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Exelon Nuclear

10 CFR 50.90

RS-01-083

April 13, 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

Reference: (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 (Commonwealth Edison Company) to U.S. NRC, "Request for Technical Specifications Changes for Dresden Nuclear Power Station, Units 2 and 3, LaSalle County Station, Units 1 and 2, and Quad Cities Nuclear Power Station, Units 1 and 2, to Implement Improved Standard Technical Specifications," dated March 3, 2000

(3) Letter from R.M. Krich (Commonwealth Edison Company) to U.S. NRC, "Revision A to Request for Technical Specifications Changes for Dresden Nuclear Power Station, Units 2 and 3, LaSalle County Station, Units 1 and 2, and Quad Cities Nuclear Power Station, Units 1 and 2, to Implement Improved Standard Technical Specifications," dated June 5, 2000

(4) Letter from R. M. Krich (Commonwealth Edison Company) to U.S. NRC, "Revision B to Request for Technical Specifications Changes for Dresden Nuclear Power Station, Units 2 and 3, LaSalle County Station, Units 1 and 2, and Quad Cities Nuclear Power Station, Units 1 and 2, to Implement Improved Standard Technical Specifications," dated September 1, 2000

A001

(5) Letter from R. M. Krich (Commonwealth Edison Company) to U.S. NRC, "Revision C to Request for Technical Specifications Changes for Dresden Nuclear Power Station, Units 2 and 3, LaSalle County Station, Units 1 and 2, and Quad Cities Nuclear Power Station, Units 1 and 2, to Implement Improved Standard Technical Specifications," dated December 18, 2000

(6) Letter from R. M. Krich (Commonwealth Edison Company) to U.S. NRC, "Revision D to Request for Technical Specifications Changes for Dresden Nuclear Power Station, Units 2 and 3, LaSalle County Station, Units 1 and 2, and Quad Cities Nuclear Power Station, Units 1 and 2, to Implement Improved Standard Technical Specifications," dated February 15, 2001

(7) Letter from U.S. NRC to O.D. Kingsley, (Exelon Generation Company), "Issuance of Amendments," dated March 30, 2001

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 Reference 1 for the Dresden Nuclear Power Station (DNPS), Units 2 and 3, and the Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2. These additional proposed changes raise the allowable value for the reactor scram on fixed neutron flux – high from 120% of rated thermal power (RTP) to 122% of RTP. The proposed changes also raise the maximum allowable value for the reactor scram on flow-biased neutron flux – high from 120% RTP to 122% RTP. In addition, changes in the supporting documentation for the Reference 1 request are provided for NRC review.

In Reference 1, ComEd submitted a TS amendment request for DNPS and QCNPS to allow operation at uprated power levels. This extended power uprate (EPU) amendment request was based on the content of the proposed conversion to Improved Technical Specifications (ITS) as described in Reference 2, and the supplements provided in References 3, 4, and 5, which were current at the time the EPU amendment request was submitted. Following submittal of the EPU amendment request, the proposed content of the ITS was revised as described in Reference 6. This revision lowered the allowable value for the reactor scram on fixed neutron flux – high and flow-biased neutron flux – high from 122% RTP to 120% RTP to be consistent with the then current TS requirements. EGC has determined that, with the approval of the proposed EPU changes, these allowable values can be raised from 120% RTP to 122% RTP. The justification for these proposed changes is provided in the attachments to this letter.

In addition to these proposed changes, changes to the supporting documentation for the Reference 1 request are provided for NRC review. EGC has determined that the changes to the supporting documentation do not affect the information supporting a finding of no significant hazards consideration provided in Reference 1. These changes include the following.

- Revised copy of Reference 1, Attachment G, "Plant Modifications Required to Support Power Uprate." The revised copy provides changes in the list of modifications required. These changes resulted from evaluations that were listed as incomplete in the original EPU amendment request. EGC does not expect that any significant additional modifications will be required to support the implementation of the EPU. The changes are indicated with revision bars in the margin.

- Revised pages related to the proposed credit for containment overpressure for DNPS. The revised pages correct two errors discovered during an internal review of the proposed changes and supporting documents. The errors were a result of incorrectly transferring results from calculations to the submittal documents. The changes are indicated with revision bars in the margin.

- Revised DNPS TS Bases page B 3.3.1.1-31. The revised page corrects an editorial error in marking up the power level at which turbine valve functions can be bypassed. The change is indicated with a revision bar in the margin.

- Marked-up TS pages for the EPU amendment request based on the recently approved conversion to the ITS as provided in Reference 7. As described above, the Reference 1 EPU amendment request was submitted to the NRC based on an earlier revision of the proposed conversion to ITS.

This supplement to the Reference 1 amendment request contains separate enclosures for DNPS and QCNPS. The enclosures are subdivided as follows.

1. Attachment A contains a detailed description of the additional proposed changes to the allowable values for the reactor scram on fixed neutron flux – high and flow-biased neutron flux - high.
2. Attachment B provides the proposed markups to the TS for the proposed allowable value changes. These are provided on the marked-up pages for the EPU amendment request.
3. Attachment C provides the information supporting a finding of no significant hazards consideration for the proposed allowable value changes in accordance with 10 CFR 50.92(c), "Issuance of Amendment."
4. Attachment D provides information supporting an Environmental Assessment for the proposed allowable value changes.
5. Attachment E provides revised pages for the Reference 1 request, including the revised list of plant modifications required to support EPU for both DNPS and QCNPS, the corrected pages related to the proposed credit for containment overpressure for DNPS, and the corrected TS Bases page for DNPS.
6. Attachment F provides the marked-up TS pages for the Reference 1 amendment request based on the recently approved conversion to the ITS.

The proposed changes to the allowable values for the reactor scram on fixed neutron flux – high and flow-biased neutron flux – high, the changes to proposed credit for containment overpressure for DNPS, and the revised list of plant modifications 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 EPU operation in Reference 1.

Should you have any questions related to this request, please contact Mr. Allan R. Haeger at (630) 663 6645.

Respectfully,

R.M. Krich
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 Pages for Proposed Changes
Attachment C: Information Supporting a Finding of No Significant Hazards Consideration
Attachment D: Information Supporting an Environmental Assessment
Attachment E: Revised Pages for Power Uprate License Amendment Request
Attachment F: Marked-up Power Uprate Technical Specifications Based on Approved Version of Improved Technical Specifications

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
Attachment C: Information Supporting a Finding of No Significant Hazards Consideration
Attachment D: Information Supporting an Environmental Assessment
Attachment E: Revised Pages for Power Uprate License Amendment Request
Attachment F: Marked-up Power Uprate Technical Specifications Based on Approved Version of Improved Technical Specifications

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 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.

RM Krich

R. M. Krich
Director - Licensing
Mid-West Regional Operating Group

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

for the State above named, this 13th day of

April, 20 01

OFFICIAL SEAL
LINDA S. SANDERS
NOTARY PUBLIC, STATE OF ILLINOIS
MY COMMISSION EXPIRES 12-29-2002

Linda S. Sanders
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 the Commonwealth Edison (ComEd) Company, is requesting changes to the Technical Specifications (TS) for Dresden Nuclear Power Station (DNPS), Units 1 and 2. These proposed changes are in addition to the changes proposed to support uprated power operation at DNPS in Reference I.1. These additional proposed changes raises the allowable value for the reactor scram on fixed neutron flux – high from 120% of rated thermal power (RTP) to 122% of RTP. These proposed changes also affect the clamped (i.e., maximum) allowable value for the reactor scram on flow-biased neutron flux - high.

The additional changes proposed in this supplemental amendment request are related to the previously proposed changes for power uprate. The previously requested amendment for power uprate included changes related to the Average Power Range Monitor (APRM) and Rod Block Monitor (RBM) TS and are referred to in this supplemental request as the Partial APRM RBM TS (ARTS) changes. As described below, the previous limitation of 120% for this allowable value is no longer necessary. The allowable value proposed in this supplemental request is now based on the analytical limit for the neutron flux scram assumed in the transient and accident analyses for DNPS.

B. DESCRIPTION OF THE CURRENT REQUIREMENTS

TS Section 3.3.1.1, "Reactor Protection System Instrumentation"

TS Table 3.3.1.1-1 Function 2.b

Table 3.3.1.1-1, "Reactor Protection System Instrumentation," function 2.b identifies the allowable value for the APRM Flow Biased Neutron Flux - High function. For two-loop operation, the allowable value is ≤ 0.58 W + 63.5% RTP and $\leq$ 120% RTP.

TS Table 3.3.1.1-1 Function 2.c

Table 3.3.1.1-1, Function 2.c identifies the allowable value for the APRM Fixed Neutron Flux – High Function. The allowable value is $\leq$ 120% RTP.

C. BASES FOR THE CURRENT REQUIREMENTS

The APRM neutron flux trip level is varied as a function of reactor recirculation drive flow. At lower core flows, the setpoint is reduced proportional to the reduction in power experienced as core flow is reduced. This value is clamped at an upper limit that is equivalent to the APRM Neutron Flux – High function allowable value. The APRM Flow Biased Neutron Flux - High function provides protection against transients where

thermal power increases slowly and protects the fuel cladding integrity by ensuring that the Minimum Critical Power Ratio (MCPR) Safety Limit (SL) is not exceeded. During any transient event that occurs at a reduced reactor recirculation flow, because of a lower scram trip setpoint, the APRM Flow Biased Neutron – High function will initiate a scram before the clamped allowable value is reached.

The APRM gain or APRM Flow Biased Neutron Flux-High function allowable value is required to be adjusted when operating under conditions of excessive power peaking to maintain acceptable margins to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit. The condition of excessive power peaking protection of the fuel cladding integrity by ensuring that the MCPR SL is not exceeded is determined by the Fuel Design Limiting Ratio for Centerline Melt (FDLRC), which is defined as follows.

$$\text{FDLRC} = \frac{(\text{LHGR})\,(1.2)}{(\text{TLHGR})\,(\text{FRTP})}$$

where

LHGR	= Linear Heat Generation Rate
FRTP	= Fraction of Rated Thermal Power
TLHGR	= Transient Linear Heat Generation Rate

Maintaining FDLRC less than or equal to 1.0 ensures the fuel does not experience centerline melt during abnormal operational occurrences (AOOs) beginning at any power level and terminating at 120% RTP (i.e., the APRM Fixed Neutron Flux - High allowable value).

The APRM Fixed Neutron Flux – High function generates a trip signal to prevent fuel damage or excessive Reactor Coolant System (RCS) pressure. For the overpressurization protection analysis, the APRM Fixed Neutron Flux – High function is assumed to terminate the main steam isolation valve (MSIV) closure event and, along with the safety valves, limits the peak reactor pressure vessel (RPV) pressure to less than the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code limits. The control rod drop accident (CRDA) analysis takes credit for the APRM Fixed Neutron Flux – High function to mitigate the consequences of the CRDA.

D. NEED FOR REVISION OF THE REQUIREMENTS

Because the analyses performed for the power uprate proposed changes are based on the Partial ARTS operating strategy, use of FDLRC as a fuel thermal limit is no longer required. With the Partial ARTS operating strategy, monitoring to ensure that the MCPR SL is not exceeded is performed with power and flow dependent limits. These power and flow dependent limits are fully described in Reference I.1. Use of the Partial ARTS operating strategy removes the need to limit the allowable value for the neutron flux scram to the 120% value dictated by the FDLRC equation, and allows use of an allowable value that results directly from the safety analyses.

Raising the allowable value for the neutron flux scram provides operational flexibility and reduces the potential for unnecessary actuations of safety systems.

E. DESCRIPTION OF THE PROPOSED CHANGES

TS Table 3.3.1.1-1 Function 2.b

The clamped portion of the allowable value for the APRM Flow-Biased Neutron Flux – High is changed from ≤ 120% RTP to ≤ 122% RTP. This is in addition to the changes to the flow-biased portion that were discussed in Reference I.1.

TS Table 3.3.1.1-1 Function 2.c

The allowable value is changed from ≤ 120% RTP to ≤ 122% RTP.

F. SUMMARY SAFETY ANALYSIS OF THE PROPOSED CHANGES

The analyses that were performed for the power uprate proposed changes related to the transient events listed in the bases above assume an analytical limit of 125% RTP for the APRM Fixed Neutron Flux - High function and the clamped value of the the APRM Flow-Biased Neutron Flux – High function. The APRM setpoint calculations determined that, based on this analytical limit, an allowable value of 122% is appropriate and ensures the analytical limit is maintained. An allowable value of 120% had been previously used because this value is conservative and consistent with the FDLRC equation. Because the analyses performed for the power uprate proposed changes are based on the Partial ARTS operating strategy, use of FDLRC is no longer required. With the Partial ARTS operating strategy, monitoring to ensure that the MCPR SL is not exceeded is performed with power and flow dependent limits. These power and flow dependent limits are fully described in Reference I.1. Use of the Partial ARTS operating strategy removes the need to limit the allowable value for the neutron flux scram to the 120% value dictated by the FDLRC equation, and allows use of an allowable value that results directly from the safety analyses.

The setpoint calculations for the proposed change were performed in accordance with the the General Electric setpoint methodology specified in Reference I.2.

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.

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

H. SCHEDULE REQUIREMENTS

EGC requests that these proposed changes be reviewed and approved as part of the requested changes for power uprate operation in Reference I.1, which were requested to be approved by October 15, 2001, to support operation at uprated power following the DNPS refueling outage scheduled to begin on October 20, 2001.

I. REFERENCES

1. Letter from R. M. Krich (ComEd) to U.S. NRC, "Request for Technical Specifications Changes for Dresden Nuclear Power Station, Units 2 and 3 and Quad Cities Nuclear Power Station, Units 1 and 2, to Allow Operation at Uprated Power Levels," dated December 27, 2000

2. General Electric Nuclear Energy, "General Electric Instrument Setpoint Methodology," NEDC-31336P-A, Class III (Proprietary), September 1996 (Note: will add citation of approval letter before sending)

ENCLOSURE 1 - ATTACHMENT B

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

MARKED-UP TS PAGES FOR PROPOSED CHANGES

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

REVISED PAGES

3.3.1.1-8
3.3.1.1-9

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux – High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 121/125 divisions of full scale
	5(a)	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 121/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.18	NA
	5(a)	3	H	SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.18	NA
2. Average Power Range Monitors					
a. Neutron Flux – High, Setdown	2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.7 SR 3.3.1.1.9 SR 3.3.1.1.15 SR 3.3.1.1.18	≤ 17.1% RTP
b. Flow Biased Neutron Flux – High	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18 SR 3.3.1.1.19	0.56 W + 67.4% ≤ ~~0.58 W + 63.5%~~ RTP and ≤ 120% RTP(b) 122%

(continued)

Revised

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) 0.58 W + 59.2% and ≤ 118.5% RTP when reset for single loop operation per LCO 3.4.1. "Recirculation Loops Operating."
0.56W + 63.2%

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
c. Fixed Neutron Flux - High	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.15 SR 3.3.1.1.18 SR 3.3.1.1.19	≤ 120% RTP 122%
d. Inop	1,2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.18	NA
3. Reactor Vessel Steam Dome Pressure - High	1,2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.18 SR 3.3.1.1.19	≤ 1058 psig
4. Reactor Vessel Water Level - Low	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.12 SR 3.3.1.1.17 SR 3.3.1.1.18 SR 3.3.1.1.19	≥ 10.24 inches 2.65
5. Main Steam Isolation Valve - Closure	1, 2[(c)]	8	F	SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.17 SR 3.3.1.1.18 SR 3.3.1.1.19	≤ 9.5% closed
6. Drywell Pressure - High	1,2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.13 SR 3.3.1.1.18 SR 3.3.1.1.19	≤ 1.94 psig

| Revised

(continued)

(c) With reactor pressure ≥ 600 psig.

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.

Exelon Generation Company (EGC), LLC is requesting changes to the Technical Specifications (TS), for Dresden Nuclear Power Station (DNPS), Units 2 and 3. The proposed changes raise the allowable value for the reactor scram on fixed neutron flux – high from 120% of rated thermal power (RTP) to 122% RTP. The proposed changes also raise the maximum allowable value for the reactor scram on flow-biased neutron flux – high from 120% RTP to 122% RTP. The proposed changes are associated with previously-proposed changes to revise the maximum power level specified in each unit's license, and TS definition of rated thermal power.

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

The proposed changes modifiy the allowable value for an instrument setpoint for the neutron flux at which a reactor scram initiates to mitigate an accident or transient. The changes do not modify the analytical limit (i.e., the value assumed in the safety analyses) for this function. Because these changes do not affect the way in which plant equipment functions or create any new failure mechanisms and therefore do not affect the initiation of any accidents previously evaluated. Therefore the proposed changes do not affect the probability of an accident previously evaluated.

The proposed changes do not affect the analytical limit at which the reactor scram is assumed to actuate in the transient and accident analyses for DNPS. The calculation for the revised allowable value maintains the same analytical limit, and therefore the same assumed actuation level for the reactor scram function. Therefore, the proposed changes do not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed changes modify the allowable value for an instrument setpoint for the neutron flux at which a reactor scram initiates to mitigate an accident or transient. The changes do not modify the analytical limit (i.e., the value assumed in the safety analyses) for this function. These changes do not affect the way in which plant equipment functions or create any new failure mechanisms. Therefore the proposed changes do 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 changes do not affect the analytical limit at which the reactor scram is assumed to actuate in the transient and accident analyses for DNPS. The calculation for the revised allowable value maintains the same analytical limit as currently assumed in the DNPS TS and in the analyses performed for the power uprate license amendment request. Therefore the assumed actuation level for the reactor scram function remains unchaged. Thus, the assumptions and results in the transient and accident analyses are unchanged. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

Conclusion

The proposed changes do not involve a significant hazards consideration.

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

INFORMATION SUPPORTING AN ENVIRONMENTAL ASSESSMENT

Exelon Generation Company (EGC), LLC has evaluated these proposed changes 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 these proposed changes meet 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 proposed changes meet the following specific criteria.

(i) The proposed changes involve no significant hazards consideration.

As demonstrated in Attachment C, the proposed changes do 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 changes raise the allowable value for the reactor scram on fixed neutron flux –high from 120% of rated thermal power (RTP) to 122% of RTP. The proposed changes also raise the maximum allowable value for the reactor scram on flow-biased neutron flux – high from 120% RTP to 122% RTP. The changes do not allow for an increase in the unit power level, do not increase the production, nor alter the flow path or method of disposal of radioactive waste or byproducts. Therefore, the proposed changes do not affect actual unit effluents.

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

The proposed changes 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 changes 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 these changes.

ENCLOSURE 1 – ATTACHMENT E
Supplement to Request for Power Uprate Operation
Dresden Nuclear Power Station, Units 2 and 3

REVISED PAGES FOR POWER UPRATE LICENSE AMENDMENT REQUEST

The attached pages replace the corresponding pages in the letter from R.M. Krich (Commonwealth Edison Company) to U.S. NRC, "Request for License Amendment for Power Uprate Operation," dated December 27, 2000

REVISED PAGE FROM DESCRIPTION AND SUMMARY SAFETY ANALYSES FOR PROPOSED CHANGES (Attachment A of License Amendment Request)

Page 16 of 32

REVISED LICENSE PAGES (Attachment B of License Amendment Request

Appendix B (Unit 2)
Appendix B (Unit 3)

REVISED BASES PAGE (Attachment B of License Amendment Request)

Page B 3.3.1.1-33

REVISED SAFETY ANALYSIS REPORT PAGE (Attachment E of License Amendment Request)

Page 4-14

REVISED LIST OF PLANT MODIFICATIONS (Attachment G of License Amendment Request)

Page 1 of 2
Page 2 of 2

E.2. Operating License Condition on Containment Overpressure

The allowance for containment overpressure in the license conditions is revised to state, "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)
0-290	9.5
290-5,000	4.8
5,000-30,000	~~4.25~~ 5.2

E.3. TS Definition of Rated Thermal Power

Section 1.1, "Definitions," RTP is revised to reflect the increase from 2527 MWt to 2957 MWt.

E.4. TS Definition of Fuel Design Limiting Ratio for Centerline Melt

The definition of FDLRC in Section 1.1, "Definitions," is deleted.

E.5. TS Section 3.2.4, "Average Power Range Monitor Gain and Setpoint"

TS Section 3.2.4 is deleted.

E.6. TS Section 3.3.1.1, "Reactor Protection System Instrumentation"

TS SR 3.3.1.1.2

The reference to TS Section 3.2.4 is removed so that SR 3.3.1.1.2 states, "Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is ≤ 2% RTP."

TS SR 3.3.1.1.14

The thermal power applicability is changed from ≥ 45% to ≥ 38.5% so that SR 3.3.1.1.14 states, "Verify Turbine Stop Valve – Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure – Low Functions are not bypassed when THERMAL POWER is ≥ 38.5%."

APPENDIX B

ADDITIONAL CONDITIONS

DPR
FACILITY OPERATING LICENSE NO. DRP-19

Commonwealth Edison Company 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)	
0-290	0-~~240~~	9.5	9.5
290-5000	~~240~~-480	~~2.9~~	4.8
5000-30,000	480-~~6000~~	~~1.9~~	~~4.25~~ 5.2
	~~6000~~-accident end	2.5	

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

APPENDIX B

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. DRP-25

Commonwealth Edison Company shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Condition	Implementation Date
152	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.	Prior to Unit 3 returning to Mode 3 from refueling outage D3R14.

	Time (seconds)	Containment Pressure (PSIG)	
0-290	~~0-240~~	~~9.5~~	9.5
290-5000	~~240-480~~	~~2.9~~	4.8
5000-30,000	~~480-6000~~	~~1.9~~	~~4.25~~ 5.2
	~~6000-accident end~~	~~2.5~~	

Amendment Number	Additional Condition	Implementation Date
152	The licensee shall complete the evaluation of the torus attached piping.	Prior to Unit 3 returning to Mode 3 from refueling outage D3R14.
152	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. 152.
155	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. 155

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.11 and SR 3.3.1.1.16 (continued)

Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The 92 day Frequency of SR 3.3.1.1.11 is based on the reliability analysis of Reference 13. The 24 month Frequency of SR 3.3.1.1.16 is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency.

SR 3.3.1.1.12

Calibration of trip units provides a check of the actual trip setpoints. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value specified in Table 3.3.1.1-1. If the trip setting is discovered to be less conservative than accounted for in the appropriate setpoint methodology, but is not beyond the Allowable Value, the channel performance is still within the requirements of the plant safety analysis. Under these conditions, the setpoint must be readjusted to be equal to or more conservative than accounted for in the appropriate setpoint methodology.

The Frequency of 92 days is based on the reliability analysis of Reference 13.

SR 3.3.1.1.14

This SR ensures that scrams initiated from the Turbine Stop Valve—Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Functions will not be inadvertently bypassed when THERMAL POWER is ≥ ~~45%~~ RTP. This involves calibration of the bypass channels. Adequate margins for

~~33.5%~~ 38.5%

(continued)

Table 4-2

NPSH Overpressure Credit

Current

Time (seconds)	**Credited Pressure** (psig)	**Minimum Available Pressure** (psig)	**Difference between Credited and Available Pressure** (psi)
0 – 240	9.5	10.2	0.7
240 – 480	2.9	3.4	0.5
480 – 6000	1.9	2.3	0.4
6000 – accident end	2.5	2.6	0.1

EPU

Time (seconds)	**Credited Pressure** (psig)	**Minimum Available Pressure** (psig)	**Difference between Credited and Available Pressure** (psi)
0 – 290	9.5	10.5	1.0
290 – 5,000	4.8	5.4	0.6
5,000 – 30,000 *	~~5.3~~ 5,2	6.1	0.8

* From 30,000 seconds to the end of the accident, the available pressure and required pressure decrease in parallel fashion. Minimum margin between available pressure and required pressure during this period is 2.4 psi.

ATTACHMENT G - REVISED

Proposed Changes to Operating Licenses and Technical Specifications for Dresden Nuclear Power Station, Units 2 and 3

PLANT MODIFICATIONS REQUIRED TO SUPPORT POWER UPRATE

The following presents an overview of the facility changes necessary to achieve the target electrical power output of 912 MWe.

- An additional 125 VDC cable will be added to the safety-related DC system to provide the additional capacity anticipated at uprated power conditions.
- Various instruments will require scaling/setpoint changes.
- A modification to provide tripping of the 4th condensate pump on a LOCA will be implemented to allow the continued use of the feedwater pumps.
- A fault current limiting arrangement will be implemented to maintain non-safety bus short circuit ratings after a postulated loss of an auxiliary transformer in conjunction with a short circuit.
- A reactor recirculation pump runback on a loss of feedwater flow or the loss of a condensate pump will be implemented to reduce the potential for a scram on reactor low water level and allow continued operation.
- An additional steam line steam resonance compensator card designed to attenuate third order harmonics will be installed in the electro-hydraulic control system to reduce electrical noise in the system.
- A new high-pressure turbine rotor will be installed as a result of the increased steam flow associated with operation at uprated power conditions.
- Turbine cross around relief valve alterations will be performed to ensure that pressure limitations are not exceeded.
- Selected heater drain valve normal drain trim replacements will be performed due to the increase in drain flow.
- Some feedwater heater relief valves will be adjusted or replaced and the heaters will be rerated to compensate for the increased feedwater flow and the associated pressure change.
- Condenser tube staking is planned for the main condensers to provide adequate protection against tube vibration damage at uprated power conditions.
- An additional condensate prefilter will be installed to process the increased flow.
- Additional cooling towers will be installed to ensure that the temperature of the water released to the environment remains within existing limits.

- Various support and piping modifications will be performed due to the increased temperature in torus-attached piping and increased temperature and flow in the main steam and feedwater systems.
- Restriction orifices to the stator water cooling system will be resized to accommodate the increased heat load.
- Modifications to the steam dryer will be performed to reduce moisture carryover.
- Modifications will be performed to increase the cooling capacity of the iso-phase bus duct cooling system.
- Clamps will be added to selected reactor vessel jet pump sensing lines to reduce vibration

ENCLOSURE 1 - ATTACHMENT F

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

MARKED-UP POWER UPRATE TECHNICAL SPECIFICATION PAGES BASED ON APPROVED VERSION OF IMPROVED TECHNICAL SPECIFICATIONS

REVISED PAGES

1.1-3
1.1-4
3.2.4-1
3.2.4-2
3.3.1.1-2
3.3.1.1-4
3.3.1.1-6
3.3.1.1-8
3.3.1.1-9
3.3.1.1-10
3.3.5.2-2
3.3.6.1-5
3.3.6.1-7
3.4.3-2
5.5-11
5.6-3

1.1 Definitions

DOSE EQUIVALENT I-131 (continued)	conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites;" Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977; or ICRP 30, Supplement to Part 1, pages 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."
FUEL DESIGN LIMITING RATIO FOR CENTERLINE MELT (FDLRC)	The FDLRC shall be 1.2 times the LHGR existing at a given location divided by the product of the transient LHGR limit and the fraction of RTP.
LEAKAGE	LEAKAGE shall be: a. <u>Identified LEAKAGE</u> 1. LEAKAGE into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or 2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; b. <u>Unidentified LEAKAGE</u> All LEAKAGE into the drywell that is not identified LEAKAGE; c. <u>Total LEAKAGE</u> Sum of the identified and unidentified LEAKAGE; and d. <u>Pressure Boundary LEAKAGE</u> LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall.

(continued)

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 logic components required for OPERABILITY 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.
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 are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2527 MWt. 2957

(continued)

3.2 POWER DISTRIBUTION LIMITS

3.2.4 Average Power Range Monitor (APRM) Gain and Setpoint

LCO 3.2.4

a. FDLRC shall be less than or equal to 1.0; or

b. Each required APRM Flow Biased Neutron Flux - High Function Allowable Value shall be modified by 1/FDLRC; or

c. Each required APRM gain shall be adjusted such that the APRM readings are ≥ 100% times the Fraction of RTP (FRTP) times FDLRC.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Requirements of the LCO not met.	A.1	Satisfy the requirements of the LCO.	6 hours
B. Required Action and associated Completion Time not met.	B.1	Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.2.4.1	-------------------NOTE------------------- Not required to be met if SR 3.2.4.2 is satisfied for LCO 3.2.4.b or LCO 3.2.4.c requirements. -- Verify FDLRC is within limits.	Once within 12 hours after ≥ 25% RTP AND 24 hours thereafter
SR 3.2.4.2	-------------------NOTE------------------- Not required to be met if SR 3.2.4.1 is satisfied for LCO 3.2.4.a requirements. -- Verify each required: a. APRM Flow Biased Neutron Flux — High Function Allowable Value is modified by 1/FDLRC; or b. APRM gain is adjusted such that the APRM reading is ≥ 100% times the FRTP times FDLRC.	12 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more Functions with RPS trip capability not maintained.	C.1 Restore RPS trip capability.	1 hour
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1 Reduce THERMAL POWER to < ~~45%~~ RTP. 38.5%	4 hours
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1 Be in MODE 2.	8 hours
	<u>AND</u>	
	F.2 ---------NOTE-------- Only required to be met for Function 5, Main Steam Isolation Valve — Closure, and Function 10, Turbine Condenser Vacuum — Low. --------------------- Reduce reactor pressure to < 600 psig.	8 hours

(continued)

SURVEILLANCE REQUIREMENTS

------------------------------------NOTES------------------------------------

1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.

2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.1.2	-----------------NOTE----------------- Not required to be performed until 12 hours after THERMAL POWER ≥ 25% RTP. -------------------------------------- Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is ≤ 2% RTP. ~~plus any gain adjustment required by LCO 3.2.4, "Average Power Range Monitor (APRM) Gain and Setpoint" while operating at ≥ 25% RTP.~~	7 days
SR 3.3.1.1.3	Adjust the channel to conform to a calibrated flow signal.	7 days

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.11	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.1.1.12	Calibrate the trip units.	92 days
SR 3.3.1.1.13	Perform CHANNEL CALIBRATION.	92 days
SR 3.3.1.1.14	Verify Turbine Stop Valve — Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure — Low Functions are not bypassed when THERMAL POWER is ≥ ~~45%~~ 38.5% RTP.	92 days
SR 3.3.1.1.15	------------------NOTES------------------ 1. Neutron detectors are excluded. 2. For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 24 hours after entering MODE 2. 3. For Function 2.b, not required for the flow portion of the channels. --- Perform CHANNEL CALIBRATION.	184 days
SR 3.3.1.1.16	Perform CHANNEL FUNCTIONAL TEST.	24 months

(continued)

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux – High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 121/125 divisions of full scale
	5(a)	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 121/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.18	NA
	5(a)	3	H	SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.18	NA
2. Average Power Range Monitors					
a. Neutron Flux – High, Setdown	2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.7 SR 3.3.1.1.9 SR 3.3.1.1.15 SR 3.3.1.1.18	≤ 17.1% RTP
b. Flow Biased Neutron Flux – High	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18 SR 3.3.1.1.19	0.56 W + 67.4% ≤ ~~0.58 W + 63.5%~~ RTP and ≤ ~~120%~~ RTP(b) 122%

| Revised

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) ~~0.58 W + 59.2%~~ and ≤ 118.5% RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

0.56W + 63.2%

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
c. Fixed Neutron Flux – High	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.15 SR 3.3.1.1.18 SR 3.3.1.1.19	≤ ~~120%~~ RTP 122% Revised
d. Inop	1,2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.18	NA
3. Reactor Vessel Steam Dome Pressure – High	1,2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.18 SR 3.3.1.1.19	≤ 1058 psig
4. Reactor Vessel Water Level – Low	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.12 SR 3.3.1.1.17 SR 3.3.1.1.18 SR 3.3.1.1.19	≥ ~~10.24~~ inches 2.65
5. Main Steam Isolation Valve – Closure	1, 2[(c)]	8	F	SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.17 SR 3.3.1.1.18 SR 3.3.1.1.19	≤ 9.5% closed
6. Drywell Pressure – High	1,2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.13 SR 3.3.1.1.18 SR 3.3.1.1.19	≤ 1.94 psig

(continued)

(c) With reactor pressure ≥ 600 psig.

Table 3.3.1.1-1 (page 3 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
7. Scram Discharge Volume Water Level – High					
a. Thermal Switch (Unit 2) Float Switch (Unit 3)	1,2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 37.9 gallons (Unit 2) ≤ 39.1 gallons (Unit 3)
	5(a)	2	H	SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 37.9 gallons (Unit 2) ≤ 39.1 gallons (Unit 3)
b. Differential Pressure Switch	1,2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 37.9 gallons (Unit 2) ≤ 39.1 gallons (Unit 3)
	5(a)	2	H	SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 37.9 gallons (Unit 2) ≤ 39.1 gallons (Unit 3)
8. Turbine Stop Valve – Closure	≥ ~~45%~~ RTP 38.5%	4	E	SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.14 SR 3.3.1.1.17 SR 3.3.1.1.18 SR 3.3.1.1.19	≤ 9.5% closed
9. Turbine Control Valve Fast Closure, Trip Oil Pressure – Low	≥ ~~45%~~ RTP 38.5%	2	E	SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.14 SR 3.3.1.1.17 SR 3.3.1.1.18 SR 3.3.1.1.19	≥ 466 psig
10. Turbine Condenser Vacuum – Low	1, 2(c)	2	F	SR 3.3.1.1.5 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.18 SR 3.3.1.1.19	21.4 ≥ ~~21.15~~ inches Hg vacuum
11. Reactor Mode Switch – Shutdown Position	1,2	1	G	SR 3.3.1.1.16 SR 3.3.1.1.18	NA
	5(a)	1	H	SR 3.3.1.1.16 SR 3.3.1.1.18	NA
12. Manual Scram	1,2	1	G	SR 3.3.1.1.8 SR 3.3.1.1.18	NA
	5(a)	1	H	SR 3.3.1.1.8 SR 3.3.1.1.18	NA

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(c) With reactor pressure ≥ 600 psig.

SURVEILLANCE REQUIREMENTS

-----------------------------------NOTE-----------------------------------
When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the Reactor Vessel Pressure—High Function maintains IC initiation capability.

SURVEILLANCE		FREQUENCY
SR 3.3.5.2.1	Perform CHANNEL FUNCTIONAL TEST.	31 days
SR 3.3.5.2.2	----------------NOTE------------------ Not required for the time delay portion of the channel. --------------------------------------- Perform CHANNEL CALIBRATION. The Allowable Value shall be $\leq$ 1068 psig.	92 days
SR 3.3.5.2.3	Perform CHANNEL CALIBRATION for the time delay portion of the channel. The Allowable Value shall be $\leq$ ~~17~~ 15 seconds.	24 months
SR 3.3.5.2.4	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months

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 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~~ 259.2 psid (Unit 2) ≤ 117.1 252.6 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 2.65
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 3.3.6.1-1 (page 3 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
5. Reactor Water Cleanup System Isolation					
a. SLC System Initiation	1,2	1	H	SR 3.3.6.1.7	NA
b. Reactor Vessel Water Level — Low	1,2,3	2	F	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 2.65
6. Shutdown Cooling System Isolation					
a. Recirculation Line Water Temperature — High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 346°F
b. Reactor Vessel Water Level — Low	3,4,5	2(b)	I	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 2.65

(b) In MODES 4 and 5, provided Shutdown Cooling System integrity is maintained, only one channel per trip system with an isolation signal available to one shutdown cooling pump suction isolation valve is required.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	Verify the safety function lift setpoints of the safety valves are as follows: Number of Safety Valves — Setpoint (psig) 1 — 1135 ± 11.3 ~~2 — 1240 ± 12.4~~ 2 — 1250 ± 12.5 4 — 1260 ± 12.6	In accordance with the Inservice Testing Program
SR 3.4.3.2	-------------------NOTE------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify each relief valve opens when manually actuated.	24 months
SR 3.4.3.3	-------------------NOTE------------------- Valve actuation may be excluded. -- Verify each relief valve actuates on an actual or simulated automatic initiation signal.	24 months

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

b. A loss of safety function exists when, assuming no concurrent single failure, and assuming no concurrent loss of offsite power or loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

1. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

2. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

3. A required system redundant to support system(s) for the supported systems described in b.1 and b.2 above is also inoperable.

c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.12 Primary Containment Leakage Rate Testing Program

a. This program shall establish the leakage testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemption. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program," dated September 1995.

b. The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 48 psig.
43.9

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

3. The LHGR for Specification 3.2.3.

4. The ~~LHGR~~ and transient linear heat generation ~~rate~~ limit ~~for~~ Specification 3.2.4.

4. ~~5.~~ Control Rod Block Instrumentation Setpoint for the Rod Block Monitor — Upscale Function Allowable Value for Specification 3.3.2.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. ANF-1125(P)(A), "Critical Power Correlation - ANFB."

2. ANF-524(P)(A), "ANF Critical Power Methodology for Boiling Water Reactors."

3. XN-NF-79-71(P)(A), "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors."

4. XN-NF-80-19(P)(A), "Exxon Nuclear Methodology for Boiling Water Reactors."

5. XN-NF-85-67(P)(A), "Generic Mechanical Design for Exxon Nuclear Jet Pump Boiling Water Reactors Reload Fuel."

6. ANF-913(P)(A), "CONTRANSA2: A Computer Program for Boiling Water Reactor Transient Analysis."

7. XN-NF-82-06(P)(A), Qualification of Exxon Nuclear Fuel for Extended Burnup Supplement 1 Extended Burnup Qualification of ENC 9x9 BWR Fuel.

8. ANF-89-14(P)(A), Advanced Nuclear Fuels Corporation Generic Mechanical Design for Advance Nuclear Fuels Corporation 9x9-IX and 9x9-9X BWR Reload Fuel.

(continued)

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 the Commonwealth Edison (ComEd) Company, is requesting changes to the Technical Specifications (TS) for Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2. These proposed changes are in addition to the changes proposed to support uprated power operation at QCNPS in Reference I.1. These additional proposed changes raises the allowable value for the reactor scram on fixed neutron flux – high from 120% of rated thermal power (RTP) to 122% of RTP. These proposed changes also affect the clamped (i.e., maximum) allowable value for the reactor scram on flow-biased neutron flux - high.

The additional changes proposed in this supplemental amendment request are related to the previously proposed changes for power uprate. The previously requested amendment for power uprate included changes related to the Average Power Range Monitor (APRM) and Rod Block Monitor (RBM) TS and are referred to in this supplemental request as the Partial APRM RBM TS (ARTS) changes. As described below, the previous limitation of 120% for this allowable value is no longer necessary. The allowable value proposed in this supplemental request is now based on the analytical limit for the neutron flux scram assumed in the transient and accident analyses for QCNPS.

B. DESCRIPTION OF THE CURRENT REQUIREMENTS

TS Section 3.3.1.1, "Reactor Protection System Instrumentation"

TS Table 3.3.1.1-1 Function 2.b

Table 3.3.1.1-1, "Reactor Protection System Instrumentation," function 2.b identifies the allowable value for the APRM Flow Biased Neutron Flux - High function. For two-loop operation, the allowable value is ≤ 0.58 W + 63.5% RTP and ≤ 120% RTP.

TS Table 3.3.1.1-1 Function 2.c

Table 3.3.1.1-1, function 2.c identifies the allowable value for the APRM Fixed Neutron Flux – High function. The allowable value is ≤ 120% RTP.

C. BASES FOR THE CURRENT REQUIREMENTS

The APRM neutron flux trip level is varied as a function of recirculation drive flow. At lower core flows, the setpoint is reduced proportional to the reduction in power experienced as core flow is reduced. This value is clamped at an upper limit that is equivalent to the APRM Neutron Flux – High function allowable value. The APRM Flow Biased Neutron Flux - High function provides protection against transients where

thermal power increases slowly and protects the fuel cladding integrity by ensuring that the Minimum Critical Power Ratio (MCPR) Safety Limit (SL) is not exceeded. During any transient event that occurs at a reduced recirculation flow, because of a lower scram trip setpoint, the APRM Flow Biased Neutron – High function will initiate a scram before the clamped allowable value is reached.

The APRM gain or APRM Flow Biased Neutron Flux-High function allowable value is required to be adjusted when operating under conditions of excessive power peaking to maintain acceptable margins to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit. The condition of excessive power peaking protection of the fuel cladding integrity by ensuring that the MCPR SL is not exceeded is determined by the Fuel Design Limiting Ratio for Centerline Melt (FDLRC), which is defined as follows.

$$\text{FDLRC} = \frac{(\text{LHGR})\,(1.2)}{(\text{TLHGR})\,(\text{FRTP})}$$

where

LHGR	= Linear Heat Generation Rate
FRTP	= Fraction of Rated Thermal Power
TLHGR	= Transient Linear Heat Generation Rate

Maintaining FDLRC less than or equal to 1.0 ensures the fuel does not experience centerline melt during abnormal operational occurrences (AOOs) beginning at any power level and terminating at 120% RTP (i.e., the APRM Fixed Neutron Flux - High allowable value).

The APRM Fixed Neutron Flux – High function generates a trip signal to prevent fuel damage or excessive Reactor Coolant System (RCS) pressure. For the overpressurization protection analysis, the APRM Fixed Neutron Flux – High function is assumed to terminate the main steam isolation valve (MSIV) closure event and, along with the safety valves, limits the peak reactor pressure vessel (RPV) pressure to less than the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code limits. The control rod drop accident (CRDA) analysis also takes credit for the APRM Fixed Neutron Flux – High function to mitigate the consequences of the CRDA.

D. NEED FOR REVISION OF THE REQUIREMENTS

Because the analyses performed for the power uprate proposed changes are based on the Partial ARTS operating strategy, use of FDLRC as a fuel thermal limit is no longer required. With the Partial ARTS operating strategy, monitoring to ensure that the MCPR SL is not exceeded is performed with power and flow dependent limits. These power and flow dependent limits are fully described in Reference I.1. Use of the Partial ARTS operating strategy removes the need to limit the allowable value for the neutron flux scram to the 120% value dictated by the FDLRC equation, and allows use of an allowable value that results directly from the safety analyses.

Raising the allowable value for the neutron flux scram provides operational flexibility and reduces the potential for unnecessary actuations of safety systems.

E. DESCRIPTION OF THE PROPOSED CHANGES

<u>TS Table 3.3.1.1-1 Function 2.b</u>

The clamped portion of the allowable value for the APRM Flow-Biased Neutron Flux – High is changed from ≤ 120% RTP to ≤ 122% RTP. This is in addition to the changes to the flow-biased portion that were discussed in Reference I.1.

<u>TS Table 3.3.1.1-1 Function 2.c</u>

The allowable value is changed from ≤ 120% RTP to ≤ 122% RTP.

F. SUMMARY SAFETY ANALYSIS OF THE PROPOSED CHANGES

The analyses that were performed for the power uprate related to the transient events listed in the bases above assume an analytical limit of 125% RTP for the APRM Fixed Neutron Flux - High function and the clamped value of the the APRM Flow-Biased Neutron Flux – High function. The APRM setpoint calculations determined that, based on this analytical limit, an allowable value of 122% is appropriate and ensures the analytical limit is maintained. An allowable value of 120% had been previously used because this value is conservative and consistent with the FDLRC equation. Because the analyses performed for the power uprate proposed changes are based on the Partial ARTS operating strategy, use of FDLRC is no longer required. With the Partial ARTS operating strategy, monitoring to ensure that the MCPR SL is not exceeded is performed with power and flow dependent limits. These power and flow dependent limits are fully described in Reference I.1. Use of the Partial ARTS operating strategy removes the need to limit the allowable value for the neutron flux scram to the 120% value dictated by the FDLRC equation, and allows use of an allowable value that results directly from the safety analyses.

The setpoint calculations for the proposed changes were performed in accordance with the the General Electric setpoint methodology specified in Reference I.2.

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 Reference I.1.

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

H. SCHEDULE REQUIREMENTS

EGC requests that these proposed changes be reviewed and approved as part of the proposed changes for power uprate operation in Reference I.1, which were proposed to be approved by January 15, 2002 to support operation at uprated power following the QCNPS refueling outage scheduled to begin in February 2002.

I. REFERENCE

1. Letter from R. M. Krich (ComEd) to U.S. NRC, "Request for Technical Specifications Changes for Dresden Nuclear Power Station, Units 2 and 3 and Quad Cities Nuclear Power Station, Units 1 and 2, to Allow Operation at Uprated Power Levels," dated December 27, 2000

2. General Electric Nuclear Energy, "General Electric Instrument Setpoint Methodology," NEDC-31336P-A, Class III (Proprietary), September 1996 (Note: will add letter citation before sending)

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

3.3.1.1-7
3.3.1.1-8

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux − High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.16 SR 3.3.1.1.17	≤ 121/125 divisions of full scale
	5(a)	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.16 SR 3.3.1.1.17	≤ 121/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.17	NA
	5(a)	3	H	SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.17	NA
2. Average Power Range Monitors					
a. Neutron Flux − High, Setdown	2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.7 SR 3.3.1.1.9 SR 3.3.1.1.14 SR 3.3.1.1.17	≤ 17.1% RTP
b. Flow Biased Neutron Flux − High	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.14 SR 3.3.1.1.16 SR 3.3.1.1.17 SR 3.3.1.1.18	0.56W + 67.4% ≤ 0.58 W + 63.4% RTP and ≤ 120% RTP(b) 122% Revised

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) 0.58 W + 59.1% and ≤ 118.4% RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."
0.56W + 63.2%

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
c. Fixed Neutron Flux – High	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.14 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ ~~120%~~ 122% RTP
d. Inop	1,2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.17	NA
3. Reactor Vessel Steam Dome Pressure – High	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.5 SR 3.3.1.1.10 SR 3.3.1.1.11 SR 3.3.1.1.16 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 1050 psig
4. Reactor Vessel Water Level – Low	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.5 SR 3.3.1.1.10 SR 3.3.1.1.11 SR 3.3.1.1.16 SR 3.3.1.1.17 SR 3.3.1.1.18	≥ ~~11.8~~ 3.8 inches
5. Main Steam Isolation Valve – Closure	1	8	F	SR 3.3.1.1.5 SR 3.3.1.1.10 SR 3.3.1.1.16 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 9.8% closed
6. Drywell Pressure – High	1,2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.10 SR 3.3.1.1.12 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 2.43 psig

| Revised

(continued)

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.

Exelon Generation Company (EGC), LLC is proposing changes to the Technical Specifications (TS), for Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2. The proposed changes raise the allowable value for the reactor scram on fixed neutron flux – high from 120% of rated thermal power (RTP) to 122% RTP. The proposed changes also raise the maximum allowable value for the reactor scram on flow-biased neutron flux – high from 120% RTP to 122% RTP. The proposed changes are associated with previously-proposed changes to revise the maximum power level specified in each unit's license, and TS definition of rated thermal power.

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

The proposed changes modifiy the allowable value for an instrument setpoint for the neutron flux at which a reactor scram initiates to mitigate an accident or transient. The changes do not modify the analytical limit (i.e., the value assumed in the safety analyses) for this function. Because these changes do not affect the way in which plant equipment functions or create any new failure mechanisms and therefore do not affect the initiation of any accidents previously evaluated. Therefore the proposed changes do not affect the probability of an accident previously evaluated.

The proposed changes do not affect the analytical limit at which the reactor scram is assumed to actuate in the transient and accident analyses for QCNPS. The calculation for the revised allowable value maintains the same analytical limit, and therefore the same assumed actuation level for the reactor scram function. Therefore, the proposed changes do not involve a significant increase in the consequences of an accident previously evaluated.

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

The proposed changes modify the allowable value for an instrument setpoint for the neutron flux at which a reactor scram initiates to mitigate an accident or transient. The changes do not modify the analytical limit (i.e., the value assumed in the safety analyses) for this function. These changes do not affect the way in which plant equipment functions or create any new failure mechanisms. Therefore the proposed changes do 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 changes do not affect the analytical limit at which the reactor scram is assumed to actuate in the transient and accident analyses for QCNPS. The calculation for the revised allowable value maintains the same analytical limit as currently assumed in the DNPS TS and in the analyses performed for the power uprate license amendment request. Therefore the assumed actuation level for the reactor scram function remains unchaged. Thus, the assumptions and results in the transient and accident analyses are unchanged. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

Conclusion

The proposed changes do not involve a significant hazards consideration.

ENCLOSURE 2 - ATTACHMENT D

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

INFORMATION SUPPORTING AN ENVIRONMENTAL ASSESSMENT

Exelon Generation Company (EGC), LLC has evaluated these proposed changes 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 these proposed changes meet 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 proposed changes meet the following specific criteria.

(i) The proposed changes involve no significant hazards consideration.

As demonstrated in Attachment C, the proposed changes do 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 changes raise the allowable value for the reactor scram on fixed neutron flux –high from 120% of rated thermal power (RTP) to 122% of RTP. The proposed changes also raise the maximum allowable value for the reactor scram on flow-biased neutron flux – high from 120% RTP to 122% RTP. The changes do not allow for an increase in the unit power level, do not increase the production, nor alter the flow path or method of disposal of radioactive waste or byproducts. Therefore, the proposed changes do not affect actual unit effluents.

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

The proposed changes 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 changes 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 these changes.

ENCLOSURE 2 – ATTACHMENT E
Supplement to Request for Power Uprate Operation
Quad Cities Nuclear Power Station, Units 1 and 2

REVISED PAGES FOR POWER UPRATE LICENSE AMENDMENT REQUEST

The attached pages replace the corresponding pages in the Letter from R.M. Krich (Commonwealth Edison Company) to U.S. NRC, "Request for License Amendment for Power Uprate Operation," dated December 27, 2000

<u>REVISED LIST OF PLANT MODIFICATIONS (Attachment G of License Amendment Request)</u>

Page 1 of 2
Page 2 of 2

ATTACHMENT G – REVISED

Proposed Changes to Operating Licenses and Technical Specifications for Quad Cities Nuclear Power Station, Units 1 and 2

PLANT MODIFICATIONS REQUIRED TO SUPPORT POWER UPRATE

The following presents an overview of the facility changes necessary to achieve the target electrical power output of 912 MWe.

- Various instruments will require scaling/setpoint changes.
- A modification to provide tripping of the 4th condensate pump on a LOCA will be implemented to allow the continued use of the feedwater pumps.
- A fault current limiting arrangement will be implemented to maintain non-safety bus short circuit ratings after a postulated loss of an auxiliary transformer in conjunction with a short circuit.
- A reactor recirculation pump runback on a loss of feedwater flow or the loss of a condensate pump will be implemented to reduce the potential for a scram on reactor low water level and allow continued operation.
- An additional steam line resonance compensator card designed to attenuate third order harmonics will be installed in the electro-hydraulic control system to reduce electrical noise in the system.
- A new high-pressure turbine rotor will be installed as a result of the increased steam flow associated with operation at uprated power conditions.
- Turbine cross around relief valve alterations will be performed to ensure that pressure limitations are not exceeded.
- Selected heater drain valve normal drain trim replacements will be performed due to the increase in drain flow.
- Some feedwater heater relief valves will be adjusted or replaced and the heaters will be rerated to compensate for the increased feedwater flow and the associated pressure change.
- Condenser tube staking was planned for the main condensers to provide adequate protection against tube vibration damage at uprated power conditions. This modification has been determined to be unnecessary as a result of further condenser analysis.
- An additional condensate demineralizer will be added to process the increased flow.
- Various support and piping modifications will be performed due to the increased temperature in torus-attached piping and increased temperature and flow in the main steam and feedwater systems.
- Restriction orifices to the stator water cooling system was planned to be resized to accommodate the increased heat load. This modification has been determined to be unnecessary as a result of further analysis.

- Modifications to the steam dryer will be performed to reduce moisture carryover.
- Clamps will be added to selected reactor vessel jet pump sensing lines to reduce vibration

ENCLOSURE 2 - ATTACHMENT F

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

MARKED-UP POWER UPRATE TECHNICAL SPECIFICATION PAGES BASED ON APPROVED VERSION OF IMPROVED TECHNICAL SPECIFICATIONS

<u>REVISED PAGES</u>

1.1-3
1.1-4
1.1-5
3.2.4-1
3.2.4-2
3.3.1.1-2
3.3.1.1-3
3.3.1.1-5
3.3.1.1-7
3.3.1.1-8
3.3.1.1-9
3.3.6.1-5
3.3.6.1-7
5.5-11
5.6-3

1.1 Definitions

DOSE EQUIVALENT I-131 (continued)	conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites;" Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977; or ICRP 30, Supplement to Part 1, pages 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."
~~FUEL DESIGN LIMITING RATIO FOR CENTERLINE MELT (FDLRC)~~	~~The FDLRC shall be 1.2 times the LHGR existing at a given location divided by the product of the transient LHGR limit and the fraction of RTP.~~
LEAKAGE	LEAKAGE shall be: a. <u>Identified LEAKAGE</u> 1. LEAKAGE into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or 2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; b. <u>Unidentified LEAKAGE</u> All LEAKAGE into the drywell that is not identified LEAKAGE; c. <u>Total LEAKAGE</u> Sum of the identified and unidentified LEAKAGE; and d. <u>Pressure Boundary LEAKAGE</u> LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall.

(continued)

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 logic components required for OPERABILITY 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)

1.1 Definitions

OPERABLE — OPERABILITY (continued)

are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).

RATED THERMAL POWER (RTP)

RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~2511~~ 2957 MWt.

REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME

The RPS RESPONSE TIME shall be that time interval from the opening of the sensor contact until the opening of the trip actuator. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

SHUTDOWN MARGIN (SDM)

SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that:

a. The reactor is xenon free;

b. The moderator temperature is 68°F; and

c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn.

   With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.

STAGGERED TEST BASIS

A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are

(continued)

3.2 POWER DISTRIBUTION LIMITS

3.2.4 Average Power Range Monitor (APRM) Gain and Setpoint

LCO 3.2.4

a. FDLRC and the ratio of MFLPD to Fraction of RTP (FRTP) shall be less than or equal to 1.0; or

b. Each required APRM Flow Biased Neutron Flux—High Function Allowable Value shall be modified by the lesser of 1/FDLRC or FRTP/MFLPD; or

c. Each required APRM gain shall be adjusted such that the APRM readings are $\geq$ 100% times the higher of FRTP times FDLRC or of MFLPD.

APPLICABILITY: THERMAL POWER $\geq$ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Satisfy the requirements of the LCO.	6 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	-------------------NOTE------------------- Not required to be met if SR 3.2.4.2 is satisfied for LCO 3.2.4.b or LCO 3.2.4.c requirements. -- Verify FDLRC and the ratio of MFLPD to FRTP are within limits.	Once within 12 hours after ≥ 25% RTP AND 24 hours thereafter
SR 3.2.4.2	-------------------NOTE------------------- Not required to be met if SR 3.2.4.1 is satisfied for LCO 3.2.4.a requirements. -- Verify each required: a. APRM Flow Biased Neutron Flux — High Function Allowable Value is modified by less than or equal to the lesser of 1/FDLRC or FRTP/MFLPD; or b. APRM gain is adjusted such that the APRM reading is ≥ 100% times the higher of FRTP times FDLRC or of MFLPD.	12 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more Functions with RPS trip capability not maintained.	C.1 Restore RPS trip capability.	1 hour
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1 Reduce THERMAL POWER to < ~~45%~~ RTP. 38.5%	4 hours
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1 Be in MODE 2.	8 hours
G. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1 Be in MODE 3.	12 hours
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1 Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

------------------------------------NOTES------------------------------------

1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.

2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.1.2	-----------------NOTE----------------- Not required to be performed until 12 hours after THERMAL POWER ≥ 25% RTP. -------------------------------------- Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is ≤ 2% RTP ~~, plus any gain adjustment required by LCO 3.2.4, "Average Power Range Monitor (APRM) Gain and Setpoint" while operating at ≥ 25% RTP.~~	7 days
SR 3.3.1.1.3	Adjust the channel to conform to a calibrated flow signal.	7 days

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.11	Calibrate the trip units.	92 days
SR 3.3.1.1.12	Perform CHANNEL CALIBRATION.	92 days
SR 3.3.1.1.13	Verify Turbine Stop Valve — Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure — Low Functions are not bypassed when THERMAL POWER is ≥ 45% RTP. 38.5%	92 days
SR 3.3.1.1.14	------------------NOTES------------------ 1. Neutron detectors are excluded. 2. For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 24 hours after entering MODE 2. 3. For Function 2.b, not required for the flow portion of the channels. -- Perform CHANNEL CALIBRATION.	184 days
SR 3.3.1.1.15	Perform CHANNEL FUNCTIONAL TEST.	24 months

(continued)

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux – High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.16 SR 3.3.1.1.17	≤ 121/125 divisions of full scale
	5(a)	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.16 SR 3.3.1.1.17	≤ 121/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.17	NA
	5(a)	3	H	SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.17	NA
2. Average Power Range Monitors					
a. Neutron Flux – High, Setdown	2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.7 SR 3.3.1.1.9 SR 3.3.1.1.14 SR 3.3.1.1.17	≤ 17.1% RTP
b. Flow Biased Neutron Flux – High	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.14 SR 3.3.1.1.16 SR 3.3.1.1.17 SR 3.3.1.1.18	0.56W + 67.4% ≤ ~~0.58 W + 63.4%~~ RTP and ≤ ~~120%~~ RTP(b) 122%

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) ~~0.58 W + 59.1%~~ and ≤ 118.4% RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."
0.56W + 63.2%

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
c. Fixed Neutron Flux – High	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.14 SR 3.3.1.1.17 SR 3.3.1.1.18	$\leq$ ~~120%~~ RTP 122%
d. Inop	1,2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.17	NA
3. Reactor Vessel Steam Dome Pressure – High	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.5 SR 3.3.1.1.10 SR 3.3.1.1.11 SR 3.3.1.1.16 SR 3.3.1.1.17 SR 3.3.1.1.18	$\leq$ 1050 psig
4. Reactor Vessel Water Level – Low	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.5 SR 3.3.1.1.10 SR 3.3.1.1.11 SR 3.3.1.1.16 SR 3.3.1.1.17 SR 3.3.1.1.18	$\geq$ ~~17.8~~ inches 3.8
5. Main Steam Isolation Valve – Closure	1	8	F	SR 3.3.1.1.5 SR 3.3.1.1.10 SR 3.3.1.1.16 SR 3.3.1.1.17 SR 3.3.1.1.18	$\leq$ 9.8% closed
6. Drywell Pressure – High	1,2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.10 SR 3.3.1.1.12 SR 3.3.1.1.17 SR 3.3.1.1.18	$\leq$ 2.43 psig

(continued)

Table 3.3.1.1-1 (page 3 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
7. Scram Discharge Volume Water Level − High					
a. Thermal Switch	1,2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.10 SR 3.3.1.1.16 SR 3.3.1.1.17	≤ 38.9 gallons
	5(a)	2	H	SR 3.3.1.1.5 SR 3.3.1.1.10 SR 3.3.1.1.16 SR 3.3.1.1.17	≤ 38.9 gallons
b. Differential Pressure Switch	1,2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.10 SR 3.3.1.1.16 SR 3.3.1.1.17	≤ 32.3 gallons
	5(a)	2	H	SR 3.3.1.1.5 SR 3.3.1.1.10 SR 3.3.1.1.16 SR 3.3.1.1.17	≤ 32.3 gallons
8. Turbine Stop Valve − Closure	≥ ~~45%~~ RTP 38.5%	4	E	SR 3.3.1.1.5 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.16 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 9.7% closed
9. Turbine Control Valve Fast Closure, Trip Oil Pressure − Low	≥ ~~45%~~ RTP 38.5%	2	E	SR 3.3.1.1.5 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.16 SR 3.3.1.1.17 SR 3.3.1.1.18	≥ 475 psig
10. Turbine Condenser Vacuum − Low	1	2	F	SR 3.3.1.1.5 SR 3.3.1.1.10 SR 3.3.1.1.12 SR 3.3.1.1.17 SR 3.3.1.1.18	21.6 ≥ ~~21.8~~ inches Hg vacuum
11. Reactor Mode Switch − Shutdown Position	1,2	1	G	SR 3.3.1.1.15 SR 3.3.1.1.17	NA
	5(a)	1	H	SR 3.3.1.1.15 SR 3.3.1.1.17	NA
12. Manual Scram	1,2	1	G	SR 3.3.1.1.8 SR 3.3.1.1.17	NA
	5(a)	1	H	SR 3.3.1.1.8 SR 3.3.1.1.17	NA

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

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	≥ -55.2 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	≤ 0.331 seconds
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 3.3.6.1.6 SR 3.3.6.1.7	254.3 psid ≤ ~~138% rated steam flow~~
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	≤ 198°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	3.8 ≥ ~~11.8~~ 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	≤ 2.43 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	≤ 70 R/hr

(continued)

Table 3.3.6.1-1 (page 3 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
5. Reactor Water Cleanup System Isolation					
a. SLC System Initiation	1,2	1	H	SR 3.3.6.1.7	NA
b. Reactor Vessel Water Level − Low	1,2,3	2	F	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	3.8 ≥ 11.8 inches
6. RHR Shutdown Cooling System Isolation					
a. Reactor Vessel Pressure − High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.7	≤ 130 psig
b. Reactor Vessel Water Level − Low	3,4,5	2(b)	I	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	3.8 ≥ 11.8 inches

(b) In MODES 4 and 5, provided RHR Shutdown Cooling System integrity is maintained, only one channel per trip system with an isolation signal available to one shutdown cooling pump suction isolation valve is required.

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

b. A loss of safety function exists when, assuming no concurrent single failure, and assuming no concurrent loss of offsite power or loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

1. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

2. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

3. A required system redundant to support system(s) for the supported systems described in b.1 and b.2 above is also inoperable.

c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.12 Primary Containment Leakage Rate Testing Program

a. This program shall establish the leakage testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemption. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program," dated September 1995.

b. The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is ~~48~~ psig.

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(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

3. The LHGR for Specification 3.2.3.

4. The LHGR and transient linear heat generation rate limit for Specification 3.2.4.

4. Control Rod Block Instrumentation Setpoint for the Rod Block Monitor—Upscale Function Allowable Value for Specification 3.3.2.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel."

2. Commonwealth Edison Topical Report NFSR-0085, "Benchmark of BWR Nuclear Design Methods."

3. Advanced Nuclear Fuels Methodology for Boiling Water Reactors, XN-NF-80-19(P)(A).

4. Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel, XN-NF-85-67(P)(A).

5. Qualification of Exxon Nuclear Fuel for Extended Burnup, XN-NF-82-06(P)(A).

6. Advanced Nuclear Fuels Corporation Generic Mechanical Design for Advanced Nuclear Fuels 9x9-IX and 9x9-9X BWR Reload Fuel, ANF-89-014(P)(A).

7. Generic Mechanical Design Criteria for BWR Fuel Designs, ANF-89-98(P)(A).

8. Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, XN-NF-79-71(P)(A).

9. ANFB Critical Power Correlation, ANF-1125(P)(A).

(continued)