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RS-05-069

May 25, 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: Additional Information Related to Dresden and Quad Cities Nuclear Power Stations 54 Effective Full Power Years Pressure-Temperature Limits Curves

Reference: Letter from Patrick R. Simpson (Exelon Generation Company, LLC) to U. S. NRC, "Request for Changes Related to Technical Specifications Section 3.4.9, 'Reactor Coolant System Pressure and Temperature (PIT) Limits,'" dated November 4, 2004

In the referenced letter, Exelon Generation Company, LLC (EGC) requested an amendment to Technical Specifications (TS), Section 3.4.9, "Reactor Coolant System Pressure and Temperature (P/T) Limits," for Dresden Nuclear Power Station (DNPS), Units 2 and 3, and Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2. The request change revises the P/T limits curves for 54 effective full power years (EFPY) to support 20-year license extensions for both DNPS and QCNPS to 60 years (i.e., 54 EFPY), and resolves a non-conservative condition for TS Section 3 .4.9, Figure 3.4.9-2, "Non-Nuclear Heatup/Cooldown Curve," for QCNPS.

In a communication from Mr. Larry W. Rossbach on May 9, 2005, the NRC requested additional information concerning the proposed change. The attachments to this letter provide the requested information as follows.

1. Attachment 1 contains Operability Evaluation #179235 – 08, Revision 4, performed to address the non-conservative condition in the QCNPS TS for (P/T) limits during non-nuclear heatup/cooldown.

2. Attachment 2 contains the in-progress notes from EGC corrective action Issue Report # 179235, Assignment 13, related to the above listed Operability Evaluation.

A001

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

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 25th day of May 2005.

Respectfully,

Patrick R. Simpson

Patrick R. Simpson
Manager – Licensing

Attachments:

1. Operability Evaluation #179235 – 08, Revision 4
2. In-Progress Notes: Issue Report 179235, Assignment 13

ATTACHMENT 1

Operability Evaluation #179235 – 08, Revision 4

1.0 ISSUE IDENTIFICATION:

1.1 CR #: 179235 / 221865

1.2 OpEval #: 179235-08 Revision: 04

General Information:

1.3 Affected Station(s): Quad Cities

1.4 Unit(s): 01 & 02

1.5 System: RX (Reactor Pressure Vessel)

1.6 Component(s) Affected: Reactor Pressure Vessel Pressure and Temperature (P/T) Limits

1.7 Detailed description of what SSC is degraded or the nonconforming condition and by what means and when first discovered:

Revision 1 to this Operability Evaluation changes the due date of Corrective Action Number 1. General Electric is not able to complete the finite element analysis until the end of March 2004. Because of this, the due date for Corrective Action Number 1 is being changed to April 30, 2004. This is acceptable because General Electric is confident that the analysis will result in no change to the existing figure in Technical Specification Figure 3.4.9-2. In the meantime, the station will maintain Compensatory Measure Number 1, which adds an 11 deg F conservatism to the station procedures implementing Technical Specification Figure 3.4.9-2. Therefore, the safety consequences are minimal and it is acceptable to exceed the end of refuel outage Q2R17.

Revision 2 to this Operability Evaluation changes the due date of Corrective Action Number 1 and adds a new corrective action (number 2). General Electric still has not been able to complete the finite element analysis, however, preliminary work indicates that a license amendment will be required. The existing Compensatory Measure Number 1, which adds an 11 deg F conservatism to the station procedures implementing Technical Specification Figure 3.4.9-2, will still remain bounding under all required conditions. Therefore, the safety consequences are minimal and it is acceptable to exceed this Operability Evaluation beyond the end of refuel outage Q1R18. Corrective Action #2 is to submit and track to implementation, the submittal of a license amendment to Technical Specification Figure 3.4.9-2.

Revision 3 to this Operability Evaluation changes Compensatory Action #1 to incorporate a 14.6°F shift in the P/T curve (QCOS 0201-02, Attachment C) for the reactor pressure vessel bottom head as shown in Technical Specification Figure 3.4.9-2). The change shifts Attachment C of QCOS 0201-02 to the right (represented by dashed lines on Technical Specification Figure 3.4.9-2), which only applies for the region of the curve above a pressure limit in the reactor vessel top head of 470 psig where the curve becomes non-linear. The 14.6°F shift that is required is from an analysis performed for LaSalle Station. GE has confirmed that the Quad Cities specific analysis will be received on May 28, 2004, and has verbally transmitted that the results of the Quad Cities analysis will be the same as that of the LaSalle report. The LaSalle report received a 3rd Party Review as documented by GE-NE-0000-0003-5526-02R1.

Revision 4 to this Operability Evaluation changes the due date of Corrective Action Number 2. The Technical Specification change was submitted on 11/4/2004 and the NRC is not expected to approve the change until 11/7/2005. A new due date of 12/31/2005 will allow the Technical Specification Change to be implemented. This is acceptable because no physical changes are required and the procedure changes that implement the Technical Specification Change have already implemented the conservative Technical Specification limits. Therefore, the safety consequences are minimal and it is acceptable to exceed the end of refuel outage Q1R18.

A non-conforming design condition exists with a potential nonconservative Technical Specification (TS) value that may not assure safety. This nonconservative value relates to the Reactor Coolant System (RCS) pressure and temperature (P/T) limits during non-nuclear heatup/cooldown. Specifically, the P/T curve for the reactor pressure vessel bottom head as shown in Technical Specification Figure 3.4.9-2 "Non-Nuclear Heatup/Cooldown Curve" is non-conservative by 14.6°F for Quad Cities Unit 1 and 11.6°F for Quad Cities Unit 2 during a bounding anticipated operational occurrence for the improper start of an idle recirculation pump which is more severe than the (normal upset condition) of Loss of Feedwater transient. This only applies for the region of curve above a pressure limit in the reactor vessel top head of 470 psig where the curve becomes non-linear, see Attachment A. Since the P/T limits are established based upon Appendix G to Section XI of the ASME Boiler & Pressure Code, this potential non-conservative P/T limit implies that a safety factor of two for primary membrane stress and primary bending stress for the reactor pressure vessel bottom head (CRD penetration region) may not be maintained if the reactor vessel bottom head is operating near the existing Technical Specification Figure 3.4.9-2 "Non-Nuclear Heatup/Cooldown Curve."

Technical Specification 3.4.9, RCS Pressure and Temperature (P/T) Limits, requires that the RCS pressure, temperature, heatup and cooldown rates, and the recirculation pump starting temperature requirements be maintained within limits at all times. The RCS P/T limits are provided in Technical Specification Figure 3.4.9-1 "Non-Nuclear Inservice Leak and Hydrostatic Testing Curve", Figure 3.4.9-2 "Non-Nuclear Heatup/Cooldown Curve" and Figure 3.4.9-3 "Critical Operations Curve." In response to an industry issue regarding the calculational methodology that General Electric (GE) has used to develop the P/T curves, GE was tasked to validate the assumptions used in the P/T curves since 1988. As a result of this GE review, Quad Cities Station was informed on October 3, 2003 by GE that the current "Non-Nuclear Heatup/Cooldown Curve" associated with the reactor vessel bottom head P/T limit curve is potentially non-conservative for the bounding anticipated operational transient. Technical Specification Figure 3.4.9-1 "Non-Nuclear Inservice Leak and Hydrostatic Testing Curve" and Figure 3.4.9-3 "Critical Operations Curve" are accurate along with the Upper Vessel and Beltline limit curve as shown in Figure 3.4.9-2 "Non-Nuclear Heatup/Cooldown Curve". GE has stated that this potential non-conservative limit only applies to the Bottom Head

Curve (non-linear portion) as shown in Figure 3.4.9-2 "Non-Nuclear Heatup/Cooldown Curve".

A finite element analysis was required to determine if the current Technical Specification Figure 3.4.9-2 is acceptable or the potential non-conservatism is reduced and this evaluation addresses the current identified non-conservative Technical Specification difference of 14.6°F for Quad Cities Unit 1 and 11.6°F for Quad Cities Unit 2 in the bounding anticipated operational transient with regards to reactor pressure vessel bottom head P/T limits that may not assure safety. GE indicates that a generic finite element analysis to refine the analyses has recently been completed and this has resulted in the differences indicated above.

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):

- Does the SSC receive/initiate an RPS or ESF actuation signal?
- Is the SSC in the main flow path of an ECCS or support system?
- Is the SSC used to:
  - Maintain reactor coolant pressure boundary integrity?
  - Shutdown the reactor?
  - Maintain the reactor in 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.
- Does the SSC provide required support (i.e., cooling, lubrication, etc.) to a TS required SSC?
- Is the SSC used to provide isolation between safety trains, or between safety and non-safety ties?
- Is the SSC required to be operated manually to mitigate a design basis event?
- Have all safety functions described in TS been included?
- Have all safety functions described in UFSAR or pending revisions been included?
- Have all safety functions of the SSC required during normal operation and potential accident conditions been included?
- Is the SSC used to assess conditions for Emergency Action Levels (EALs)?

<u>The purpose of the Reactor Pressure Vessel (RPV) is to support and contain the reactor core, the reactor internals, and the reactor core coolant-moderator and to serve as a high integrity barrier against leakage of radioactive materials to the drywell. The RPV ensures structural integrity of the Reactor Coolant system Pressure Boundary (RCPB), provides the capability to shutdown the reactor and maintain it in a safe shutdown condition, and mitigates the consequences of accidents that could result in potential offsite exposure comparable to the limits in 10CFR100. The major safety consideration for the RPV is the ability to function as a radioactive material barrier. The reactor vessel is a primary barrier against the release of fission products to the environs. In order to provide assurance that this barrier is maintained at a high degree of integrity, pressure-temperature (P/T) limits have been established for the operating conditions to which the reactor vessel can be subjected.</u>

<u>All components of the Reactor Coolant System (RCS) are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. The Limiting Condition for Operation (LCO) as contained in Technical Specification 3.4.9 limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation. 10CFR50, Appendix G requires the establishment of Pressure and Temperature (P/T) limits for material fracture toughness requirements of the RCPB materials. 10CFR50 Appendix G requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of and supplements the requirements of Appendix G to Section XI of the ASME Boiler & Pressure Code.</u>

<u>Technical Specification 3.4.9 contains P/T limit curves for heatup, cooldown, and inservice leak and hydrostatic testing, and criticality, and also limits the maximum rate of change of reactor coolant temperature. The P/T limit curves are applicable for 32 Effective Full Power Years (EFPY). Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curve is operational guidance during heatup and cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that the vessel temperature is within the allowable region. The Technical Specification LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the RCPB. The vessel is the component most subject to brittle failure. Therefore, the LCO limits apply mainly to the vessel (e.g. bottom head).</u>

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:

<u>The P/T limit curves are composite curves established by superimposing limits derived from flaw evaluations of those reactor vessel components that are the most restrictive. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.</u>

Technical Specification Figure 3.4.9-2 "Non-Nuclear Heatup/Cooldown Curve" applies during heatups with non-nuclear heat (e.g. recirculation pump heat), and during cooldowns when the reactor is not critical (e.g. following a scram or unit shutdown). The curve provides the minimum reactor vessel coolant temperatures based on the most limiting vessel stress. The maximum heatup/cooldown rate of 100°F/hour is applicable for this curve. The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, a condition that is unanalyzed. Since the P/T limits are not derived from any DBA, there are no acceptance limits related to the P/T limits. Rather, the P/T limits are acceptance limits themselves since they preclude operation in an unanalyzed condition.

Procedure QCOS 0201-02 "Primary System Boundary Thermal Limitations" provides the steps for the Technical Specification required recording and monitoring of reactor coolant heatup or cooldown rates, reactor vessel flange and head flange temperatures and reactor pressurization temperatures. This procedure should be initiated subsequent to normal Reactor heatup or cooldown as directed by the applicable site procedure, and *as soon as practical following a Reactor scram as directed by the Unit Supervisor.* In addition, if the Reactor is in operational mode 4 and the reactor coolant is < 113°F or if the Reactor is in operational mode 5, then this procedure should also be initiated. The applicable site procedures that refer to QCOS 0201-02 (in particular, reference of Technical Specification 3.4.9 and/or Figure 3.4.9-7) are: QCGP 1-1 "Normal Unit Startup," QCGP 2-3 "Reactor Scram," QCOP 1000-05 "Shutdown Cooling Operation," QCOP 1000-17 "Shutdown Cooling, Reactor Temperature Trending," QCOP 1000-38 "Alternate Shutdown Cooling," QCOP 1000-42 "Noble Metal Injection Procedure," QCOS 0500-10 "Transitioning from Operational Mode 4 to Operational Mode 5," and QCOS 1000-24 "Shutdown Cooling Outage Report." Furthermore, the Reactor Recirculation System startup and shutdown procedures (QCOP 0202 block) refers to the monitoring of the reactor pressure vessel coolant and bottom head coolant temperatures. Specifically, following a loss of both Recirculation pumps under scram or low power conditions (loss of forced circulation), reactor coolant thermal stratification may occur as cold water injection to the vessel (i.e., CRD, Feedwater, HPCI, RCIC, SSMP, etc.) accumulates in the lower head region. Under these conditions Reactor Vessel metal Pressure-Temperature limits and cooldown rate limits may be challenged. Also, Reactor Vessel metal heat-up rate limits may be challenged if stratification has occurred and increased circulation is re-established by starting the Recirculation pumps or by drawing off steam with the Bypass or Relief Valves.

Both the loss of the recirculation pumps and the starting of an idle recirculation pump at incorrect temperatures are transients as discussed in the UFSAR. UFSAR Section 15.3, Decrease in Reactor Coolant System Flow Rate, discusses Single and Multiple Recirculation Pump trips in Section 15.3.1 and in UFSAR Section 15.4, Reactivity and Power Distribution Anomalies, discusses the Startup of Idle Recirculation Loop at Incorrect Temperature in Section 15.4.4. Although, the most likely conditions in which operation near the bottom head curve for non-nuclear heatup/cooldown would be during a hydrostatic test set-up or after a scram, the emergency/faulted transients with the most severe bottom head thermal conditions were sudden start of an idle recirculation loop (cooldown) and improper startup from a hot standby condition with the bottom head drain closed (heatup).

The consequence of violating the P/T limits (LCO limits) under the various conditions as described above for the potential non-conservative non-nuclear heatup/cooldown bottom head curve is that the RCS may have been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the vessel bottom head. ASME Code, Section XI, Appendix E, provides a recommended methodology for evaluating an operating event that causes an excursion outside the P/T limits.

YES NO

2.3 Is SSC operability supported? Explain basis (e.g., analysis, test, operating experience, engineering judgment, etc.): [X] [ ]

The P/T curve for the reactor pressure vessel bottom head as shown in Technical Specification Figure 3.4.9-2 "Non-Nuclear Heatup/Cooldown Curve" is for Quad Cities Unit 1 and 2. A common curve was developed based upon the most limiting P/T curve from the unique Quad Cities Unit 1 and Quad Cities Unit 2 analyses. The non-conservative value of 4°F for Quad Cities Unit 1 and 1°F for Quad Cities Unit 2 is in reference to the curves provided in report "Pressure-Temperature Curves For ComEd Quad Cities Unit 1," GE Nuclear Energy GE-NE-B13-02057-00-02R1, Revision 1, May 2000 and report "Pressure-Temperature Curves For ComEd Quad Cities Unit 2," GE Nuclear Energy GE-NE-B13-02057-00-01R2, Revision 2, May 2000. Based upon the review of the data and curves within these reports (Attachment B and C), Technical Specification Figure 3.4.9-2 "Non-Nuclear Heatup/Cooldown Curve" for the bottom head P/T limits was based upon Quad Cities Unit 1 analysis as contained in GE Nuclear Energy report GE-NE-B13-02057-00-02R1. The Unit 1 data contains the most limiting P/T curve in which the bottom head temperature limits are greater than 2°F at the various reactor vessel top head pressure limits than the Unit 2 bottom head P/T limits. Therefore, a shift of 1°F for Quad Cities Unit 2 non-conservative P/T limit would still be encompassed by the P/T curve in Technical Specification Figure 3.4.9-2. As a result, this is not an issue for Quad Cities Unit 2 because Technical Specification Figure 3.4.9-2 is from Quad Cities Unit 1 and it bounds the Unit 2 bottom head curve by more than the 1°F error applicable to the Unit 2 P/T limits. If operation of Unit 2 was maintained within the P/T limits as specified in existing Technical Specification 3.4.9, operation in an unanalyzed condition would not occur; therefore, Unit 2 need not be considered further in this evaluation as potentially exceeding a bottom head P/T limit for the bounding anticipated operational occurrence (normal upset condition) of Loss of Feedwater transient. In May 2004, General Electric has determined that for the improper start of an idle recirculation pump, which is more severe than the loss of feedwater pump transient, the shift from the original Technical Specification curves is 14.6°F (Unit 1) and 11.6°F (Unit 2).

General Electric (GE) has performed an engineering evaluation to determine the effects of an out-of-limit condition on the structural integrity of the reactor vessel, Attachment D, under the bounding anticipated operational occurrence (normal upset condition) of Loss of Feedwater transient. As stated above, this evaluation would only be applicable to Unit 1. The out-of-limit condition evaluated is a potential non-conformance of 4°F for Quad Cities Unit 1. In May 2004, General Electric has determined that for the improper start of an idle recirculation pump, which is more severe than the loss of feedwater pump transient, the shift from the original Technical Specification curves is 14.6°F (Unit 1) and 11.6°F (Unit 2). Three methods were used to perform the evaluation: 1) an evaluation using Appendix E to Section XI of the ASME Boiler & Pressure Code, 2) an evaluation using Appendix G to Section XI of the ASME Boiler & Pressure Code with a reduced postulated flaw size and 3) an evaluation using Appendix G to Section XI of the ASME Boiler & Pressure Code with a reduced safety factor. The evaluation using Appendix E to Section XI of the ASME Boiler &

Pressure Vessel Code, 1995 Edition with Addenda through 1996 (current commitment) demonstrates that the existing Technical Specification Figure 3.4.9-2 P/T limit curve has adequate structural integrity. The evaluation using Appendix G to Section XI of the ASME Boiler & Pressure Code, 1995 Edition with Addenda through 1996 demonstrates that the existing Technical Specification Figure 3.4.9-2 P/T limit curve has adequate structural integrity with a reduced postulated flaw size of 13% from 1/4T or with a reduced safety factor from 2.0 to 1.8 on the primary stress. Therefore, since all three evaluations demonstrate that the core-not-critical bottom head (CRD Penetration) curve in report GE Nuclear Energy GE-NE-B13-02057-00-02R1 (i.e. Technical Specification Figure 3.4.9-2) ensures adequate structural integrity of the reactor pressure vessel, the impact of a 14.6°F shift for Quad Cities Unit 1 would not have caused permanent damage to the RPV, had the limits been violated (within the 14.6°F shift) during the bounding improper start of an idle Recirculation Pump transient. As a result, the RPV operability with respect to the P/T limit of the bottom head non-conforming condition is supported by the above evaluations.

This operability evaluation is the result of GE being recently requested by the NRC to provide more information supporting assumptions regarding the core-not-critical pressure-temperature bottom head curve. Since a detailed evaluation would be required to prove that the assumption is correct, GE performed plant specific simplified evaluations to demonstrate that the existing curves are bounding or, alternately, if any preliminary adjustments would be required. This condition only affects plants with a separate core-not-critical bottom head curve that was based on a material K_{Ic} fracture toughness. GE had evaluated this condition for Quad Cities Units 1 and 2 for both the inside and outside surfaces of the bottom head shell adjacent to the penetration. The stress intensity for the outside surface is more limiting. The evaluation was performed for a single pressure, however, when scaling the pressure stress by a ratio of the pressure and the thermal stress by a ratio of the corresponding ΔT, the conclusions are conservative for pressures lower than 1106 psig and the same for pressures up to 1250 psig. The evaluation shows that the core-not-critical bottom head curves would require a preliminary adjustment of 14.6°F for Quad Cities Unit 1 for the bounding anticipated operational occurrence (normal upset condition) of improper start of an idle Recirculation Pump transient now that the more rigorous finite element evaluation has been completed. An interim compensatory measure to address the non-conforming condition is required. Because a design bases 14.6°F adjustment is required (for a maximum heatup/cooldown rate of 100°F/hour), this interim compensatory measure will revise procedure QCOS 0201-02 "Primary System Boundary Thermal Limitations" to incorporate a conservative 14.6°F shift in the P/T curve (QCOS 0201-02, Attachment C) for the reactor pressure vessel bottom head as shown in Technical Specification Figure 3.4.9-2 "Non-Nuclear Heatup/Cooldown Curve." This 14.6°F change covers all Emergency/Faulted conditions for the bottom head as delineated in report "Pressure-Temperature Curves For ComEd Quad Cities Unit 1," GE Nuclear Energy GE-NE-B13-02057-00-02R1, Revision 1, May 2000 and the report "Pressure-Temperature Curves For ComEd Quad Cities Unit 2," GE Nuclear Energy GE-NE-B13-02057-00-01R2, Revision 2, May 2000. The 14.6°F change would shift QCOS 0201-02 Attachment C curve to the right (represented by dashed lines on Technical Specification Figure 3.4.9-2). The shift only applies for the region of the curve above a pressure limit in the reactor vessel bottom head of 470 psig where the curve becomes non-linear. In performing this interim compensatory measure, operation near the revised P/T limit for all normal operation including anticipated operational occurrences and all emergency/faulted conditions will ensure that a safety factor of two for primary membrane stress and primary bending stress for the reactor pressure vessel bottom head (CRD penetration region) is maintained.

If 2.3 = NO, notify Operations Shift Management **immediately**.
If 2.3 = YES, clearly document the basis for the determination.

		YES	NO
2.4	Are compensatory and/or corrective actions required?	[X]	[ ]

If 2.4 = YES, complete section 3.0 (if NO, N/A section 3.0).

2.5 Reference Documents:

2.5.1 Technical Specifications Section(s):

3.4.9, RCS Pressure and Temperature (P/T) Limits

B 3.4.9, RCS Pressure and Temperature (P/T) Limits

2.5.2 UFSAR Revision 7 Section(s):

5.2, Integrity of Reactor Coolant Pressure Boundary

5.2.3.3.1, Fracture Toughness – Reactor Pressure Vessel

5.3, Reactor Vessels

5.3.2.1, Limit Curves

15.3, Decrease in Reactor Coolant System Flow Rate

15.3.1, Multiple Recirculation Pump Trips

15.4, Reactivity and Power Distribution Anomalies

15.4.4, Startup of Idle Recirculation Loop at Incorrect Temperature

2.5.3 Other:

10 CFR 50, Appendix G, December 1995 "Fracture Toughness Requirements"

ASME, Boiler and Pressure Code, 1989 Edition, Section III, Appendix G "Fracture Toughness Criteria for Protection Against Failure"

ASME, Boiler and Pressure Code, 1989 Edition, Section XI, Appendix G "Fracture Toughness Criteria for Protection Against Failure"

ASME, Boiler and Pressure Code, 1989 Edition, Section XI, Appendix E "Evaluation of Unanticipated Operating Events"

Regulatory Guide 1.99, Revision 2, May 1988 "Radiation Embrittlement of Reactor Vessel Materials"

Letter from S.N. Bailey (NRC) to ComEd, "Quad Cities – Issuance of Amendments – Revised Pressure-Temperature Limits," dated February 4, 2000

Pressure-Temperature Curves For ComEd Quad Cities Unit 1, GE Nuclear Energy GE-NE-B13-02057-00-02R1, Revision 1, May 2000

Pressure-Temperature Curves For ComEd Quad Cities Unit 2, GE Nuclear Energy GE-NE-B13-02057-00-01R2, Revision 2, May 2000

Letter from Daryl J. Bouchie, Technical Products Manager, Nuclear Services, General Electric to John Rommel, Exelon Corporation, LaSalle Station dated May 18, 2004, titled "Transmittal of Revised PT Curve Reports for LaSalle Station", GE-NE-0000-0003-5526-02R1.

3.0 ACTION ITEM LIST:

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 #1: Revise procedure QCOS 0201-02 "Primary System Boundary Thermal Limitations" to incorporate a 14.6°F shift in the P/T curve (QCOS 0201-02, Attachment C) for the reactor pressure vessel bottom head as shown in Technical Specification Figure 3.4.9-2 "Non-Nuclear Heatup/Cooldown Curve." In addition perform a 10 CFR 50.59 review for this interim compensatory action involving this procedure change. The change should shift Attachment C curve to the right (represented by dashed lines on Technical Specification Figure 3.4.9-2), which only applies for the region of curve above a pressure limit in the reactor vessel bottom head of 470 psig where the curve becomes non-linear. This procedure change shall also be annotated to identify it as a compensatory measure supporting the operability evaluation for CR 179235.

Responsible Dept./Supv.:A8410OP / Dave Boyles

Action Due: 05/28/04, Action Complete

Action Tracking #: 179235-

Compensatory Action #2: N/A

Responsible Dept./Supv.: N/A

Action Due: N/A

Action Tracking #: N/A

Compensatory Action #3: N/A

Responsible Dept./Supv.: N/A

Action Due: N/A

Action Tracking #: N/A

Compensatory Action #4: N/A

Responsible Dept./Supv.: N/A

Action Due: N/A

Action Tracking #: N/A

Corrective Action #1: Evaluate the General Electric (GE) finite element analysis and based upon the analysis results either submit a license amendment request to correct Technical Specification Figure 3.4.9-2 or leave the Technical Specification Figure 3.4.9-2 "as-is" and perform the necessary design bases document revisions reflecting/referencing the new GE finite element analysis including removal of Compensatory Action 1.

Responsible Dept./Supv.: A8451NESPR / Tom Wojcik

Action Due: 7/30/04

Basis for timeliness of action: The P/T curve for the reactor pressure vessel bottom head as shown in Technical Specification Figure 3.4.9-2 "Non-Nuclear Heatup/Cooldown Curve" only applies for the region of curve above a pressure limit in the reactor vessel bottom head of 470 psig where the curve becomes non-linear. The next Unit 1 refuel outage is Q1R18, scheduled to start in on January 2005. Final corrective actions will be completed after that refueling outage. GE has completed preliminary work on the finite element analysis and has indicated that the current compensatory measures in place (14.6°F shift of the curves) will bound the final analysis needed for the license amendment submittal (Corrective Action #2)

Action Tracking #: 179235-05

Corrective Action #2: Submit a license amendment request to correct Technical Specification Figure 3.4.9-2 and track to implementation.

Responsible Dept./Supv.: A8401RAPR / Mark Wagner

Action Due: 12/31/05

Basis for timeliness of action: This is acceptable because no physical changes are required and the procedure changes that implement the Technical Specification Change have already implemented the conservative Technical Specification limits. Therefore, the safety consequences are minimal and it is acceptable to exceed the end of refuel outage Q1R18.

Action Tracking #: 179235-16

Corrective Action #3: N/A

Responsible Dept./Supv.: N/A

Action Due: N/A

Basis for timeliness of action: N/A

Action Tracking #: N/A

Corrective Action #4: N/A

Responsible Dept./Supv.: N/A

Action Due: N/A

Basis for timeliness of action: N/A

Action Tracking #: N/A

4.0 SIGNATURES:

4.1 Preparer(s) Brian R. Strub / Brian Strub Date 3-9-05

Date ______

4.2 Reviewer Kent Johnson KJ Date 3-9-05
(10 CFR 50.59 screener qualified or active SRO license holder)
3rd Party - NA No Technical Change BRS 3-9-05

4.3 Sr. Manager Design Engg/Designee Concurrence Steve Baine / SB Date 3/11/05

4.4 Operations Shift Management Approval [signature] Date 3/14/05

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

______ (OEPM) Date ______

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.

ATTACHMENT 2

In-Progress Notes:
Issue Report 179235, Assignment 13

*
179235-13
*
Action: with assistance from Engineering, as required, review previous history to determine if the actual temperatures ever exceeded the new limit.
*
CLOSURE 11/20/03

Because their curves were accurate, this review excluded hydros and pressure tests, and heatups. Looking at cooldowns over the last three years, the actual bottom head temperatures were well above the curve for the given pressures. Based on this review, there does not appear to be any issue with previous history relative to the new limit. This item is closed.

A.M. Scott
Shift Operations Superintendent