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Nuclear

10 CFR 50.90

RS-01-175

August 29, 2001

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

Dresden Nuclear Power Station, Units 2 and 3
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
Facility Operating License Nos. DPR-29 and DPR-30
NRC Docket Nos. 50-254 and 50-265

Subject: Supplement to Request for License Amendment for Power Uprate Operation

References: (1) Letter from R. M. Krich (Commonwealth Edison Company) to U. S. NRC, "Request for License Amendment for Power Uprate Operation," dated December 27, 2000

(2) Letter from R. M. Krich (Exelon Generation Company, LLC) to U. S. NRC, "Supplement to Request for License Amendment for Power Uprate Operation," dated April 13, 2001

(3) Letter from K. A. Ainger (Exelon Generation Company, LLC) to U. S. NRC, "Additional Plant Systems Information Supporting the License Amendment Request to Permit Uprated Power Operation at Dresden Nuclear Power Station and Quad Cities Nuclear Power Station," dated August 13, 2001

(4) Letter from R. M. Krich (Commonwealth Edison Company) to U. S. NRC, "Request for Technical Specifications Changes, Transition to General Electric Fuel," dated September 29, 2000

Pursuant to 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company (EGC), LLC, formerly Commonwealth Edison (ComEd) Company, is requesting additional changes to the Operating License (OL) and Technical Specifications (TS) relative to the changes proposed in References 1 and 2 for the Dresden Nuclear Power Station (DNPS), Units 2 and 3, and the Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2. These proposed changes include the following.

A001

- A revision to the proposed credit for containment overpressure specified in the OL for DNPS, Unit 2.
- Deletion of the definition of maximum fraction of limiting power density (MFLPD) from TS Section 1.1, "Definitions," for DNPS, Units 2 and 3.
- Revision of the allowable value for the reactor vessel water level – low low function in TS Table 3.3.6.1-1, "Primary Containment Isolation Instrumentation," for DNPS, Units 2 and 3.
- Revision of the allowable value for the reactor vessel water level – low function in Table 3.3.6.2-1, "Secondary Containment Isolation Instrumentation," for DNPS, Units 2 and 3 and QCNPS, Units 1 and 2, and Table 3.3.7.1-1, "Control Room Emergency Ventilation (CREV) System Isolation Instrumentation," for QCNPS, Units 1 and 2.
- Revision of the allowable value for main steam flow – high in Table 3.3.7.1-1, "Control Room Emergency Ventilation (CREV) System Isolation Instrumentation," for QCNPS, Units 1 and 2.

In References 1 and 2, ComEd submitted various proposed OL and TS changes for DNPS and QCNPS to allow operation with an extended power uprate (EPU). One of the proposed changes was a revision to the proposed credit for containment overpressure specified in the OLs for DNPS, Units 2 and 3. In Reference 3, in response to NRC questions regarding this proposed change, EGC indicated that it would revise the proposed values for containment overpressure. This supplemental amendment request provides the revised proposed values for DNPS Unit 2. As discussed in Reference 3, revised proposed values for DNPS, Unit 3 and QCNPS, Units 1 and 2 will be provided in a future submittal.

In Reference 4, ComEd submitted various proposed TS changes to support a change in fuel vendors from Siemens Power Corporation, now Framatome, to General Electric (GE) Company, and a transition to GE14 fuel. One of the proposed changes was to include the definition of MFLPD in the DNPS TS. However, once the EPU proposed changes are approved, the use of limits related to MFLPD is no longer required. Since the GE14 proposed changes will be approved before approval of the EPU proposed changes, this supplemental amendment request proposes deletion of the definition of MFLPD from the TS for DNPS, Units 2 and 3.

During review of instrumentation setpoints for the EPU project, it was determined that the allowable value for the main steam line isolation on reactor vessel water level – low low was based on an assumed instrument temperature range that was inconsistent with the assumed temperature ranges for instruments in the same loop. The EPU analyses did not change the temperature range for this instrument or the analytical limit for the low low water level function. However, to correct this inconsistency in the assumed temperature range, a change is proposed in the allowable value for this function.

The EPU proposed changes identified the allowable value changes in TS Tables 3.3.1.1-1, "Reactor Protection System Instrumentation," and 3.3.6.1-1, "Primary Containment Isolation Instrumentation." During implementation reviews for the EPU, it was recognized that the same allowable value changes are required in Table 3.3.6.2-1, "Secondary Containment Isolation Instrumentation," for DNPS and QCNPS, and Table 3.3.7.1-1, "Control Room Emergency Ventilation (CREV) System Isolation

Instrumentation," for QCNPS. Upon discovery of this oversight, EGC initiated a corrective action program condition report (CR) to determine the cause and corrective actions for the oversight. We have determined that, while the EPU technical reviews were thorough, there was an inadequate focus on assuring completeness of the TS changes. Subsequently, we have completed additional reviews of the TS and we have not identified any additional changes needed to support the EPU amendment request beyond those described in this supplemental request.

EGC has determined that with the exception of the main steam line isolation function that occurs on reactor vessel water level – low, low (DNPS only), the information contained in this letter does not affect the information provided in Reference 1 supporting a finding of no significant hazards consideration.

This supplement to the Reference 1 and 2 amendment requests contains separate enclosures for DNPS and QCNPS. Each enclosure is subdivided as follows.

1. Attachment A contains a detailed description of the additional proposed changes.
2. Attachment B provides the proposed mark-ups to the TS and OL (DNPS Unit 2 only) for the proposed changes.
3. Attachment C provides a supplement to the information supporting a finding of no significant hazards consideration for the proposed changes in accordance with 10 CFR 50.92(c), "Issuance of Amendment," for DNPS only.
4. Attachment D provides information supporting an Environmental Assessment for DNPS only.

The proposed changes have been reviewed by the Plant Operations Review Committees and approved by the Nuclear Safety Review Boards at DNPS and QCNPS in accordance with the Quality Assurance Program.

EGC is notifying the State of Illinois of this license amendment request by transmitting a copy of this letter and its attachments to the designated State Official.

EGC requests that these additional changes be reviewed and approved as part of the proposed changes for power uprate operation previously submitted in References 1 and 2.

Should you have any questions related to this request, please contact Mr. Allan R. Haeger at (630) 657-2807.

Respectfully,

K. A. Ainger

K. A. Ainger
Director – Licensing
Mid-West Regional Operating Group

Attachments:

Affidavit
Enclosure 1: Dresden Nuclear Power Station
Attachment A: Description and Summary Safety Analysis for Proposed Changes
Attachment B: Marked-Up TS and OL Pages for Proposed Changes
Attachment C: Information Supporting a Finding of No Significant Hazards Consideration
Attachment D Information Supporting an Environmental Assessment

Enclosure 2: Quad Cities Nuclear Power Station
Attachment A: Description and Summary Safety Analysis for Proposed Changes
Attachment B: Marked-Up TS Pages for Proposed Changes

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Dresden Nuclear Power Station
NRC Senior Resident Inspector – Quad Cities Nuclear Power Station
Office of Nuclear Facility Safety – Illinois Department of Nuclear Safety

STATE OF ILLINOIS	)	
COUNTY OF DUPAGE	)	
IN THE MATTER OF:	)	
EXELON GENERATION COMPANY, LLC	)	**Docket Numbers**
DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3	)	**50-237 and 50-249**
QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2	)	**50-254 and 50-265**

SUBJECT: Supplement to Request for License Amendment for Power Uprate Operation

AFFIDAVIT

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

K. A. Ainger

K. A. Ainger
Director – Licensing
Mid-West Regional Operating Group

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

for the State above named, this 29th day of

August, 2001

" OFFICIAL SEAL "
Timothy A. Byam
Notary Public, State of Illinois
My Commission Expires 11/24/2001

Timothy Byam
Notary Public

ENCLOSURE 1 - ATTACHMENT A

Supplement to Request For Power Uprate Operation Dresden Nuclear Power Station, Units 2 and 3

DESCRIPTION AND SUMMARY SAFETY ANALYSIS FOR PROPOSED CHANGES

A. SUMMARY OF PROPOSED CHANGES

Pursuant to 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company (EGC), LLC, formerly Commonwealth Edison (ComEd) Company, is requesting additional changes to the Technical Specifications (TS) relative to the changes proposed in References I.1 and I.2 for the Dresden Nuclear Power Station (DNPS), Units 2 and 3 and Operating License (OL) for DNPS, Unit 2. These proposed changes include the following.

- A revision to the proposed credit for containment overpressure specified in the OL for DNPS, Unit 2.
- Deletion of the definition of maximum fraction of limiting power density (MFLPD) from TS Section 1.1, "Definitions," for DNPS, Units 2 and 3.
- Revision of the allowable value for the main steam line isolation reactor vessel water level – low low function in TS Table 3.3.6.1-1, "Primary Containment Isolation Instrumentation," for DNPS, Units 2 and 3.
- Revision of the allowable value for the reactor vessel water level – low function in Table 3.3.6.2-1, "Secondary Containment Isolation Instrumentation," for DNPS, Units 2 and 3.

In References I.1 and I.2, ComEd submitted various proposed OL and TS changes for DNPS to allow operation with an extended power uprate (EPU). One of the proposed changes was a revision to the proposed credit for containment overpressure specified in the OLs for DNPS, Units 2 and 3. In Reference I.3, in response to NRC questions regarding this proposed change, EGC indicated that it would revise the proposed value for containment overpressure. This supplement to our amendment request provides the revised proposed values for DNPS, Unit 2. As discussed in Reference I.3, revised proposed values for DNPS, Unit 3 will be provided in a future submittal.

In Reference I.4, ComEd submitted various proposed TS changes to support a change in fuel vendors from Siemens Power Corporation, now Framatome, to General Electric (GE) Company, and a transition to GE14 fuel. One of the proposed changes was to include the definition of MFLPD in the DNPS TS. However, once the EPU proposed changes are approved, limits related to MFLPD are no longer required. Since the GE14 proposed changes will be approved before approval of the EPU proposed changes, this supplement to our amendment request proposes deletion of the definition of MFLPD from the TS for DNPS, Units 2 and 3.

During review of instrumentation setpoints for the EPU project, it was determined that the allowable value for the main steam line isolation on reactor vessel water level – low low was based on an assumed instrument temperature range that was inconsistent with other assumed temperature ranges for instruments in the same loop. The EPU analyses did not change the temperature range for this instrument or the analytical limit for the low low water level function. However, to correct this inconsistency in the assumed temperature range, a change is proposed in the allowable value for this function.

Reference I.1 proposed changes to allowable values in TS Tables 3.3.1.1-1, "Reactor Protection System Instrumentation," and 3.3.6.1-1, "Primary Containment Isolation Instrumentation." During implementation reviews for the EPU, it was recognized that the same allowable value change is required in Table 3.3.6.2-1, "Secondary Containment Isolation Instrumentation."

B. DESCRIPTION OF THE CURRENT REQUIREMENTS

B.1 OL Condition on Containment Overpressure

DNPS, Unit 2 has an OL condition associated with TS Amendment 157 that states the following.

> "The license is amended to authorize changing the UFSAR to allow credit for containment overpressure as detailed below, to assure adequate Net Positive Suction Head is available for low pressure Emergency Core Cooling System pumps following a design basis accident."

Time (seconds)	Containment Pressure (PSIG)
0-240	9.5
240-480	2.9
480-6000	1.9
6000-accident end	2.5

B.2 TS Section 1.1, "Definitions"

In the Reference I.4 amendment request, ComEd proposed to add the definition of MFLPD to the TS.

B.3 TS Section 3.3.6.1, "Primary Containment Isolation Instrumentation"

Table 3.3.6.1-1, "Primary Containment Isolation Instrumentation," Function 1.a, identifies the allowable value for the main steam line isolation that occurs on reactor vessel water level – low low. The allowable value is $\geq$ -56.77 inches.

B.4 TS Section 3.3.6.2, "Secondary Containment Isolation Instrumentation"

Table 3.3.6.2-1, "Secondary Containment Isolation Instrumentation," Function 1, identifies the allowable value for the reactor vessel water level – low function. The allowable value is $\geq$ 10.24 inches.

C. BASES FOR THE CURRENT REQUIREMENTS

C.1 OL Condition on Containment Overpressure

To ensure that there is adequate net positive suction head (NPSH) to support the operation of the emergency core cooling system (ECCS) pumps during design basis accident (DBA) conditions, the analyses take credit for containment overpressure. This allowance was approved in TS Amendment 157 for DNPS Unit 2 (Reference I.5).

C.2 TS Section 1.1, "Definitions"

The definition of MFLPD is included in the Reference I.4 amendment request to support the use of reactor thermal limits using this GE parameter.

C.3 TS Section 3.3.6.1, "Primary Containment Isolation Instrumentation"

The function of the primary containment isolation on low-low reactor vessel water level is to limit fission product release during and following postulated DBAs.

C.4 TS Section 3.3.6.2, "Secondary Containment Isolation Instrumentation"

A low reactor vessel water level indicates that the capability to cool the fuel may be threatened. Should the reactor vessel water level decrease too far, fuel damage could result. An isolation of the secondary containment and actuation of the standby gas treatment system are initiated in order to minimize the potential of an offsite release.

D. NEED FOR REVISION OF THE REQUIREMENTS

D.1 OL Condition on Containment Overpressure

The analysis associated with the postulated LOCA at increased power levels results in an increase in suppression pool water temperature. Because of the increase in water temperature, the need for additional credit for containment overpressure to maintain adequate NPSH for the ECCS pumps has been identified.

D.2 TS Section 1.1, "Definitions"

One of the proposed changes in the Reference I.4 amendment request for GE14 fuel was to include the definition of MFLPD in the DNPS TS. The Reference I.1 amendment request for EPU proposed to delete the thermal limits related to MFLPD and substitute power and flow dependent limits known as the Average Power Range Monitor (APRM) Rod Block Monitor (RBM) TS changes (i.e., ARTS changes). Therefore, once the EPU proposed changes are approved, the use of limits related to MFLPD is no longer required. Since the GE14 proposed changes will be approved before approval of the EPU proposed changes, this supplement to our amendment request proposes deletion of the definition of MFLPD from the TS for DNPS, Units 2 and 3.

D.3 TS Section 3.3.6.1, "Primary Containment Isolation Instrumentation"

During a review of instrumentation setpoints for the EPU project, it was determined that the allowable value for the main steam line isolation on reactor vessel water level – low low was based on an assumed instrument temperature range that was inconsistent with other assumed temperature ranges for instruments in the same loop. The EPU analyses did not change the temperature range for this instrument or the analytical limit for the low low reactor vessel water level function. However, to correct this inconsistency in the assumed temperature range, a change is proposed in the allowable value for this function.

D.4 TS Section 3.3.6.2, "Secondary Containment Isolation Instrumentation"

The loss of feedwater transient was reanalyzed under EPU conditions. Due to increased core heat generation as a result of EPU, the reactor pressure vessel (RPV) water level decreases more rapidly in this transient. Therefore, the Reference I.1 amendment request proposed to lower the reactor vessel low water level scram setpoint in order to increase the potential for recovery before reaching the scram setpoint and thus prevent unnecessary challenges to safety systems and provide additional time for operator action.

The proposed change to the allowable value for the secondary containment isolation function on reactor vessel water level – low is directly related to the proposed change for the reactor scram setpoint reduction. To maintain the secondary containment isolation function at the same level as the reactor scram, the allowable value for TS Table 3.3.6.2-1, Function 1, must also be revised.

E. DESCRIPTION OF THE PROPOSED CHANGES

E.1 OL Condition on Containment Overpressure

The allowance for containment overpressure in the DNPS, Unit 2 OL condition is revised to state the following.

> "The license is amended to authorize changing the UFSAR to allow credit for containment overpressure as detailed below, to assure adequate Net Positive Suction Head is available for low pressure Emergency Core Cooling System pumps following a design basis accident."

Period	Requested Credit (psi)
0 – 290 sec	9.5
290 - 5,000 sec	4.8
5,000 – 30,000 sec	6.6
30,000 - 40,000 sec	6.0
40,000 - 45,500 sec	5.4
45,500 - 52,500 sec	4.9
52,500 - 60,500 sec	4.4
60,500 - 70,000 sec	3.8

70,000 - 84,000 sec	3.2
84,000 - 104,000 sec	2.5
104,000 - 136,000 sec	1.8
136,000 sec – accident end	1.1

E.2 TS Section 1.1, "Definitions"

The definition of MFLPD is deleted.

E.3 TS Section 3.3.6.1, "Primary Containment Isolation Instrumentation"

The allowable value for TS Table 3.3.6.1-1, Function 1.a, is revised from ≥ -56.77 inches to ≥ -56.34 inches.

E.4 TS Section 3.3.6.2, "Secondary Containment Isolation Instrumentation"

The allowable value for Table 3.3.6.2-1, Function 1, is revised from ≥ 10.24 inches to ≥ 2.65 inches.

F. SUMMARY SAFETY ANALYSIS OF THE PROPOSED CHANGES

F.1 OL Condition on Containment Overpressure

Additional credit for containment overpressure is required because the suppression pool temperature increases at a faster rate and peaks at a higher value compared to the pre-EPU conditions during a loss of coolant accident (LOCA). Because vapor pressure increases as the suppression pool temperature increases, the net positive suction head available (NPSHa) for each ECCS pump is reduced. To offset this reduction in NPSHa, more overpressure credit is required. More overpressure is also available, since the containment and suppression pool pressures also increase at a faster rate and peak at a higher value than before EPU.

<u>Containment Response</u>
The design basis accident (DBA) LOCA containment response for NPSH evaluations is analyzed for two time periods: short term (i.e., before 600 seconds), and long term (i.e., after 600 seconds). The long term temperature and pressure conditions of the suppression pool are determined based on assumptions that maximize the pool temperature and minimize the overpressure, including operation of containment sprays and vacuum breakers. Specific assumptions include the following.

1. The DBA LOCA is an instantaneous double-ended guillotine break of the recirculation suction line at the reactor vessel nozzle safe-end to pipe weld. The effective break area is 4.261 ft^2.

2. The reactor is operating at 102% of EPU (i.e., 3016 megawatts-thermal (MWt)) with an initial reactor pressure of 1005 pounds per square inch - gauge (psig). Concurrent with occurrence of the break, reactor scram occurs.

3. The reactor core power includes fission energy, fuel stored energy, metal-water reaction energy and decay heat calculated in accordance with American Nuclear Society (ANS) Standard 5.1-1979, "Decay Heat Source Term for Containment Long-Term Pressure and Temperature Analysis," for a 24 month fuel cycle with two sigma adder.

4. The initial suppression pool water volume corresponds to the low water level to maximize the suppression pool temperature response.

5. Containment cooling is achieved by operating one low pressure coolant injection (LPCI)/ containment cooling loop at 600 seconds in the containment spray mode, with drywell and wetwell sprays. This minimizes the containment pressure response, since cold water sprays will bring down the pressure.

The short term conditions are based on similar assumptions, with the following exceptions.

1. There is a single failure of the loop selection logic. Consequently, the flow from all four LPCI pumps goes into the broken recirculation loop and subsequently discharges into the drywell directly. The maximum runout flow rate is assumed.

2. Both core spray pumps are operating with the maximum flow rate.

Procedures

Existing plant emergency operating procedures include cautions concerning exceeding ECCS pump NPSH limits. The procedures also contain ECCS pump curves of pump flow versus torus pressure and temperature conditions. The same cautions and NPSH curves are included in the emergency operating procedures that control use of containment sprays. Thus, the operators have sufficient procedural direction to control both ECCS pump flow and containment pressure within limits.

Methodology and Results for DNPS

That the proposed overpressure credit is based on the methodology previously approved for DNPS in a 1997 license amendment regarding containment overpressure (Reference I.5). This methodology followed the original design basis of one ECCS suction strainer completely blocked, with the remaining three strainers in a clean condition. The head loss across the three clean strainers was assumed to be the same as the head loss for the original suction strainers, although those strainers were subsequently replaced with higher capacity strainers. Thus, the assumed head loss is slightly higher than the actual head loss expected with the new strainers. This assumption maintains consistency with the basis for approval of the Reference I.5 amendment request. We also expect that the head loss used to develop the requested overpressure will result in adequate overpressure when compared to the results of future calculations of suction strainer head loss discussed in the paragraph below.

NRC Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors," requested that licensees calculate suction strainer head loss assuming that debris from the primary containment is distributed across all of the ECCS suction strainers. In accordance with this request, we will perform calculations of the suction strainer head loss and will submit a description of the calculational methods and the results to the NRC.

NPSH calculations have been performed for EPU conditions with the strainer head loss assumptions described above for two short term and two long term flow conditions. The limiting short term ECCS flow case is all four LPCI pumps and both core spray pumps operating at maximum flow conditions. The limiting long term ECCS flow rate is the same as in the 1997 calculations that formed the basis of the currently approved overpressure credit. This limiting flow rate is 19,000 gallons per minute (gpm) distributed as follows: two core spray pumps operating at 4,500 gpm each, one LPCI pump at 5,000 gpm, and two additional LPCI pumps at 2,500 gpm each. This flow case is significantly more than the minimum long term flow of 9,750 gpm required to maintain adequate core and containment cooling after EPU. The minimum flow case, one core spray pump operating at 4,750 gpm and one LPCI pump operating at 5,000 gpm, is the other case analyzed in the calculations.

The graphs showing the results of the ECCS NPSH calculations for the limiting short term and long term flow cases are provided in Figures 1 and 2. Core spray flow is the limiting NPSH case in the short term, and LPCI flow is limiting for NPSH in the long term. Figures 1 and 2 also show NPSH required (NPSHr) for both the old strainer and new strainer cases (e.g., one blocked, three clean). The higher head loss of the old strainers, as indicated above, is the basis for the requested overpressure.

In the short term, there is a period from approximately 290 seconds to 600 seconds during which some ECCS pump cavitation may occur, since the available NPSH is less than the required NPSH. This period is after the time at which the peak cladding temperature (PCT) has been reached at approximately 240 seconds. Prior to 290 seconds, the requested overpressure ensures that adequate NPSH is available to meet the core cooling requirements assumed in the PCT calculations. After 600 seconds, ECCS pump throttling restores adequate NPSH. Pump cavitation for the brief time from 290 seconds to 600 seconds is not of concern due to the short duration of the cavitation, as discussed in Reference I.5.

The long term overpressure curves are plotted out to 200,000 seconds. From this point, NPSHa and NPSHr both vary directly as a function of the vapor pressure. The result is that both decrease in parallel fashion, maintaining a margin between available and required NPSH.

F.2 TS Section 1.1, "Definitions"

The Reference I.1 amendment request describes the safety analysis for removing the MFLPD limit from the TS. With the approval of the Reference I.1 proposed changes, the removal of the definition of MFLPD is an administrative change, since no other TS items make use of this definition.

F.3 TS Section 3.3.6.1, "Primary Containment Isolation Instrumentation"

The allowable value proposed does not represent a change in the analytical limit assumed in the safety analysis for the low-low reactor vessel water level. The allowable value was recalculated following the review of instrumentation setpoints described in Section D.3 above, using a wider and thus more conservative temperature range for the low-low reactor vessel water level instrumentation. This allowable value was calculated in accordance with the Exelon Nuclear Mid-West Regional Operating Group setpoint methodology procedure NES-EIC-20.04, "Analysis of Instrument Channel Setpoint Error and Instrument Loop Accuracy," Revision 3.

The current allowable values for other functions in the TS related to reactor vessel water level low and low-low have been determined using the appropriate temperature ranges and require no adjustment.

F.4 TS Section 3.3.6.2, "Secondary Containment Isolation Instrumentation"

The reactor vessel water level - low function is assumed in the analysis of the recirculation line break and is credited in the loss of normal feedwater flow event. The reactor scram associated with the function reduces the amount of energy required to be absorbed and, along with the actions of the emergency core cooling systems, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors." The associated secondary containment isolation function is initiated in order to minimize the potential of an offsite release. The allowable value for the secondary containment isolation function is chosen to be the same as the allowable value for the reactor protection system setpoint and is not analyzed separately. The proposed change in the reactor scram setpoint does not result in a change to the current safety analyses. Thus, the change in the allowable value for the secondary containment isolation function continues to ensure that any offsite releases are within the limits calculated in the safety analysis.

G. IMPACT ON PREVIOUS SUBMITTALS

All submittals currently under review by the NRC were evaluated to determine the impact of these proposed changes. These proposed changes supplement the changes proposed to support uprated power operation at DNPS in Reference I.1.

In addition, these proposed changes affect the proposed changes submitted in Reference I.6, which requested that the NRC consider the proposed changes to the reactor vessel water level – low setpoint separately from the EPU amendment request. The additional proposed change being submitted in this amendment request is also being submitted to the NRC separately as a supplement to the Reference I.6 amendment request.

No other submittals currently under review by the NRC are affected by the information presented in this supplement to our Reference I.1 license amendment request.

H. SCHEDULE REQUIREMENTS

We request that these proposed changes be reviewed and approved as part of the proposed changes for power uprate operation previously submitted in References I.1 and I.2.

I. REFERENCES

1. Letter from R. M. Krich (ComEd) to U. S. NRC, "Request for License Amendment for Power Uprate Operation," dated December 27, 2000

2. Letter from R. M. Krich (EGC) to U. S. NRC, "Supplement to Request for License Amendment for Power Uprate Operation," dated April 13, 2001

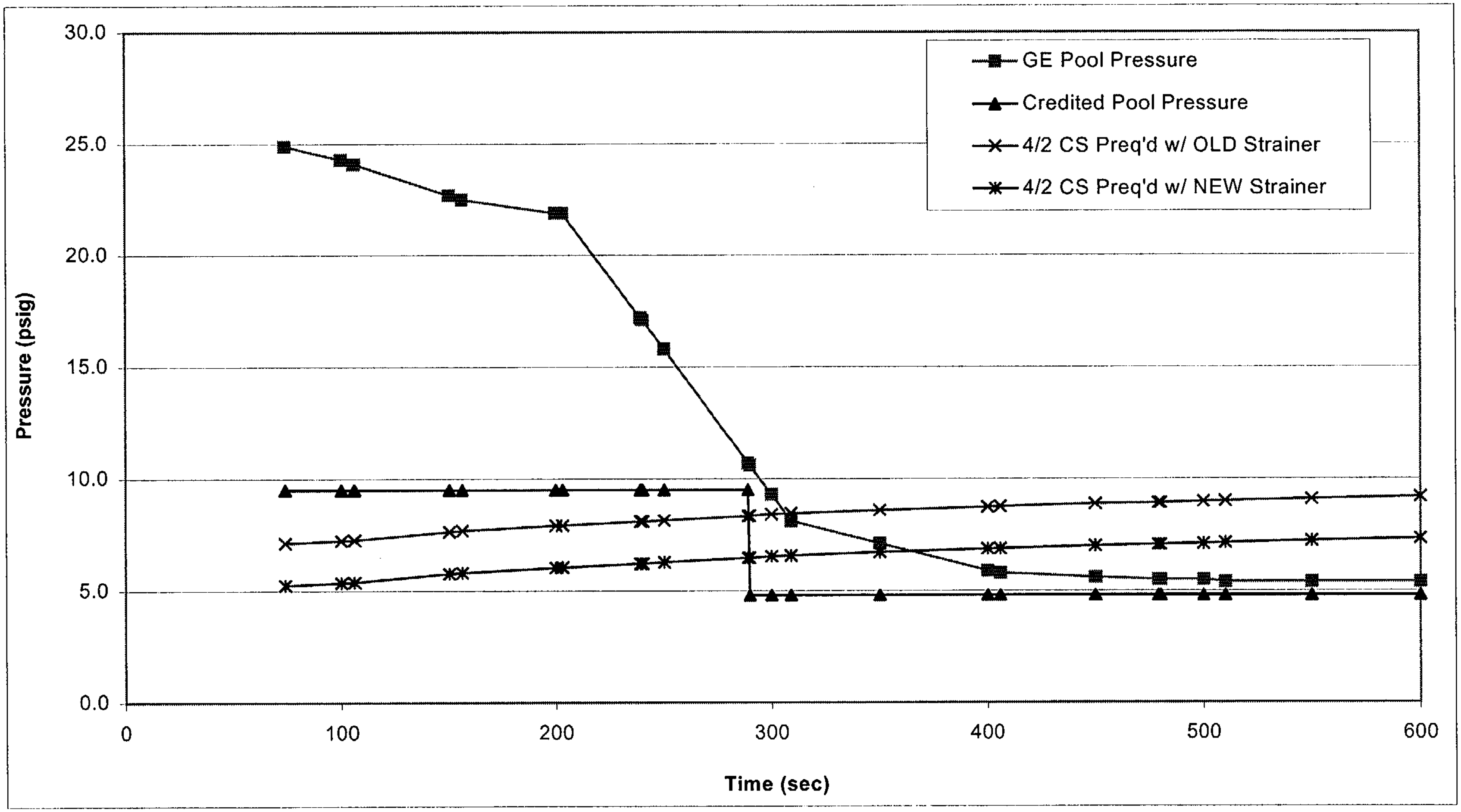
3. Letter from K. A. Ainger (EGC, LLC) to U. S. NRC, "Additional Plant Systems Information Supporting the License Amendment Request to Permit Uprated Power Operation at Dresden Nuclear Power Station and Quad Cities Nuclear Power Station," dated August 13, 2001

4. Letter from R.M. Krich (ComEd) to U. S. NRC, "Request for Technical Specifications Changes, Transition to General Electric Fuel," dated September 29, 2000

5. Letter from U. S. NRC to I. Johnson (ComEd), "Issuance of Amendments," dated April 30, 1997

6. Letter from R. M. Krich (EGC) to U. S. NRC, "Request for License Amendment for Reactor Vessel Low Water Level Setpoint," dated February 22, 2001

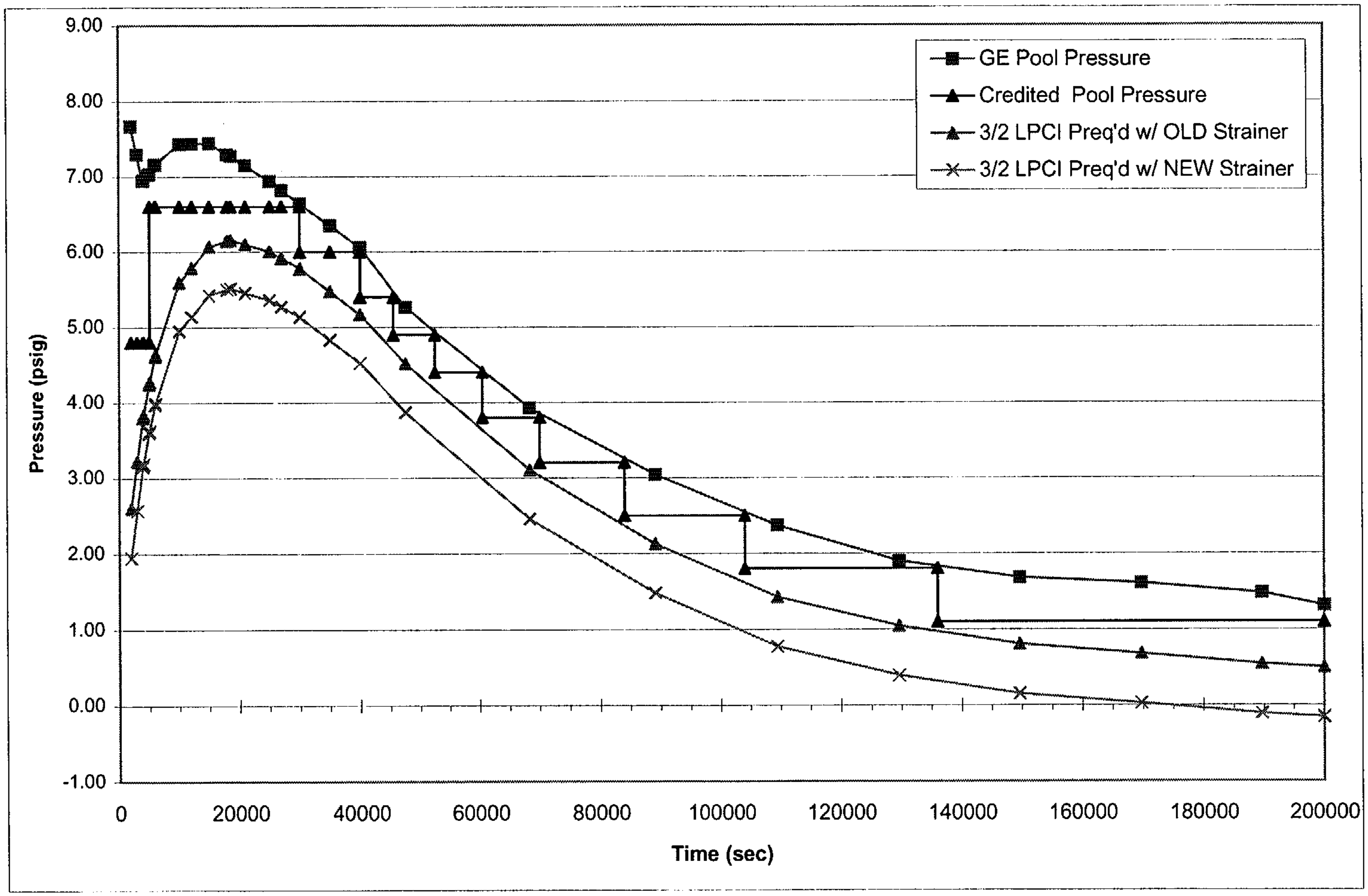
Figure 1
Short Term NPSH Curves



ENCLOSURE 1 - ATTACHMENT A

Supplement to Request For Power Uprate Operation
Dresden Nuclear Power Station, Units 2 and 3

Figure 2
Long Term NPSH Curves



ENCLOSURE 1 - ATTACHMENT B

Supplement to Request For Power Uprate Operation
Dresden Nuclear Power Station, Units 2 and 3

MARKED-UP OPERATING LICENSE PAGE FOR PROPOSED CHANGES

<u>REVISED PAGE</u>
Appendix B, Page 1 (DPR-19)

MARKED-UP TS PAGES FOR PROPOSED CHANGES

<u>REVISED PAGES</u>
1.1-4
3.3.6.1-5
3.3.6.2-4

ENCLOSURE 1 - ATTACHMENT B

Supplement to Request For Power Uprate Operation
Dresden Nuclear Power Station, Units 2 and 3

MARKED-UP OPERATING LICENSE PAGE FOR PROPOSED CHANGES

<u>REVISED PAGE</u>
Appendix B, Page 1 (DPR-19)

MARKED-UP TS PAGES FOR PROPOSED CHANGES

<u>REVISED PAGES</u>
1.1-4
3.3.6.1-5
3.3.6.2-4

APPENDIX B

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. DPR-19

The licensee shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Condition	Implementation Date
157	The license is amended to authorize changing the UFSAR to allow credit for containment overpressure as detailed below, to assure adequate Net Positive Suction Head is available for low pressure Emergency Core Cooling System pumps following a design basis accident.	Effective as of the issuance of Amendment No. ~~157~~ and shall be implemented within 30 days.

~~Time (seconds)~~	~~Containment Pressure (PSIG)~~
~~0-240~~	~~9.5~~
~~240-480~~	~~2.9~~
~~480-6000~~	~~1.9~~
~~6000-accident end~~	~~2.5~~

Replace with Insert

Amendment Number	Additional Condition	Implementation Date
157	The EOPs shall be changed to alert operator to NPSH concerns and to make containment spray operation consistent with the overpressure requirements for NPSH.	Shall be implemented within 30 days after issuance of Amendment No. 157.
160	This amendment authorizes the licensee to incorporate in the Updated Final Safety Analysis Report (UFSAR), the description of the Reactor Coolant System design pressure, temperature and volume that was removed from Technical Specification Section 5.4, and evaluated in a safety evaluation dated June 12, 1997.	30 days from the date of issuance of Amendment No. 160.
163	The licensee shall review the Dresden Operation Annunciator and General Abnormal Conditions Procedures and revise them as required to ensure operator action is taken in a timely manner to limit occupational doses and environmental releases.	60 days from the date of issuance of Amendment No. 163

INSERT TO APPENDIX B (DPR-19)

Period	Requested Credit (psi)
0 – 290 sec	9.5
290 - 5,000 sec	4.8
5,000 – 30,000 sec	6.6
30,000 - 40,000 sec	6.0
40,000 - 45,500 sec	5.4
45,500 - 52,500 sec	4.9
52,500 - 60,500 sec	4.4
60,500 - 70,000 sec	3.8
70,000 - 84,000 sec	3.2
84,000 - 104,000 sec	2.5
104,000 - 136,000 sec	1.8
136,000 sec – accident end	1.1

1.1 Definitions (continued)

LINEAR HEAT GENERATION RATE (LHGR)	The LHGR shall be the heat generation rate per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.
LOGIC SYSTEM FUNCTIONAL TEST	A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all required logic components (i.e., all required relays and contacts, trip units, solid state logic elements, etc.) of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.
~~MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD)~~	~~The MFLPD shall be the largest value of the fraction of limiting power density (FLPD) in the core. The FLPD shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle type.~~
MINIMUM CRITICAL POWER RATIO (MCPR)	The MCPR shall be the smallest critical power ratio (CPR) that exists in the core for each class of fuel. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.
MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE — OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that

(continued)

Table 3.3.6.1-1 (page 1 of 3)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level − Low Low	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ ~~56.77~~ -56.34 inches
b. Main Steam Line Pressure − Low	1	2	E	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.7	≥ 831 psig
c. Main Steam Line Pressure − Timer	1	2	E	SR 3.3.6.1.2 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 0.280 seconds (Unit 2) ≤ 0.236 seconds (Unit 3)
d. Main Steam Line Flow − High	1,2,3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.7	≤ 160.5 psid (Unit 2) ≤ 117.1 psid (Unit 3)
e. Main Steam Line Tunnel Temperature − High	1,2,3	2 per trip string	D	SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 200°F
2. Primary Containment Isolation					
a. Reactor Vessel Water Level − Low	1,2,3	2	G	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ 10.24 inches
b. Drywell Pressure − High	1,2,3	2	G	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.7	≤ 1.94 psig
c. Drywell Radiation − High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 77 R/hr

(continued)

Table 3.3.6.2-1 (page 1 of 1)
Secondary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level – Low	1,2,3, (a)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.5 SR 3.3.6.2.6	≥ ~~10.24~~ 2.65 inches
2. Drywell Pressure – High	1,2,3	2	SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.6	≤ 1.94 psig
3. Reactor Building Exhaust Radiation – High	1,2,3, (a),(b)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.6	≤ 14.9 mR/hr
4. Refueling Floor Radiation – High	1,2,3, (a),(b)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.6	≤ 100 mR/hr

(a) During operations with a potential for draining the reactor vessel.

(b) During CORE ALTERATIONS and during movement of irradiated fuel assemblies in secondary containment.

ENCLOSURE 1 - ATTACHMENT C
Supplement to Request For Power Uprate Operation
Dresden Nuclear Power Station, Units 2 and 3

INFORMATION SUPPORTING A FINDING OF NO SIGNIFICANT HAZARDS CONSIDERATION

According to 10 CFR 50.92(c), "Issuance of Amendment," a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

> Involve a significant increase in the probability or consequences of an accident previously evaluated; or
>
> Create the possibility of a new or different kind of accident from any accident previously evaluated; or
>
> Involve a significant reduction in a margin of safety.

In support of this determination, an evaluation of each of the three criteria set forth in 10 CFR 50.92 is provided below regarding the proposed license amendment.

Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change revises the allowable value for the main steam line isolation function that occurs on reactor vessel water level – low low. The allowable value was recalculated using a wider and thus more conservative temperature range for the low low reactor vessel water level instrumentation. This primary containment isolation function is not involved in the initiation of accidents or transients previously evaluated. The proposed change does not result in any hardware changes. Existing operating margin between plant conditions and actual plant setpoints is not significantly reduced due to this change. As a result, the proposed change will not result in unnecessary plant transients or significantly increase the probability of an accident previously evaluated.

The purpose of the main steam line isolation function is to mitigate and thereby limit the consequences of accidents. The allowable value has been developed to ensure that the design and safety analysis limits will be satisfied. The methodology used for the development of the allowable value ensures the affected instrumentation remains capable of mitigating design basis events as described in the safety analyses and that the results and consequences described in the safety analyses remain bounding. Additionally, the proposed change does not alter the plant's ability to detect and mitigate events.

Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Does the proposed change create the possibility of a new of different kind of accident from any accident previously evaluated?

The proposed change is the result of application of the instrumentation setpoint methodology specific to the analysis of instrument channel setpoint error and instrument loop accuracy. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. This is based on the fact that the method and manner of plant operation is unchanged. The use of the proposed allowable value does not impact safe operation of the plant because the safety analysis limits will be maintained. The proposed allowable value involves no system additions or physical modifications to plant systems. This allowable value was developed using a methodology to ensure the affected instrumentation remains capable of mitigating accidents and transients. Plant equipment will not be operated in a manner different from previous operation. Since operational methods remain unchanged and the operating parameters have been evaluated to maintain the station within existing design basis criteria, no different type of failure or accident is created.

Therefore the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Does the proposed change involve a significant reduction in a margin of safety?

The proposed change has been developed using a methodology to ensure safety analysis limits are not exceeded. As such, this proposed change does not involve a significant reduction in a margin of safety.

Conclusion

Therefore, the proposed change involves no significant hazards consideration.

INFORMATION SUPPORTING AN ENVIRONMENTAL ASSESSMENT

Exelon Generation Company (EGC), LLC has evaluated this proposed change against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21, "Criteria for and identification of licensing and regulatory actions requiring environmental assessments." EGC has determined that this proposed change meets the criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9), "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," and as such, has determined that no irreversible consequences exist in accordance with 10 CFR 50.92(b), "Issuance of amendment." This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," which changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, "Standards for Protection Against Radiation," or that changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria.

(i) The amendment involves no significant hazards consideration.

As demonstrated in Attachment C, the proposed change does not involve any significant hazards considerations.

(ii) There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

The proposed change revises the allowable value for the main steam line isolation function that occurs on reactor vessel water level – low low. The change does not allow for an increase in the unit power level, does not increase the production, nor alter the flow path or method of disposal of radioactive waste or byproducts. Therefore, the proposed change does not affect actual unit effluents.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed change will not result in changes in the operation or configuration of the facility. There will be no change in the level of controls or methodology used for processing radioactive effluents or handling of solid radioactive waste. The proposed change will not result in any change in the normal radiation levels within the plant. Therefore, there will be no increase in individual or cumulative occupational radiation exposure resulting from this change.

ENCLOSURE 2 - ATTACHMENT A

Supplement to Request For Power Uprate Operation
Quad Cities Nuclear Power Station, Units 1 and 2

DESCRIPTION AND SUMMARY SAFETY ANALYSIS FOR PROPOSED CHANGES

A. SUMMARY OF PROPOSED CHANGES

Pursuant to 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company (EGC), LLC, formerly Commonwealth Edison (ComEd) Company, is requesting additional changes to the Technical Specifications (TS) relative to the changes proposed in References I.1 and I.2 for the Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2. This proposed change identifies two additional TS tables that require changes to support the proposed change to the reactor vessel water level scram and isolation setpoint for QCNPS submitted in Reference I.1.

In Reference I.1, ComEd submitted a TS amendment request for QCNPS to allow operation with an extended power uprate (EPU). The amendment request proposed various TS changes, which included a change to the allowable value for the reactor vessel water level - low scram and isolation functions. The proposed change identified the allowable value change in TS Tables 3.3.1.1-1, "Reactor Protection System Instrumentation," and 3.3.6.1-1, "Primary Containment Isolation Instrumentation." During implementation reviews for the EPU, it was recognized that the same allowable value change was required in Table 3.3.6.2-1, "Secondary Containment Isolation Instrumentation," and Table 3.3.7.1-1, "Control Room Emergency Ventilation (CREV) System Isolation Instrumentation." The changes proposed in this attachment revise this allowable value in TS Tables 3.3.6.2-1 and 3.3.7.1-1.

Reference I.1 also proposed a change to the allowable value for the main steam line flow high isolation function contained in TS Table 3.3.6.1-1, "Primary Containment Isolation Instrumentation." During implementation reviews for the EPU, it was recognized that the same allowable value change should have been proposed for Table 3.3.7.1-1, "Control Room Emergency Ventilation (CREV) System Isolation Instrumentation." The changes proposed in this attachment revise this allowable value in TS Table 3.3.7.1-1.

B. DESCRIPTION OF THE CURRENT REQUIREMENTS

B.1 Reactor Vessel Water Level – Low

Table 3.3.6.2-1, Function 1, identifies the allowable value for the reactor vessel water level – low function. The allowable value is $\geq$ 11.8 inches.

Table 3.3.7.1-1, Function 1, identifies the allowable value for the reactor vessel water level - low function. The allowable value is $\geq$ 11.8 inches.

B.2 Main Steam Line Flow – High

Table 3.3.7.1-1, Function 3, identifies the allowable value for the main steam line flow – high function. The allowable value is $\leq$ 138% rated steam flow.

C. BASES FOR THE CURRENT REQUIREMENTS

C.1 Reactor Vessel Water Level - Low

A low reactor pressure vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should the RPV water level decrease too far, fuel damage could result. An isolation of the secondary containment and actuation of the standby gas treatment system are initiated in order to minimize the potential of an offsite release. An isolation of the CREV system occurs since this could be a precursor to a potential radiation release and subsequent radiation exposure to control room personnel.

C.2 Main Steam Line Flow – High

High main steam line flow could indicate a break of a main steam line and therefore automatically initiates an isolation of the CREV system, since this could be a precursor to a potential radiation release and subsequent radiation exposure to control room personnel.

D. NEED FOR REVISION OF THE REQUIREMENTS

D.1 Reactor Vessel Water Level - Low

The loss of feedwater transient was reanalyzed under EPU conditions. Due to increased core heat generation as a result of EPU, the RPV water level decreases more rapidly in this transient. Therefore, the Reference I.1 amendment request proposed to lower the reactor vessel low water level scram setpoint in order to increase the potential for recovery before reaching the scram setpoint and thus prevent unnecessary challenges to safety systems and provide additional time for operator action.

The proposed changes to the allowable values for the secondary containment isolation and CREV system isolation functions on reactor vessel water level – low are directly related to the proposed change for the reactor scram setpoint reduction. To maintain the secondary containment isolation and CREV system isolation functions at the same level as the reactor scram function, the allowable values for TS Table 3.3.6.2-1, Function 1, and Table 3.3.7.1-1, Function 1, must be revised.

D.2 Main Steam Line Flow – High

The proposed change to the allowable value for the CREV system isolation on main steam line flow – high is directly related to the proposed change in Reference I.1 for the primary containment isolation function on main steam line flow – high in TS Table 3.3.6.1-1, Function 1.d. To maintain the CREV isolation function at the same level, the allowable value for TS Table 3.3.7.1-1, Function 3, must also be revised.

E. DESCRIPTION OF THE PROPOSED CHANGES

E.1 Reactor Vessel Water Level - Low

The allowable value for Table 3.3.6.2-1, Function 1, is revised from ≥ 11.8 inches to ≥ 3.8 inches.

The allowable value for Table 3.3.7.1-1, Function 1, is revised from ≥ 11.8 inches to ≥ 3.8 inches.

E.2 Main Steam Line Flow – High

The allowable value for Table 3.3.7.1-1, Function 3, is revised from ≤ 138% rated steam flow to ≤ 254.3 pounds per square inch differential (psid).

F. SUMMARY SAFETY ANALYSIS OF THE PROPOSED CHANGES

F.1 Reactor Vessel Water Level - Low

The reactor vessel water level - low function is assumed in the analysis of the recirculation line break and is credited in the loss of normal feedwater flow event. The reactor scram associated with the function reduces the amount of energy required to be absorbed and, along with the actions of the emergency core cooling systems, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46, "Acceptable criteria for emergency core cooling systems for light-water nuclear power reactors." The associated secondary containment isolation function is initiated in order to minimize the potential of an offsite release. Additionally, the CREV system isolation is initiated in order to minimize the potential dose to the control room operators. The allowable values for the secondary containment isolation function and CREV system isolation function are chosen to be the same as the allowable value for the reactor protection system setpoint and are not analyzed separately. The proposed change in the reactor scram setpoint does not result in a change to the current safety analyses. Thus, the change in the allowable value for the secondary containment isolation function continues to ensure that any offsite releases are within the limits calculated in the safety analysis. For the CREV system isolation function, the change in allowable value continues to ensure that the radiation exposure of control room personnel, as a result of a LOCA, does not exceed the limits set by GDC 19 "Control Room," of 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

F.2 Main Steam Line Flow – High

This proposed change is associated with the units of measurement for the allowable value. The proposed change revises the allowable value from units of percent rated steam flow to units of psid. This proposed change preserves the same allowable value in terms of percent rated steam flow (i.e., 254.3 psid is equivalent to 138% of uprated steam flow). Because of the increase in rated steam flow associated with the EPU, the proposed change increases the actual mass flow rate of steam required to actuate the isolation function. Since the maximum steam flow following a main steam line (MSL) break does not change due to the flow restrictors, the proposed changes result in a decrease in the difference between the allowable value and the maximum flow. The purpose of the main steam line flow - High isolation function is to provide protection against pipe breaks in the MSL outside the drywell. For a complete severance of one MSL, steam flow increases almost instantaneously to the maximum steam flow as limited by the flow restrictors. Thus, the present and proposed setpoints would be attained virtually at the same time. Therefore, the consequences of a MSL break as evaluated in the UFSAR will remain unchanged with the increase in high flow setpoint.

G. IMPACT ON PREVIOUS SUBMITTALS

All submittals currently under review by the NRC were evaluated to determine the impact of these proposed changes. These proposed changes supplement the changes proposed to support uprated power operation at QCNPS in References I.1 and I.2.

In addition, these proposed changes affect the proposed changes submitted in Reference I.3, which requested that the NRC consider the proposed changes to the reactor water level – low setpoint separately from the EPU amendment request. The additional proposed change being submitted in this amendment request is also being submitted to the NRC separately as a supplement to the Reference I.3 amendment request.

No other submittals currently under review by the NRC are affected by the information presented in this supplemental license amendment request.

H. SCHEDULE REQUIREMENTS

We request that these proposed changes be reviewed and approved as part of the proposed changes for power uprate operation previously submitted in References I.1 and I.2.

I. REFERENCES

1. Letter from R. M. Krich (ComEd) to U. S. NRC, "Request for License Amendment for Power Uprate Operation," dated December 27, 2000

2. Letter from R. M. Krich (EGC) to U. S. NRC, "Supplement to Request for License Amendment for Power Uprate Operation," dated April 13, 2001

3. Letter from R. M. Krich (EGC) to U. S. NRC, "Request for License Amendment for Reactor Vessel Low Water Level Setpoint," dated February 22, 2001

ENCLOSURE 2 - ATTACHMENT B

Supplement to Request For Power Uprate Operation
Quad Cities Nuclear Power Station, Units 1 and 2

MARKED-UP TS PAGES FOR PROPOSED CHANGES

The marked-up Technical Specifications are provided in the following pages.

REVISED PAGES
3.3.6.2-4
3.3.7.1-4

Table 3.3.6.2-1 (page 1 of 1)
Secondary Containment Isolation Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1.	Reactor Vessel Water Level — Low	1,2,3, (a)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.5 SR 3.3.6.2.6	$\geq$ ~~11.8~~ 3.8 inches
2.	Drywell Pressure — High	1,2,3	2	SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.6	$\leq$ 2.43 psig
3.	Reactor Building Exhaust Radiation — High	1,2,3, (a),(b)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.6	$\leq$ 9 mR/hr
4.	Refueling Floor Radiation — High	1,2,3, (a),(b)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.6	$\leq$ 100 mR/hr

(a) During operations with a potential for draining the reactor vessel.

(b) During CORE ALTERATIONS and during movement of irradiated fuel assemblies in secondary containment.

Table 3.3.7.1-1 (page 1 of 1)
Control Room Emergency Ventilation (CREV) System Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level − Low	1,2,3, (a)	2	C	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.5 SR 3.3.7.1.6	≥ ~~11.8~~ inches 3.8
2. Drywell Pressure − High	1,2,3	2	C	SR 3.3.7.1.2 SR 3.3.7.1.4 SR 3.3.7.1.6	≤ 2.43 psig
3. Main Steam Line Flow − High	1,2,3	2 per MSL	B	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.5 SR 3.3.7.1.6	254.3 psid ≤ ~~130% rated steam flow~~
4. Refueling Floor Radiation − High	1,2,3, (a),(b)	2	B	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.4 SR 3.3.7.1.6	≤ 100 mR/hr
5. Reactor Building Ventilation Exhaust Radiation − High	1,2,3, (a),(b)	2	B	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.4 SR 3.3.7.1.6	≤ 9 mR/hr

(a) During operations with a potential for draining the reactor vessel.

(b) During CORE ALTERATIONS and during movement of irradiated fuel assemblies in the secondary containment.