, as they are non-safety related components inside the reactor. They are designed to withstand the normal operating loads and design basis accident loads .

The mechanism for the feedwater sparger damage is rubbing and wear over time. As experienced in the industry, the probe has failed, traveled to the feedwater spargers and caused damage to the sparger nozzles.

As stated previously, the feedwater sparger damage is time dependent and the extent of damage is increased with time and has been limited to the sparger itself. Examination of the Unit 2 damaged sparger and the surrounding environment revealed no other damages.

300 series austenitic stainless steels are tough and ductile materials. As a result, it would not be expected to break off large fragments from component impacts. The stainless steel material would be expected to deform/yield without breaking.

A visual inspection of the failed Unit 2 probe in the Dresden Chemistry lab revealed the probe was losing material due to external wear. All of the worn regions on the probe exhibited smooth, polished features that were typical of a gradual wear mechanism.

Based on the orientation of the sparger nozzle damage, the probe and nozzle wear occurred while the probe was stuck in the nozzle opening. The flowing water would have caused the probe to move within the nozzle, which would have resulted in relative movement between the contacting surfaces of the probe and nozzle. This would result in wear by either a sliding contact or fretting mechanism. In general, the debris that is generated by these wear mechanisms would be expected to consist of small particles. For example, the ASM Metals Handbook reports that the particles generated by fretting wear are normally less than 0.004" in diameter.

Based on the sparger repairs performed during D2R18, our experience that sparger damage is caused by the sample probe rubbing and fretting over a considerable amount of time and the conclusions of our evaluation that there are no consequences from lost parts, the safest configuration is to continue operation until the next outage of sufficient duration (no later than the end of D2R19) when a retrieval can be planned and executed. .

The following cases are analyzed although the chance of a probe exiting the sparger is considered to be very low based on D2R18, D3M10 and Perry experience.

a) Low Pressure Coolant Injection (LPCI) System

If the probe were to pass through the Recirculation Pump, it is of sufficient length that it would cause damage to the pump impeller and the probe. The resulting pieces are expected to be relatively large and pass through the discharge valves based on geometry and flowrate. Both the 2-0202-5A/B valves have been stroke timed November 2003 and were found to be acceptable. The recirculation pumps are monitored for vibrations and have had no abnormal indications. Since the pumps have been operating normally, this supports the conclusion that the probe has not passed through a recirculation pump. The LPCI Loop Select Logic would not be impacted.

b) Fuel bundle flow blockage and fuel damage due to overheating of the fuel cladding

The probe diameter is smaller than the jet pump nozzle and could pass into the lower plenum via the recirculation loop. Jet pump suction flow could also allow the probe to pass into the lower plenum. The results of the analyses to determine the orifice and lower tie-plate blockage necessary to cause boiling transition at Extended Power Uprate (EPU) conditions with GE 14 and Atrium 9B fuel are contained in the GE Lost Parts Analysis. The probe is not small enough to enter and plug the orifice. Also, the lower plenum geometry and lift force preclude the probe from blocking a side entry orifice. Peripheral fuel bundles have bottom entry orifices, however; even if an orifice is completely blocked, adequate cooling from reverse flow from the lower tie plate bypass flow holes and fuel channel clearances will occur. The GE report concluded that there is no safety concern for the reasons noted above.

c) Control Rod Operations

Per the GE Lost Parts Analysis, foreign material that is greater than 0.5" cubed would be too large to interfere with control rod operation regardless of the migration path they traveled. Since the probe is 1" in diameter, interference with control rod operation will not occur. Therefore, no potential for interference with the safety functions of the control rods exist. Refer to section 2.2.g and 2.2.i below for additional details.

d) Corrosion or adverse chemical reactions with other reactor materials

In BWR NSSS systems, the low conductivity demineralized water used does not contain significant quantities of ions that would accelerate galvanic corrosion effects. Stainless Steel (such as this probe constructed of 316L) has been used in numerous components in the reactor pressure vessel and throughout the NSSS and nuclear plant recirculation systems for many years without corrosion or adverse chemical reaction with other reactor materials. It is concluded that there will be no significant corrosive or adverse chemical reactions with other reactor materials.

e) Interference with Reactor Water Cleanup (RWCU) System

During normal operation, the flow velocity through the SDC line due to operation of RWCU is expected to be too low to carry the probe through the SDC piping and up into the RWCU piping due to the large diameter of the SDC piping (16")

versus the small diameter (8") of the RWCU piping. In the event that the probe were to get trapped in a recess area inside one of the RWCU isolation valve bodies, it could prevent the complete operation of the valve. If the probe does not hang-up on pipe fittings, then it is unlikely to stay in the gate valve used for isolation. (Can a stronger argument be made here?) Therefore, it is not expected that the probe would interfere with operation of the RWCU valves such that closure would be prevented or impaired. These valves have been used as isolation for repairs as well as stroked and timed during the refuel outage and there is no evidence of foreign material in these valves. The RWCU system is on and running with no indication of any foreign material affecting system performance.

In addition, the RWCU system would be expected to filter any fretting wear products (as discussed in section 2.2.g below) as a normal part of system operation.

f) Interference with the Nuclear Boiler or Neutron Monitoring Instrumentation

The sample probe will not migrate to the Nuclear Boiler Instrumentation, due to an absence of fow through the pressure and differential pressure instrument lines associated with vessel water level and recirculation flow.

Fretting, or damage, to a Nuclear Instrumentation (NI) tube is bounded because the loss of a single NI has no safety significance. There are multiple bypass flow paths for LPRMs. Core bypass flow is approximately 10% of the total core flow; therefore, neutron monitoring would not be impacted even if one path became plugged by the probe. Normal NI monitoring and surveillances will provide early identification of any equipment challenges, thus allowing further evaluation and corrective actions within the Technical Specification Action Limits.

The Nuclear Instrument guide tubes are vulnerable to fretting but this is not likely because of the lower flow velocities in this area. Fretting damage would take place over a long period of time and the station would take action based on Technical Specification Action Limits. Fretting damage is described in further detail in section 2.2.g below.

g) Potential Fuel Fretting

Based on the orientation of the sparger nozzle damage seen during D2R18, the probe and nozzle wear occurred while the probe was stuck in the nozzle opening. The flowing water would have caused the probe to move within the nozzle, which would have resulted in relative movement between the contacting surfaces of the probe and nozzle. This would result in wear by either a sliding contact or fretting mechanism. In general, the debris that is generated by these wear mechanisms would be expected to consist of small particles. For example, the ASM Metals Handbook reports that the particles generated by fretting wear are normally less than 0.004" in diameter. Therefore we would expect that debris generated by wear damage caused by the probe on the sparger would result in similar sized particles. Since the probe is constructed of 300 series austenitic stainless steel and it is a tough and ductile material, it would be expected to deform/yield without breaking. This material would be generated over an extended time and would be removed by the RWCU system.

Particles from the probe or sparger will pass through a lower tie-plate of the Atrium 9B fuel and will not wear a hole in the fuel cladding. If fuel cladding leakage did occur, the Offgas System would detect it so that appropriate actions

could be taken to maintain the off-gas radiation release within acceptable limits. All applicable TS actions would be carried out as required.

h) SDC Valves

The SDC suction line (16") comes off the side of a vertical run of the larger Recirculation System pipe (28" line) upstream of the Recirculation Pump suction. The SDC isolation valves (2-1001-1A(B)) are closed during normal operation. Since this probe would not have an opportunity to enter the system during normal operation, the part will not interfere with the safety function of these valves.

i) Potential for Impact Damage on Reactor Internals

If the probe were to exit the feedwater sparger it is likely to land in the reactor annulus, where it would lie dormant. It is possible that it could migrate through the recirculation system and into the lower plenum. The size of the jet pump nozzle would allow for the probe to be drawn in through a jet pump and come to rest in the lower plenum. Once in the lower plenum:

- As evaluated in GE evaluation GE-NE-0000-0023-5200-R0, a large part, such as the size of the probe, cannot adversely affect the operation of any CRD.
- Through-wall wear of the CRD housing stub tube will be detected and action would be taken based on Technical Specification Action Limits.
- Through-wall leakage is not expected to affect the scram capability of a CRD, but in the event that this did occur, the shutdown margin used in the safety analysis is performed with one CRD fully withdrawn. (Reference section 2.2.c above for additional details)

With the exception of the Nuclear Instrument guide tubes, components at the bottom of the vessel are thick and not expected to wear through from fretting. Therefore, there is no expected damage from the probe and no impact on the reactor pressure boundary or the ability to maintain two-thirds core coverage following a LOCA.

j) Potential for FME To Travel Up the Recirculation Sample Line

The HRSS Recirculation Sample line will not be affected by this issue. AO 2-0220-44/45 provides containment isolation for that line and are normally open valves. This ¾" line taps off the center of the 22" Recirculation Discharge pipe on the horizontal run of the pipe. The diameter of the probe is larger than the pipe diameter and is therefore unable to enter.

In the cases detailed above, the lost probe was determined to have minimal to no impact on the potentially affected systems, structures, and components.

		YES	NO
2.3	Is SSC operability supported? Explain basis (e.g., analysis, test, operating experience, engineering judgment, etc.): If 2.3 = NO, notify Operations Shift Management immediately. If 2.3 = YES, clearly document the basis for the determination.	[X]	[ ]

The Reactor internals and associated systems, containment, feedwater sparger and feedwater system are operable. This determination is based on the following:

- The industry experience, including the Dresden Unit 2 D2R18, D3M10 and Perry Nuclear Power Plant have demonstrated that the damage caused by a sample probe is limited to damage of the sparger only and no other damage has been reported to vessel internals, as revealed by searching Operating Experience documents.

- The feedwater sparger for Unit 2 was inspected in 2003 and the damage that was discovered was repaired per the GE plan. Since the sample probe rubbing and fretting against the sparger causes this damage, it is our judgment that this damage would take a considerable amount of time.

- The 'As Found' testing of the Local Leak Rate Testing for the Containment Isolation Check Valves in the feedwater system did not reveal evidence to infer the presence of such probe in the valves. In addition, size and design of these valves are such that the probe would pass through the valves given the quantity and velocity of the flow through these valves.

- Degraded conditions of the feedwater spargers may cause a change in the direction of the feedwater flow. The impinged flow onto the vessel surface could lead to thermal fatigue in the vessel cladding. The D2R18 IVVI (EVT-1) inspection of the vessel wall did not reveal any indications of thermal fatigue cracking. Industry experience from D2R18 and Perry indicate that the sample probe would be lifted in the flow stream towards the outlet nozzles causing damage to the top surfaces of the sparger. This is not expected to direct flow onto the vessel wall. As a result of the damage found during D2R18 on the top surfaces of the feedwater sparger, flow was not diverted directly on to the vessel cladding. This was also the case for the damage on the Perry sparger. As a result, no thermal fatigue damage occurred due to the probe induced sparger damage.

- No damage was discovered during the D3M10 Unit 3 internal and external inspections despite the fact that two isokinetic probes were discovered within the 150° feedwater sparger.

- Fuel safety has been analyzed through the lost parts analysis (NFM-MW: 01-0351) performed in November 2001 for Unit 2. A third lost parts analysis was performed under EC# 345441 to evaluate the isokinetic probe being lost in the RPV. These reports discuss the possibility of the probe entering the reactor and the probability of it being entrained in the suction inlet of a jet pump inlet mixer where it would pass through to the lower vessel plenum. It also discusses the possibility of the probe residing in a low flow region on top of the shroud support plate, or being entrained in recirculation flow and potentially passing through the recirculation pump and into the lower plenum via the jet pump nozzle. Once located along the bottom of the vessel, the size and shape of this probe would not permit it to be swept by the flow toward the inlet orifices of a fuel support casting. There is no potential for fuel bundle flow blockage; therefore, boiling transition will not be initiated in any of the fuel bundles. NFM determined that there are no fuel or control rod drive safety concerns associated with operation of Dresden Units 2 & 3 with the missing final feedwater isokinetic sample probe.

CONCLUSION

This evaluation reviewed the potential effects of lost parts and feedwater flow impingement on the RPV cladding. The lost parts generated by a postulated event similar to that occurring on Dresden Unit 2 during D2R18, were analyzed and it was concluded that there is no fuel or control rod drive safety concerns. Similarly, feedwater impingement on the vessel wall is not expected to occur. Additionally, LLRTs last performed on the Containment Isolation Check Valves in the feedwater system did not

reveal evidence to infer the presence of such probe in the valves. The probe would not be expected to become lodged in a containment isolation valve based on size and geometry. Thus, the containment, reactor vessel, feedwater sparger and feedwater system are operable. Corrective actions are discussed below.

COMPENSATORY AND CORRECTIVE ACTIONS

No Compensatory Actions are required to ensure operability of the Dresden Unit 2 feedwater spargers.

A Corrective Action is required to locate and retrieve Unit 2 Feedwater Sample Probe to prevent damage to the spargers during the next refuel or maintenance outage of sufficient duration.

PRUDENT ACTIONS

Normal in-place monitoring will provide early identification of any equipment challenges, thus allowing further evaluation and corrective actions.

- Drywell unidentified leakage
- Normal CRD exercising per Technical Specifications
- Normal NI monitoring and surveillances
- Jet Pump flows
- Recirc Pump Vibration Monitoring

		YES	NO
2.4	Are compensatory and/or corrective actions required? If 2.4 = YES, complete section 3.0 (if NO, N/A section 3.0).	[X]	[ ]

2.5 Reference Documents:

Technical Specifications Section(s):

3.1.3 Control Rod Operability
3.3.5.1 ECCS Instrumentation
3.4.2 Jet Pumps
3.4.4 Reactor Coolant System Operational leakage
3.6.1.3 Primary Containment Isolation Valves

UFSAR Section(s):

3.9.5.1 Reactor Pressure Vessel Internals
15.0 Accident and Transient Analyses
6.3.2.2 Low Pressure Coolant Injection Sub-System

Other:

1. NF-MW: 02-0401
2. NFM-MW: 01-0351
3. EC# 345672
4. PNPP CR# 92077
5. EC# 345690
6. GE-NE-0000-0023-5200-R0
7. GE-NE-0000-0023-5205-Draft A
8. Email From J. Chynoweth to L. Dyas dated 12/12/03
9. Quad Cities OpEval 188333-08
10. GE-NE-0000-0022-9168-SE

ACTION ITEM LIST

3.0 If, through evaluating SSC operability, it is determined that the degraded or nonconforming SSC does **not** prevent accomplishment of the specified safety function(s) in the TS or UFSAR and the intention is to continue operating the plant in that condition, then record below, as appropriate, any required compensatory actions to support operability and/or corrective actions required to restore full qualification. For corrective actions, document when the actions should be completed (e.g., immediate, within next 13 week period, next outage, etc.) and the basis for timeliness of the action. Corrective action timeframes longer than the next refueling outage are to be explicitly justified as part of the OpEval or deficiency tracking documentation being used to perform the corrective action.

Compensatory Action: None

Corrective Action #1: Inspect susceptible areas on the Unit 2 feedwater sparger during D2R19 and remove sample probe if found.

Responsible Dept./Supv.: Engineering Programs/A8351NESPR

Action Due: 12/30/05

Action Tracking #: 187258

Corrective Action #2: Obtain GE lost parts analysis for sample probe to further substantiate the conclusions of this evaluation.

Responsible Dept./Supv.: Engineering Programs/A8351NESPR

Action Due: 12/29/03

Action Tracking #: 187258

Prudent Action:

Normal in-place monitoring will provide early identification of any equipment challenges, thus allowing further evaluation and corrective actions.

- Drywell unidentified leakage
- Normal CRD exercising per Technical Specifications
- Normal NI monitoring and surveillances
- Jet Pump flows
- Recirc Pump Vibration Monitoring

4.0 SIGNATURES:

4.1 Preparer(s) _____ Date 12/15/03

_____ Date _____

4.2 Reviewer _____ Date 12/15/03

(10 CFR 50.59 screener qualified or active SRO license holder)

4.3 Sr. Manager Design Eng/Designee Concurrence _____ Date 12/15/03

4.4 Operations Shift Management Approval _____ Date 12/15/03

4.5 Ensure the completed form is forwarded to the OEPM for processing and Action Tracking entry as appropriate.

5.0 OPERABILITY EVALUATION CLOSURE:

5.1 Corrective actions are complete, as necessary, and the OpEval is ready for closure

_____ Date _____

(OEPM)

5.2 Operations Shift Management Approval _____ Date _____

5.3 Ensure the completed form is forwarded to the OEPM for processing, Action Tracking entry, and cancellation of any open compensatory actions, as appropriate.