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ComEd

November 12, 1999

SVP-99-193

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D C 20555

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

Subject: Request for an Amendment to Technical Specifications Section 3 /4.6.K, "Primary System Boundary", Section 3 /4.12.C "Special Test Exceptions", and Request for Exemption from 10 CFR 50.60 "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation"

Reference: (1) Letter from W. R. McCollum, Jr. (Duke Energy) to USNRC dated May 11, 1999

(2) Letter from D. E. Labarge (NRC) to W. R. McCollum, Jr. (Duke Energy) "Oconee Nuclear Station, Units 1, 2, and 3 RE: Exemption From the Requirements of 10 CFR Part 50, Section 50.60(a) (TAC NOS. MA5473, MA5474, and MA5475)" dated July 29, 1999

(3) Letter from D. E. Labarge (NRC) to W. R. McCollum, Jr. (Duke Energy) Amendment No. 307 to Facility Operating Licenses DPR-38, DPR-47, and DPR-55 dated October 1, 1999.

In accordance with 10 CFR 50.90, we request a change to Technical Specifications (TS) of Facility Operating License Nos. DPR-29, and DPR-30, for the Quad Cities Nuclear Power Station, Units 1 and 2, respectively. The proposed change is to TS Section 3/4.6.K "Primary System Boundary" and Section 3/4.12.C "Special Test Exceptions." In support of this TS change request, we are also requesting exemption from 10 CFR 50.60 "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation", in accordance with 10 CFR 50.12, "Specific exemptions."

The proposed change revises the pressure-temperature (P-T) limits by revising the heatup, cooldown and inservice test limitations for the Reactor Pressure Vessel (RPV) of each unit to a maximum of 32 Effective Full Power Years (EFPY). The use of 32 EFPY conservatively bounds both Units 1 and 2 that are currently at approximately 17 EFPY. Furthermore, the proposed change deletes the Special Test Exception, which

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provides for pressure testing at greater than 212° F in Mode 4. The proposed change provides potential radiation savings by increasing the effectiveness of inspectors in the containment at lower ambient temperature; potential outage schedule savings; and a reduction of burden on operators by eliminating the requirement to maintain reactor coolant system within a narrow temperature band above 212° F during pressure testing.

The proposed changes rely on recently approved American Society of Mechanical Engineers (ASME) methodology for determining allowable P-T limits. Specifically, the methodology used to generate the P-T curves is similar to the methodology utilized for the current P-T limits which were licensed in 1997. Several improvements were made including the incorporation of ASME Boiler and Pressure Vessel Code Cases N-588 "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels, Section XI, Division 1" and N-640 "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for ASME Section XI, Division 1". ASME Code Case N-588 allows the use of alternative procedures for defining the postulated flaw orientation and for calculating the applied stress intensity factors while ASME Code Case N-640 allows the use of an alternate fracture toughness curve. A similar TS amendment was recently approved for Duke Energy, Oconee Nuclear Station in Reference 3.

This TS change request also includes a request for an exemption in accordance with 10 CFR 50.12 from 10 CFR 50.60(a), "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation" requirement to meet the 10 CFR 50 Appendix G "Fracture Toughness Requirements." The requested exemption from 10 CFR 50.60(a), is to allow use of ASME Code Cases N-588 and N-640 as described below.

- ASME Code Case N-588 allows the use of alternative procedures for defining the orientation of postulated flaws in circumferential welds and for calculating the applied stress intensity factors of axial and circumferential flaws. The code case was approved for use by the appropriate ASME Boiler and Pressure Vessel Code (B&PV) Committee on December 12, 1997. A similar exemption request was granted to Duke Energy, Oconee Nuclear Station in Reference 2.

- ASME Code Case N-640 provides an alternate method for determining the fracture toughness of reactor pressure vessel materials for use in determining P-T Limits. The code case was approved for use by the appropriate ASME B&PV Code Committee on February 26, 1999. A similar exemption request was granted to Duke Energy, Oconee Nuclear Station in Reference 2.

Attachment F provides the detailed technical basis developed by Messrs. Warren Bamford and Bruce Bishop which justified the change in ASME B&PV Code, Section XI, Appendix G methodology permitted by ASME Code Case N-640.

Attachment G of this letter includes two General Electric Company reports containing proprietary information. Requests for withholding this information from disclosure, in accordance with 10 CFR 9.17(a)(4), 10 CFR 2.790(a)(4) and 2.790(d)(1), are provided in the preface of each report.

This request is subdivided as follows:

1. Attachment A gives a description and safety analysis of the proposed changes.

2. Attachment B includes the marked-up TS pages with the requested changes indicated.

3. Attachment C provides information supporting a finding of no significant hazards consideration in accordance with 10 CFR 50.92(c).

4. Attachment D provides information supporting an Environmental Assessment.

5. Attachment E, Exemption Request.

6. Attachment F provides technical basis for revised P-T Limit Curve Methodology.

7. Attachment G provides GE Nuclear Energy Reports, “Pressure Temperature Curves” Unit 2 GE-NE-B13-02057-00-0R1 and “Pressure Temperature Curves” Unit 1 GE-NE-B13-02057-00-02.

These proposed changes have been reviewed by the Plant Operations Review Committee and the Nuclear Safety Review Board in accordance with the Quality Assurance Program.

ComEd is notifying the State of Illinois of this request for changes to the TS by transmitting a copy of this letter and its attachments to the designated State Official.

Should you have any questions concerning his letter, please contact Mr. C.C. Peterson at (309) 654-2241, extension 3609.

Respectfully,

Joel P. Dimmette, Jr.

Joel P. Dimmette, Jr.
Site Vice President
Quad Cities Nuclear Power Station

Attachments:

Affidavit
Attachment A: Description and Safety Analysis for Proposed Changes
Attachment B: Marked-Up Pages for Proposed Changes
Attachment C: Information Supporting A Finding of No Significant Hazards Consideration
Attachment D: Information Supporting An Environmental Assessment

Attachment E: Exemption Request
Attachment F: Technical Basis for Revised P-T Limit Curve Methodology
Attachment G: GE Nuclear Energy Reports, Unit 2 GE-NE-B13-02057-00-01 and Unit 1 GE-NE-B13-02057-00-02

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Quad Cities Nuclear Power Station
Senior Reactor Analyst - IDNS

STATE OF ILLINOIS)

COUNTY OF ROCK ISLAND)

IN THE MATTER OF)

COMMONWEALTH EDISON (COMED) COMPANY) Docket Numbers

QUAD CITIES NUCLEAR POWER STATION UNITS 1 and 2) 50-254 and 50-265

SUBJECT: Request for Technical Specification Changes Section 3 /4.6.K, "Primary System Boundary" and Section 3 /4.12.C "Special Test Exceptions"

AFFIDAVIT

I affirm that the content of this transmittal is true and correct to the best of my knowledge, information and belief.

Joel P. Dimmette, Jr.
Joel P. Dimmette, Jr.
Site Vice President

Subscribed and sworn to before me, a Notary Public in and

for the State above named, this 12th day of

November, 1999.

"OFFICIAL SEAL"
LYNNE F. HAMILTON
Notary Public, State of Illinois
My Commission Expires 01/25/03

Lynne F. Hamilton
Notary Public

DESCRIPTION AND SAFETY ANALYSIS FOR PROPOSED CHANGES

A. SUMMARY OF PROPOSED CHANGES

In accordance with 10 CFR 50.90, ComEd is proposing changes to the Technical Specifications (TS) of Facility Operating Licenses DPR-29 and DPR-30, for Quad Cities Nuclear Power Station, Units 1 and 2, respectively. The proposed changes are to TS 3 /4.6.K "Primary System Boundary" and Section 3 /4.12.C, "Special Test Exceptions."

The proposed changes revise the heatup, cooldown and inservice test limitations for the Reactor Pressure Vessel of each unit to a maximum of 32 Effective Full Power Years (EFPY). The proposed changes also delete the TS Special Test Exception which allows for pressure testing at greater than 212° F in Mode 4.

The proposed changes are described in detail in Section E of this Attachment. The marked up TS pages are shown in Attachment B. Also, marked up bases pages are provided for completeness.

B. DESCRIPTION OF THE CURRENT REQUIREMENTS

Limiting Conditions for Operation (LCO's) and Surveillance Requirements provide for the reactor coolant system temperature and reactor pressure vessel metal temperature and pressure to be limited and monitored within the acceptable regions as shown on TS Figures 3.6.K-1 through 3.6.K-5, "Pressure-Temperature Limits For Pressure Testing-VALID 18 EFPY".

Special Test Exception 3 /4.12.C allows for operation in Mode 4 when reactor coolant temperature is in excess of 212° F provided that certain Operational Mode 3 requirements are met. These requirements include operability for secondary containment isolation, secondary containment integrity, secondary containment automatic isolation dampers, and standby gas treatment.

C. BASES FOR THE CURRENT REQUIREMENT

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 3.9.1.1.1 of the Quad Cities Nuclear Power Station (QCNPS) Updated Final Safety Analysis Report (UFSAR). During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

Four reactor pressure vessel (RPV) regions are considered for the development of the pressure-temperature (P-T) curves: 1) the reactor core beltline region; 2) the non-beltline region (other than the closure flange region and the bottom head region); 3) the closure flange region, and 4) the bottom head region. The reactor core beltline region is defined as the region of the reactor pressure vessel that directly surrounds the effective height of the active reactor core and is subject to a Reference Temperature Nil Ductility Transition (RT_{NDT}) adjustment to account for radiation embrittlement. The non-beltline, closure flange, and bottom head regions receive insufficient neutron fluence to necessitate an RT_{NDT} adjustment. These regions contain components which include; the RPV nozzles, closure flanges, top and bottom head plates, control rod drive penetrations, and shell plates that are not subjected to neutron radiation damage. Although the RPV closure flange and bottom head regions are non-beltline regions, they are treated separately for the development of the pressure-temperature curves to address 10 CFR Part 50 Appendix G "Fracture Toughness Requirements" requirements.

The purpose of the Special Test Exception LCO is to allow certain reactor coolant pressure tests to be performed in OPERATIONAL MODE 4 when the metallurgical characteristics of the RPV require pressure testing at temperatures greater than 212° F, which normally corresponds to OPERATIONAL MODE 3.

Pressure testing required by Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code are performed prior to startup after a refueling outage. The minimum temperatures at the required pressures allowed for these tests are determined from the RPV pressure and temperature limits required by TS Section 3.6.K, "Pressure/Temperature Limits." These limits are conservatively based on the fracture toughness of the reactor vessel, taking into account anticipated vessel neutron fluence. With increased RPV neutron fluence over time, the minimum allowable RPV temperature increases at a given pressure. Pressure testing will eventually be required with minimum reactor coolant temperatures that are greater than 212° F.

D. NEED FOR REVISION OF THE REQUIREMENT

The proposed changes rely on recently approved ASME methodology for determining allowable pressure-temperature (P-T) limits. Specifically, the methodology used to generate the P-T curves is similar to the methodology utilized for the current P-T Limits, which were approved for Quad Cities Nuclear Power Station, Units 1 and 2 in 1997.

The resultant benefits of the proposed changes include the following.

- Reduction in the challenges to operators in conducting pressure testing with reactor coolant in excess of 212° F and maintaining the reactor coolant within a narrow temperature band.
- Personnel safety by conducting inspections at lower coolant temperatures, eliminates steam vapor hazards.
- Potential dose savings by increasing the effectiveness of inspectors in the primary containment at lower ambient temperatures.
- Potential outage critical path schedule savings by the reduction of time to achieve reactor pressure vessel pressure and temperature requirements for testing.
- Improved leak detection afforded by observation of water leakage versus observation of steam vapor.
- Reduction in the potential to spread contamination in containment with the absence of steam vapor.

E. DESCRIPTION OF THE PROPOSED CHANGES

The proposed changes revise TS Figures 3.6.K-1 through Figure 3.6.K-5. The proposed Figures 3.6.K-1 through 3.6.K-3 are bounding P-T curves for the Unit 1 and Unit 2 Reactor Pressure Vessels. These proposed changes also delete the Special Test Exception for Inservice Leak and Hydrostatic testing operation.

Bases changes that are affected by these proposed changes are also included here for completeness.

F. SAFETY ANALYSIS OF THE PROPOSED CHANGES

The proposed changes to the P-T limits have been developed in accordance with the technical requirements of the ASME B&PV Code, Section XI, Appendix G as modified by ASME Code Cases N-588 and N-640.

ASME Code Case N-588

The current ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Appendix G approach requires the consideration of an axially oriented flaw in circumferential welds for the purpose of calculating pressure-temperature limits. Postulating the ASME Appendix G reference flaw in a circumferential weld is physically unrealistic because the length of the reference flaw is 1.5 times the reactor pressure vessel (RPV) thickness, and is much longer than the width of the vessel circumferential welds. The fabrication of reactor pressure vessels (RPVs) for nuclear power plant operation involved precise welding procedures and controls designed to optimize the resulting weld microstructure and to provide the required material properties. These procedural controls were also designed to minimize defects that could be introduced into the weld during the fabrication process. Subsequent non-destructive examinations were conducted which confirmed that the welds met the preservice inspection criteria. Experience with the repair of weld indications found during pre-service inspection, and data taken from destructive examination of actual RPV welds, confirm that any remaining flaws are small, laminar in nature, and do not cross traverse to the weld bead orientation. Because of this, any defects potentially introduced during the fabrication process and not detected during the subsequent non-destructive examinations should only be oriented along the direction of weld fabrication. For circumferential welds, this indicates a postulated defect with a circumferential orientation.

Using ASME Code Case N-588 to determine P-T limits in conjunction with ASME B&PV Code, Section XI, Appendix G, provides appropriate and conservative procedures to determine limiting maximum postulated defects and to consider those defects in the determination of the P-T limits. The application of this code case maintains the margin of safety for circumferential welds equivalent to that originally contemplated for plates/forgings and axial welds.

ASME Code Case N-640

The proposed P-T Limits have been developed using the K_{Ic} fracture toughness curve shown on ASME B&PV Code, Section XI, Appendix A, Figure A-2200-1, in lieu of the K_{Ia} fracture toughness curve of ASME B&PV Code, Section XI, Appendix G, Figure G-2210-1, as the lower bound of fracture toughness. The other margins inherent with the ASME B&PV Code, Section XI, Appendix G process to determine P-T limit curves remain unchanged.

Use of the K_{Ic} curve in determining the lower bound fracture toughness in the development of P-T operating limits is technically more correct than the K_{Ia} curve. The K_{Ic} curve appropriately implements the static initiation fracture toughness because the controlled heatup and cooldown process limits the rate at which stress is developed in the RPV wall to rates that are more appropriate for static initiation fracture toughness.

When the K_{Ia} curve was codified in 1974, the initial conservatism of the K_{Ia} curve was necessary due to limited experience and knowledge of RPV material fracture toughness. The conservatism also provided margin thought to be necessary to cover other uncertainties and the postulated material effects of operating loads.

Since 1974, additional knowledge has been gained from examination and testing of reactor pressure vessels that has reduced many of these uncertainties and resolved the postulated material effects from operating loads. Since the original formulation of the K_{Ia} and K_{Ic} curves in 1972, the fracture toughness database has been increased by orders of magnitude, and both K_{Ia} and K_{Ic} ASME B&PV Code, Section XI curves remain lower bound curves. The additional data has significantly reduced the uncertainties associated with material fracture toughness. The added data ensures the ASME B&PV Code, Section XI K_{Ic} curve statistically bounds the data, as presented in Figure 1 of Attachment F. The new information indicates the lower bound on fracture toughness provided by the K_{Ic} curve is extremely conservative. This lower bound on fracture toughness provides a greater margin of safety beyond that which is required to protect public health and safety from a potential reactor pressure vessel failure.

Details of the evaluations performed to calculate the P-T limits using this methodology are provided in Attachment G.

As a conservative measure, the bounding 32 EFPY neutron fluence value of 5.1×10^{17} n/cm^2, from the reactor pressure vessels at both Dresden and Quad Cities Nuclear Power stations, was used to adjust the beltline material RT_{NDT} values. The beltline materials best estimate chemistry initial reference temperatures and Sigma (I) term are in agreement with our letter, R. M. Krich (ComEd) to NRC dated July 30, 1998, "Response to Request for Additional Information Regarding Reactor Pressure Vessel Integrity."

G. IMPACT ON PREVIOUS SUBMITTALS

ComEd has reviewed the proposed changes regarding impact on any previous submittals and has determined that there is no impact on any outstanding previous submittals.

H. SCHEDULE REQUIREMENTS

We request approval of this amendment prior to January 20, 2000 to support activities in the Unit 2 cycle 15 refueling outage currently scheduled to begin on January 20, 2000.