Commonwealth Edison Company
1400 Opus Place
Downers Grove, IL 60515-5701

ComEd

March 1, 2000

United States Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. STN 50-456 and STN 50-457

Byron Station, Units 1 and 2
Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. STN 50-454 and STN 50-455

Subject: Supplement to Request for Technical Specifications Change
Expanded Core Operating Limits Reports

References: (1) Letter from R. M. Krich (Commonwealth Edison Company) to U.S. NRC, "Request for Technical Specifications Change - Expanded Core Operating Limits Reports," dated December 22, 1999.
(2) Teleconference with G. Dick (U.S. NRC) and K. M. Root (Commonwealth Edison Company) on February 1, 2000.

In Reference 1, in accordance with 10 CFR 50.90, we requested changes to Technical Specifications (TS), of Facility Operating License Nos. NPF-72, NPF-77, NPF-37 and NPF-66 for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, respectively. The proposed changes relocate Reactor Coolant System related cycle-specific parameter limits from the TS to the Core Operating Limits Reports (COLRs) for Braidwood Station and Byron Station.

As a result of a subsequent teleconference with the NRC (Reference 2), we are providing this supplement to Reference 1. The purpose of this supplement is to incorporate changes contained in NRC approved Technical Specifications Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-339, Revision 1, "Expanded COLR," and changes discussed in Reference 2. The affected pages have been revised to reflect this revision and are attached to this letter.

This revision does not affect the supporting safety analysis as described in Attachment A to the Reference 1 letter. Based on the fact that this revision does not alter the previously provided information supporting a finding of no significant hazards consideration, we have concluded that the public notice of the proposed change, published in the Federal Register on February 23, 2000, is unaffected by this supplement.

A001

This supplement to the proposed change has been reviewed and approved in accordance with the requirements of the ComEd Quality Assurance Program.

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

Should you have any questions relative to this submittal, please contact Ms. Kelly M. Root at (630) 663-7292.

Respectfully,

R. M. Krich
Vice President – Regulatory Services

Attachments: Affidavit
Attachment A-1 Marked-Up TS Pages for Byron Station
Attachment A-2 Marked-Up TS Pages for Braidwood Station
Attachment A-3, Clean Copy TS Pages for Byron Station
Attachment A-4, Clean Copy TS Pages for Braidwood Station
Attachment A-5, Clean Copy Bases Pages for Byron Station (Info. Only)
Attachment A-6, Clean Copy Bases Pages for Braidwood Station (Info. Only)

cc: Regional Administrator - NRC Region III
NRC Senior Resident Inspector - Braidwood Station
NRC Senior Resident Inspector - Byron Station
Office of Nuclear Facility Safety - Illinois Department of Nuclear Safety

STATE OF ILLINOIS	)	
COUNTY OF DUPAGE	)	
IN THE MATTER OF	)	
COMMONWEALTH EDISON (COMED) COMPANY	)	**Docket Nos.**
BRAIDWOOD STATION - UNITS 1 and 2	)	**STN 50-456 and STN 50-457**
BYRON STATION - UNITS 1 and 2	)	**STN 50-454 and STN 50-455**

SUBJECT: Supplement to Request for Technical Specifications Change, Expanded Core Operating Limits Reports

AFFIDAVIT

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

OFFICIAL SEAL
LAURA KAMINSKAS
NOTARY PUBLIC, STATE OF ILLINOIS
MY COMMISSION EXPIRES 6-1-2002

R. M. Krich
Vice President - Regulatory Services

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

for the State above named, this 1 day of

March, 2000.

Laura Kaminskas
Notary Public

(OFFICIAL SEAL)

ATTACHMENT A-1

MARKED-UP TS PAGES
FOR BYRON STATION, UNITS 1 AND 2

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the ~~SLs~~ limits specified in the COLR; and ~~Figure 2.1.1-1~~. the following SLs shall not be exceeded.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

2.1.1.1 In MODE 1, the Departure from Nucleate Boiling Ratio (DNBR) shall be maintained ≥ 1.25 for the WRB-2 DNB correlation.

2.1.1.2 In MODE 2, the DNBR shall be maintained ≥ 1.17 for the WRB-2 DNB correlation, and ≥ 1.30 for the W-3 DNB correlation.

2.1.1.3 In MODES 1 and 2, the peak fuel centerline temperature shall be maintained ≤ 4700°F.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

SL 2.1.1, "Reactor Core SLs";
LCO 3.1.1, "SHUTDOWN MARGIN (SDM)";
LCO 3.1.3, "Moderator Temperature Coefficient";
LCO 3.1.5, "Shutdown Bank Insertion Limits";
LCO 3.1.6, "Control Bank Insertion Limits";
LCO 3.1.8, "PHYSICS TESTS Exceptions - MODE 2";
LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)";
LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)";
LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)"; ~~and~~
LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"; and
LCO 3.9.1, "Boron Concentration";

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. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluations Methodology," July 1985.

2. WCAP-8385, "Power Distribution Control and Load Following Procedures-Topical Report," September 1974.

3. NFSR-0016, "Commonwealth Edison Company Topical Report on Benchmark of PWR Nuclear Design Methods," July 1983.

4. NFSR-0081, "Commonwealth Edison Company Topical Report on Benchmark of PWR Nuclear Design Methods Using the Phoenix-P and ANC Computer Codes," July 1990.

5. ComEd letter from D. Saccomando to the Office of Nuclear Reactor Regulation dated December 21, 1994, transmitting an attachment that documents applicable sections of WCAP-11992/11993 and ComEd application of the UET methodology addressed in "Additional Information Regarding Application for Amendment to Facility Operating Licenses-Reactivity Control Systems."

ATTACHMENT A-2

MARKED-UP TS PAGES
FOR BRAIDWOOD STATION, UNITS 1 AND 2

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the ~~SLs~~ limits specified in the COLR; and ~~Figure 2.1.1-1~~. the following SLs shall not be exceeded.

2.1.1.1 In MODE 1, the Departure from Nucleate Boiling Ratio (DNBR) shall be maintained ≥ 1.25 for the WRB-2 DNB correlation.

2.1.1.2 In MODE 2, the DNBR shall be maintained ≥ 1.17 for the WRB-2 DNB correlation, and ≥ 1.30 for the W-3 DNB correlation.

2.1.1.3 In MODES 1 and 2, the peak fuel centerline temperature shall be maintained ≤ 4700°F.

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In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2735 psig.

2.2 SL Violations

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2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

SL 2.1.1, "Reactor Core SLs";
LCO 3.1.1, "SHUTDOWN MARGIN (SDM)";
LCO 3.1.3, "Moderator Temperature Coefficient";
LCO 3.1.5, "Shutdown Bank Insertion Limits";
LCO 3.1.6, "Control Bank Insertion Limits";
LCO 3.1.8, "PHYSICS TESTS Exceptions - MODE 2";
LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)";
LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)";
LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)"; ~~and~~
LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"; and
LCO 3.9.1, "Boron Concentration";

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. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluations Methodology," July 1985.

2. WCAP-8385, "Power Distribution Control and Load Following Procedures-Topical Report," September 1974.

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ATTACHMENT A-3

CLEAN COPY TS PAGES
FOR BYRON STATION, UNITS 1 AND 2

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded.

2.1.1.1 In MODE 1, the Departure from Nucleate Boiling Ratio (DNBR) shall be maintained ≥ 1.25 for the WRB-2 DNB correlation.

2.1.1.2 In MODE 2, the DNBR shall be maintained ≥ 1.17 for the WRB-2 DNB correlation, and ≥ 1.30 for the W-3 DNB correlation.

2.1.1.3 In MODES 1 and 2, the peak fuel centerline temperature shall be maintained ≤ 4700°F.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

SL 2.1.1, "Reactor Core SLs";
LCO 3.1.1, "SHUTDOWN MARGIN (SDM)";
LCO 3.1.3, "Moderator Temperature Coefficient";
LCO 3.1.5, "Shutdown Bank Insertion Limits";
LCO 3.1.6, "Control Bank Insertion Limits";
LCO 3.1.8, "PHYSICS TESTS Exceptions - MODE 2";
LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)";
LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)";
LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)";
LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"; and
LCO 3.9.1, "Boron Concentration";

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:

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ATTACHMENT A-4

CLEAN COPY TS PAGES
FOR BRAIDWOOD STATION, UNITS 1 AND 2

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded.

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2.1.1.2 In MODE 2, the DNBR shall be maintained ≥ 1.17 for the WRB-2 DNB correlation, and ≥ 1.30 for the W-3 DNB correlation.

2.1.1.3 In MODES 1 and 2, the peak fuel centerline temperature shall be maintained ≤ 4700°F.

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2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

SL 2.1.1, "Reactor Core SLs";
LCO 3.1.1, "SHUTDOWN MARGIN (SDM)";
LCO 3.1.3, "Moderator Temperature Coefficient";
LCO 3.1.5, "Shutdown Bank Insertion Limits";
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LCO 3.1.8, "PHYSICS TESTS Exceptions - MODE 2";
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LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)";
LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)";
LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"; and
LCO 3.9.1, "Boron Concentration";

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ATTACHMENT A-5

CLEAN COPY ITS BASES PAGES (INFORMATION ONLY) FOR BYRON STATION, UNITS 1 AND 2

APPLICABLE SAFETY ANALYSES (continued)

Automatic preservation of these reactor core SLs is provided by the appropriate operation of the RPS and the MSSVs (Ref. 2).

The SLs represent a design requirement for establishing the RPS trip setpoints identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," or the assumed initial conditions of the safety analyses provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS

The figure in the COLR shows the reactor core limits of THERMAL POWER, RCS pressure, and average temperature for which the minimum DNBR is not less than the safety analyses limit, that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or that the core exit quality is within the limits defined by the DNBR correlation.

The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability that the hot fuel pellet in the core must not experience centerline fuel melting; and

b. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB.

BASES

SAFETY LIMITS (continued)

The reactor core SLs are used to defined the various RPS functions such that the above criteria are satisfied during steady state operation, normal operational transients, and AOOs. To ensure that the RPS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature ΔT and Overpower ΔT reactor trip functions. That is, it must be demonstrated that the average enthalpy in the hot leg is less than or equal to the saturation enthalpy and that the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RPS and the MSSVs ensure that for variations in the RCS average temperature, pressurizer pressure, RCS flow, AFD, and THERMAL POWER that the reactor core SLs will be satisfied during steady state operation, normal operational transients, and AOOs.

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The MSSVs or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1. In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

SAFETY LIMITS VIOLATIONS

If SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. UFSAR, Section 7.2.

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ATTACHMENT A-6

CLEAN COPY ITS BASES PAGES (INFORMATION ONLY) FOR BRAIDWOOD STATION, UNITS 1 AND 2

FOR INFORMATION ONLY

APPLICABLE SAFETY ANALYSES (continued)

Automatic preservation of these reactor core SLs is provided by the appropriate operation of the RPS and the MSSVs (Ref. 2).

The SLs represent a design requirement for establishing the RPS trip setpoints identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," or the assumed initial conditions of the safety analyses provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS

The figure in the COLR shows the reactor core limits of THERMAL POWER, RCS pressure, and average temperature for which the minimum DNBR is not less than the safety analyses limit, that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or that the core exit quality is within the limits defined by the DNBR correlation.

The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability that the hot fuel pellet in the core must not experience centerline fuel melting; and

b. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB.

BASES

SAFETY LIMITS (continued)

The reactor core SLs are used to defined the various RPS functions such that the above criteria are satisfied during steady state operation, normal operational transients, and AOOs. To ensure that the RPS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature ΔT and Overpower ΔT reactor trip functions. That is, it must be demonstrated that the average enthalpy in the hot leg is less than or equal to the saturation enthalpy and that the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RPS and the MSSVs ensure that for variations in the RCS average temperature, pressurizer pressure, RCS flow, AFD, and THERMAL POWER that the reactor core SLs will be satisfied during steady state operation, normal operational transients, and AOOs.

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The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. UFSAR, Section 7.2.

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