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10 CFR 50.90

RS-00-67

November 13, 2000

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 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: Request for Technical Specifications Change to Delete the Power Range Neutron Flux High Negative Rate Trip Function

References: (1) Westinghouse Topical Report, WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event," dated January 1990.

(2) Letter from A. C. Thadani (U. S. NRC) to R. A. Newton (Westinghouse Owners Group), "Acceptance for Referencing of Licensing Topical Reports WCAP-11394(P) and WCAP-11395(NP), 'Methodology for the Analysis of the Dropped Rod Event,'" dated October 23, 1989.

(3) Letter from R. E. Martin (U. S. NRC) to J. A. Scalice (Tennessee Valley Authority), "Issuance of Amendment Regarding Deletion of Negative Flux Rate Trip for Watts Bar Nuclear Plant, Unit 1," dated January 15, 1999.

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," we are proposing changes to the Technical Specifications (TS) of Facility Operating License Nos. NPF-72, NPF-77, NPF-37 and NPF-66, for the Braidwood Station, Units 1 and

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2, and the Byron Station, Units 1 and 2, respectively. The proposed changes revise TS 3.3.1,"Reactor Trip System (RTS) Instrumentation," to delete the "Power Range Neutron Flux High Negative Rate" Trip Function 3.b from Table 3.3.1-1, "Reactor Trip System Instrumentation." The proposed changes are consistent with the methodology presented in the Westinghouse Topical Report WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event," Reference 1, as accepted by the NRC in Reference 2. Additionally, the proposed changes are consistent with the changes previously approved by the NRC for the Watts Bar Nuclear Plant, Unit 1, as documented in Reference 3.

In Reference 2, the NRC approval of WCAP-11394-P-A stated, "A further review by the staff (for each cycle) is not necessary, given the utility assertion that the analysis described by Westinghouse has been performed and the required comparisons have been made with favorable results." For each fuel cycle design, our Nuclear Generation Group Nuclear Fuel Management administrative procedures will ensure that a dropped Rod Cluster Control Assembly (RCCA) analysis is successfully performed for each fuel cycle in accordance with the methodology described in WCAP-11394-P-A.

We request approval of the proposed changes prior to April 30, 2001, to allow the timely elimination of this unnecessary trip function and thereby reduce the potential for a transient. This would support development of needed procedure changes and required training to implement the proposed changes for all four Braidwood Station and Byron Station units in a timely manner.

This request is subdivided as follows.

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

2. Attachments B-1 and B-2 include the marked-up TS page for the proposed changes for the Braidwood Station and the Byron Station, respectively. Attachments B-3 and B-4 include the associated TS page with the proposed changes incorporated for the Braidwood Station and the Byron Station, respectively. Attachments B-5 and B-6 include the associated TS Bases pages for information only with the proposed changes incorporated for the Braidwood Station and the Byron Station, respectively.

3. Attachment C describes our evaluation performed using the criteria in 10 CFR 50.91(a)(1), "Notice for public comment," which provides information supporting a finding of no significant hazards consideration using the standards in 10 CFR 50.92(c), "Issuance of amendment."

4. Attachment D provides information supporting an environmental assessment and a finding that the proposed changes satisfy the criteria for a categorical exclusion.

The proposed changes have been reviewed by the Braidwood Station and the Byron Station Plant Operations Review Committees and the respective Nuclear Safety Review Boards in accordance with the Quality Assurance Program.

Commonwealth Edison Company is notifying the State of Illinois of this application for changes to the TS by transmitting a copy of this letter and its attachments to the designated State Official.

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

Respectfully,

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

Affidavit

Attachments:
Attachment A: Description and Safety Analysis of the Proposed Changes
Attachment B-1: Marked-Up TS Page for Proposed Changes for Braidwood Station
Attachment B-2: Marked-Up TS Page for Proposed Changes for Byron Station
Attachment B-3: Incorporated TS Page for Proposed Changes for Braidwood Station
Attachment B-4: Incorporated TS Page for Proposed Changes for Byron Station
Attachment B-5: Incorporated TS Bases Pages for Braidwood Station - Information Only
Attachment B-6: Incorporated TS Bases Pages for Byron Station - Information Only
Attachment C: Information Supporting a Finding of No Significant Hazards Consideration
Attachment D: Information Supporting an Environmental Assessment

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

bcc: Project Manager, NRR - Braidwood and Byron Stations
Nicholas Reynolds - Winston & Strawn
Site Vice President - Braidwood Station
Site Vice President - Byron Station
Regulatory Assurance Manager - Braidwood Station
Regulatory Assurance Manager - Byron Station
Licensing Vice President
Manager, Licensing and Compliance - Braidwood and Byron Stations
Nuclear Licensing Administrator - Braidwood Station
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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: Request for Technical Specifications Change to Delete the Power Range Neutron Flux High Negative Rate Trip Function

AFFIDAVIT

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

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 13 day of

November, 2000.

OFFICIAL SEAL
JEFFERY A BATARA
NOTARY PUBLIC, STATE OF ILLINOIS
MY COMMISSION EXPIRES:02/24/02

Notary Public

(OFFICIAL SEAL)

ATTACHMENT A

DESCRIPTION AND SAFETY ANALYSIS OF THE PROPOSED CHANGES

A. SUMMARY OF PROPOSED CHANGES

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," we are proposing changes to the Technical Specifications (TS) of Facility Operating License Nos. NPF-72, NPF-77, NPF-37 and NPF-66, for the Braidwood Station, Units 1 and 2, and the Byron Station, Units 1 and 2, respectively. The proposed changes revise TS 3.3.1,"Reactor Trip System (RTS) Instrumentation," to delete the "Power Range Neutron Flux High Negative Rate" Trip (i.e., Negative Flux Rate Trip (NFRT)) Function 3.b from Table 3.3.1-1, "Reactor Trip System Instrumentation." The proposed changes are consistent with the methodology presented in the Westinghouse Topical Report WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event" (Reference 1), as accepted by the NRC in Reference 2. Additionally, the proposed changes are consistent with the changes previously approved by the NRC for the Watts Bar Nuclear Plant, Unit 1, as documented in Reference 3.

The proposed changes are described in detail in Section E of this Attachment. Attachments B-1 and B-2 include the marked-up TS page for the proposed changes for the Braidwood Station and the Byron Station, respectively. Attachments B-3 and B-4 include the associated TS page with the proposed changes incorporated for the Braidwood Station and the Byron Station, respectively. Attachments B-5 and B-6 include the associated TS Bases pages for information only with the proposed changes incorporated for the Braidwood Station and the Byron Station, respectively.

We request approval of the proposed changes prior to April 30, 2001, to allow the timely elimination of an unnecessary trip function and thereby reduce the potential for a transient. This would support development of needed procedure changes and required training to implement the proposed changes for all four Braidwood Station and Byron Station units in a timely manner.

B. DESCRIPTION OF THE CURRENT REQUIREMENTS

TS 3.3.1, Table 3.3.1-1, Function 3.b, requires all four NFRT channels to be Operable in Modes 1 and 2 (i.e., "Power Operation" and "Startup," respectively). With one required NFRT channel inoperable, TS 3.3.1, Condition E, requires that a known inoperable channel be placed in the tripped condition within six hours, or that the unit be placed in Mode 3 (i.e., "Hot Standby") within 12 hours. Placing the inoperable channel in the tripped condition results in a partial trip condition requiring only one-out-of-three logic for actuation of the two-out-of-four logic trips. If the inoperable channel cannot be placed in the trip condition within the specified six-hour Completion Time, the unit must be placed in a Mode where the NFRT Function is not required to be Operable. An additional six hours is allowed to place the unit in Mode 3. Six hours is a reasonable time, based on operating experience, to place the unit in Mode 3 from full power in an orderly manner and without challenging plant systems.

C. BASES FOR THE CURRENT REQUIREMENTS

TS 3.3.1, Table 3.3.1-1, Function 3.b, requires all four NFRT channels to be Operable in Modes 1 and 2, when there is a potential for a control rod(s) (i.e., Rod Cluster Control Assemblies (RCCAs)) drop accident to occur. The NFRT Function must be Operable in Modes 1 and 2 because the Updated Final Safety Analysis Report (UFSAR) Section 15.4.3, "Rod Cluster Control Assembly Misoperation (System Malfunction or Operator Error)," accident analysis identifies the NFRT Function as the primary means to mitigate RCCA misalignment accidents. In Modes 1 and 2 Departure from Nucleate Boiling (DNB) is a concern with RCCA misalignment accidents due to resulting skewed power distributions in the reactor core. In Modes 3, 4, 5, and 6 (i.e., "Hot Standby," "Hot Shutdown," "Cold Shutdown," and "Refueling," respectively), the NFRT Function does not need to be Operable because the reactor core is not critical and DNB is not a concern.

D. NEED FOR REVISION OF THE REQUIREMENT

The deletion of the NFRT Function eliminates an unnecessary trip function and thereby reduces the potential for a transient, which could challenge safe plant operation due to spurious trip signals. A 1982 evaluation prepared by Westinghouse Electric Corporation and documented in WCAP-10297-P-A, "Dropped Rod Methodology for Negative Flux Rate Trip Plants," Reference 4, determined that the NFRT Function was only required when a dropped RCCA or RCCA bank exceeded a specific reactivity worth threshold value. Any dropped RCCA or RCCA bank which had a reactivity worth below the threshold value would not require a reactor trip to maintain DNB limits. An additional evaluation, WCAP-11394-P-A (Reference 1), was performed by Westinghouse Electric Corporation in 1987, which determined that sufficient DNB margin existed for Westinghouse plant designs and fuel types without the NFRT Function regardless of the reactivity worth of the dropped RCCA or RCCA bank, subject to a plant/cycle-specific analysis. The NRC subsequently reviewed and approved the Westinghouse analysis and results and concluded that the analysis contains an acceptable procedure for analyzing the dropped RCCA event for which no credit is taken for any direct reactor trip due to the dropped RCCA(s) or for automatic power reduction due to the dropped RCCA(s). Therefore, the NFRT Function is not required to maintain existing DNB limits and may be eliminated at the Braidwood Station and the Byron Station.

On April 15, 2000, the Braidwood Station experienced an automatic reactor trip on Power Range Neutron Flux High Negative Rate due to the failure of a control rod stationary gripper fuse that caused one control rod to drop into the Unit 2 reactor core. This event was documented in Licensee Event Report (LER) Number 2000-002-00, "Automatic Reactor Trip on Power Range Neutron Flux High Negative Rate Due to Stationary Gripper Fuse FU15 Failure for Control Rod P10 Causing the Rod to Drop into the Core," dated May 12, 2000. Based on the UFSAR discussion, this NFRT Function provides protection against one or more dropped RCCAs. As discussed above, since the NFRT Function is not required to maintain existing DNB limits, and since no credit is taken for any direct reactor trip due to the dropped RCCA(s), deletion of this NFRT Function would eliminate the potential for an unnecessary transient.

E. DESCRIPTION OF THE PROPOSED CHANGES

The proposed changes revise Braidwood Station and the Byron Station TS 3.3.1, to delete the "Power Range Neutron Flux High Negative Rate" Trip Function 3.b from Table 3.3.1-1. The proposed changes are consistent with the Westinghouse methodology presented in WCAP-11394-P-A (Reference 1), as accepted by the NRC in Reference 2. Additionally, the proposed changes are consistent with the changes previously approved by the NRC for Watts Bar, Unit 1, as documented in Reference 3.

Because the "Power Range Neutron Flux High Negative Rate" Trip Function 3.b is being deleted from TS Table 3.3.1-1, the existing "Power Range Neutron Flux High Positive Rate" Trip Function 3.a is being re-designated as Function 3.

F. SAFETY ANALYSIS OF THE PROPOSED CHANGES

The original design basis for the NFRT Function was to mitigate the consequences of one or more dropped RCCAs. The intent was that in the event of one or more dropped RCCAs, the Reactor Trip System (RTS) would detect the rapidly decreasing neutron flux (i.e., high negative flux rate) due to the dropped RCCA(s) and would trip the reactor, thus ending the transient and assuring that DNB limits were maintained. In January 1982, Westinghouse Electric Corporation submitted Topical Report WCAP-10297-P, "Dropped Rod Methodology for Negative Flux Rate Trip Plants," to the NRC, which documented a new methodology for this event. This Topical Report concluded that the NFRT Function was only required when dropped RCCAs exceeded a specific reactivity worth threshold value. The threshold value was dependent upon plant design and fuel type. Dropped RCCAs, which had a reactivity worth below the threshold value, would not require a reactor trip to maintain DNB limits. The NRC approved this methodology as documented in Reference 5.

The Westinghouse Owners Group (WOG) subsequently submitted a new Topical Report, WCAP-11394-P, "Methodology for the Analysis of the Dropped Rod Event," to the NRC in 1987 for review and approval. The methodology used in this Topical Report is an extension of the NRC approved methodology used in WCAP-10297-P-A (Reference 4). The conclusion reached in WCAP-11394-P was that sufficient DNB margin is maintained with all Westinghouse plant designs and fuel types, such that the NFRT Function is not required regardless of the reactivity worth of the dropped RCCA(s). The use of this approach is required to be demonstrated using plant/cycle-specific analysis. The NRC approved this methodology as documented in Reference 2, and stated, "A further review by the staff (for each cycle) is not necessary, given the utility assertion that the analysis described by Westinghouse has been performed and the required comparisons have been made with favorable results."

WCAP-11394-P-A demonstrates that the DNB design basis is met during the course of the dropped RCCA transient, which considers one or more dropped RCCAs. No credit is taken for any direct reactor trip due to the dropped RCCA(s) or for automatic power reduction due to the dropped RCCA(s). Following this methodology, the operation of the Watts Bar Nuclear Plant, Unit 1, without the NFRT Function was found to be acceptable based on Tennessee Valley Authority's evaluation (Reference 6), as documented in the NRC Safety Evaluation (Reference 3).

The proposed changes for the Braidwood Station and the Byron Station are consistent with the NRC approved changes for the Watts Bar Nuclear Plant, Unit 1, and with the NRC approved methodology presented in WCAP-11394-P-A. This methodology assumes no direct reactor trip or automatic power reduction to mitigate the consequences of the dropped RCCA(s). The correlations and statepoints generated for this methodology apply to the Braidwood Station, Units 1 and 2, and the Byron Station, Units 1 and 2. Due to the plant-specific nature of the core physics characteristics and the thermal-hydraulic dropped rod limit lines, plant-specific data are combined with the appropriate set of correlations and statepoints to verify that the DNB design basis is met for the dropped RCCA(s) event for every fuel cycle design. Therefore, there is no adverse impact that increases the risk to the health and safety of the public as a result of the proposed changes. The following provides an assessment of the proposed changes with respect to other Braidwood Station and Byron Station safety analyses and evaluations.

Loss of Coolant Accident (LOCA) and LOCA-Related Evaluations

The NFRT Function is not modeled in the LOCA analyses. The following LOCA-related analyses are not affected by the proposed changes: large and small break LOCA, reactor vessel and Reactor Coolant System (RCS) loop LOCA blowdown forces, post-LOCA long term core cooling subcriticality, post-LOCA long term core cooling minimum flow, and RCS hot leg switchover to prevent boron precipitation. The proposed changes do not affect the normal plant operating parameters, accident mitigation capabilities important to a LOCA, the assumptions used in the LOCA-related accidents, or create conditions more limiting than those assumed in these analyses.

Non-LOCA Related Evaluation

Although the NFRT Function is addressed in the Braidwood Station and the Byron Station safety analyses, the current non-LOCA safety analyses do not take credit for the NFRT Function. Specifically, the dropped RCCA(s) analyses utilized for the current cycles do not rely on actuation of the NFRT Function to mitigate the consequences of the accident. These analyses were performed in accordance with the NRC approved methodology for the analysis of dropped RCCA(s) events provided in WCAP-11394-P-A. The analysis statepoints consider dropped RCCA worths up to 800 percent milli rho (pcm). The analysis assumptions and confirmation that the DNB design basis is met are further confirmed as part of the reload safety analysis for each reactor core reload. Currently, the reload safety analysis limits for all four Braidwood Station and Byron Station units' cycles include dropped RCCA statepoints with maximum dropped rod worth of 800 pcm. The reload safety analysis verifies the limiting dropped rod worth is less than 800 pcm. Therefore, the conclusion presented in the UFSAR that the DNB design basis is met with respect to non-LOCA related evaluations remains valid for the proposed changes which credit the application of WCAP-11394-P-A.

Mechanical Components and Systems Evaluation

Elimination of the NFRT Function as described above does not affect the RCS component integrity or the ability of the RCS to perform its intended safety function. The proposed changes do not affect the integrity of plant systems or their ability to perform intended safety functions.

Containment Integrity Evaluation (Short Term / Long Term LOCA Case)

The NFRT Function is not credited in the containment analyses. The proposed changes do not adversely affect the short term and long term LOCA mass and energy releases of the containment analyses. The proposed changes do not affect the normal plant operating parameters, system actuations, capabilities or assumptions important to the containment analyses, or create conditions more limiting than those assumed in these analyses. Therefore, the conclusions presented in the UFSAR remain valid with respect to the containment analyses.

Main Steam Line Break (MSLB) Mass and Energy Release Evaluation

The NFRT Function is not credited in the UFSAR MSLB analyses. The proposed changes do not adversely affect the MSLB mass and energy releases, either inside or outside containment, and do not adversely affect the calculations for the steam mass release used as input to the radiological dose evaluation. The proposed changes do not affect the normal plant operating parameters, input assumptions, results or conclusions of the MSLB mass and energy release analyses, and steam release calculations. Also, conditions are not created which are more limiting than those enveloped by the current analyses and calculations. Therefore, the conclusions presented in the UFSAR remain valid with respect to MSLB mass and energy release rates and steam mass release calculations.

Emergency Operating Procedures (EOPs) Evaluation

The proposed changes do not affect the EOPs. The NFRT Function is not covered as part of the EOPs and therefore the proposed changes have no impact.

Safety Systems Allowable Values and Setpoints Evaluation

The proposed changes do not affect the RTS or the Engineered Safety Feature Actuation System (ESFAS) Allowable Values or Setpoints. The NFRT Function deletion does not change the current Allowable Value information for any other function shown in the TS, and does not change the current Setpoint information for any other function shown in the Technical Requirements Manual (TRM). Therefore, the NFRT Function deletion has no impact on the plant safety functions.

Steam Generator Tube Rupture (SGTR) Evaluation

The NFRT Function is not credited in the SGTR analyses. The proposed changes do not adversely affect the normal plant operating parameters, results or conclusions of the SGTR thermal and hydraulic analyses. Also, conditions are not created which are more limiting than those enveloped by the current analyses for break flow and steam release. Therefore, the conclusions presented in the UFSAR remain valid with respect to the SGTR event.

Control Systems Evaluation

The proposed changes have no adverse impact on the control systems evaluation. The deletion of the NFRT Function could increase plant availability because the proposed changes eliminate a potential source of inadvertent reactor trips.

For each fuel cycle design, our Nuclear Generation Group Nuclear Fuel Management (NFM) administrative procedures will ensure that a dropped RCCA analysis is successfully performed in accordance with the methodology described in WCAP-11394-P-A. The NFRT Function is not credited in the current cycle-specific dropped RCCA analysis, and the current analysis and limits conform to WCAP-11394-P-A. The NFM "Reload Design Key Parameter Checklist" for all current operating cycles reflects this analysis and acceptance criteria.

We have reviewed the February 1998 multiple RCCA drop incident at the McGuire Station, Unit 1, documented in Significant Event Notification (SEN) 181. One of the significant aspects of that event involved a failure to initiate a manual reactor trip due in part to inadequate reactor trip criteria in the station's Abnormal Operating Procedure (AOP) for dropped RCCA events. The McGuire Station NFRT Function had been deleted several years earlier without revising the AOP. The current Braidwood Station and the Byron Station AOP, BwOA ROD-3 and BOA ROD-3, "Dropped or Misaligned Rod," respectively, for responding to a dropped RCCA event, instructs the plant operators to manually trip the reactor for multiple dropped RCCAs to prevent such occurrences. Lessons learned from the McGuire Station are discussed as part of our licensed operator requalification program and our initial license training program.

G. IMPACT ON PREVIOUS SUBMITTALS

We have reviewed the proposed changes regarding their impact on any previous submittals and have determined that there is no impact on any previous submittals. Upon approval and implementation of the pending Braidwood Station and the Byron Station Power Uprate amendment, the UFSAR safety analyses will be revised.

H. SCHEDULE REQUIREMENTS

We request approval of the proposed changes prior to April 30, 2001, to allow the timely elimination of an unnecessary protective function and thereby reduce the potential for a transient. This would support development of needed procedure changes and required training to implement the proposed changes for all four Braidwood Station and Byron Station units in a timely manner.

I. REFERENCES

1. Westinghouse Topical Report, WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event," dated January 1990.

2. Letter from A. C. Thadani (U. S. NRC) to R. A. Newton (Westinghouse Owners Group), "Acceptance for Referencing of Licensing Topical Reports WCAP-11394(P) and WCAP-

11395(NP), 'Methodology for the Analysis of the Dropped Rod Event,'" dated October 23, 1989.

3. Letter from R. E. Martin (U. S. NRC) to J. A. Scalice (Tennessee Valley Authority), "Issuance of Amendment Regarding Deletion of Negative Flux Rate Trip for Watts Bar Nuclear Plant, Unit 1," dated January 15, 1999.

4. Westinghouse Topical Report, WCAP-10297-P-A, "Dropped Rod Methodology for Negative Flux Rate Trip Plants," dated June 1983.

5. Letter from C. O. Thomas (U. S. NRC) to E. P. Rahe, Jr. (Westinghouse Electric Corporation), "Acceptance for Referencing of Licensing Topical Report WCAP-10297(P), WCAP-10298 - (NS-EPR-2545) entitled 'Dropped Rod Methodology for Negative Flux Rate Trip Plants,'" dated March 31, 1983.

6. Letter from P. L. Pace (Tennessee Valley Authority) to U. S. NRC, "Watts Bar Nuclear Plant (WBN) - Unit 1 - Technical Specification (TS) Change No. 98-006 - Deletion of Power Range Neutron Flux High Negative Rate Reactor Trip Function," dated June 26, 1998.

ATTACHMENT B-1

PROPOSED TS CHANGES FOR BRAIDWOOD STATION

MARKED-UP TS PAGE

3.3.1-14

Table 3.3.1-1 (page 1 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.13	NA
	$3^{(a)}$, $4^{(a)}$, $5^{(a)}$	2	C	SR 3.3.1.13	NA
2. Power Range Neutron Flux					
a. High	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.15	≤ 110.8% RTP
b. Low	$1^{(b)}$,2	4	E	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.15	≤ 27.0% RTP
3. Power Range Neutron Flux [redacted]					
[redacted] High Positive Rate	1,2	4	E	SR 3.3.1.7 SR 3.3.1.11	≤ 6.2% RTP with time constant ≥ 2 sec
[redacted]	[redacted]	[redacted]	E	[redacted]	[redacted]
4. Intermediate Range Neutron Flux	$1^{(b)}$, $2^{(c)}$	2	F,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 30.0% RTP
5. Source Range Neutron Flux	$2^{(d)}$	2	H,I	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.15	≤ 1.42 E5 cps
	$3^{(a)}$, $4^{(a)}$, $5^{(a)}$	2	I,J	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.15	≤ 1.42 E5 cps

(continued)

(a) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.

(b) Below the P-10 (Power Range Neutron Flux) interlock.

(c) Above the P-6 (Source Range Block Permissive) interlock.

(d) Below the P-6 (Source Range Block Permissive) interlock.

ATTACHMENT B-2

PROPOSED TS CHANGES FOR BYRON STATION

MARKED-UP TS PAGE

3.3.1-14

Table 3.3.1-1 (page 1 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.13	NA
	$3^{(a)}$, $4^{(a)}$, $5^{(a)}$	2	C	SR 3.3.1.13	NA
2. Power Range Neutron Flux					
a. High	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.15	≤ 110.8% RTP
b. Low	$1^{(b)}$,2	4	E	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.15	≤ 27.0% RTP
3. Power Range Neutron Flux [redacted]					
[redacted] High Positive Rate	1,2	4	E	SR 3.3.1.7 SR 3.3.1.11	≤ 6.2% RTP with time constant ≥ 2 sec
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
4. Intermediate Range Neutron Flux	$1^{(b)}$, $2^{(c)}$	2	F,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 30.0% RTP
5. Source Range Neutron Flux	$2^{(d)}$	2	H,I	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.15	≤ 1.42 E5 cps
	$3^{(a)}$, $4^{(a)}$, $5^{(a)}$	2	I,J	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.15	≤ 1.42 E5 cps

(continued)

(a) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.

(b) Below the P-10 (Power Range Neutron Flux) interlock.

(c) Above the P-6 (Source Range Block Permissive) interlock.

(d) Below the P-6 (Source Range Block Permissive) interlock.

ATTACHMENT B-3

PROPOSED TS CHANGES INCORPORATED FOR BRAIDWOOD STATION

TS PAGE

3.3.1-14

Table 3.3.1-1 (page 1 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.13	NA
	3(a), 4(a), 5(a)	2	C	SR 3.3.1.13	NA
2. Power Range Neutron Flux					
a. High	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.15	≤ 110.8% RTP
b. Low	1(b),2	4	E	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.15	≤ 27.0% RTP
3. Power Range Neutron Flux - High Positive Rate	1,2	4	E	SR 3.3.1.7 SR 3.3.1.11	≤ 6.2% RTP with time constant ≥ 2 sec
4. Intermediate Range Neutron Flux	1(b), 2(c)	2	F,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 30.0% RTP
5. Source Range Neutron Flux	2(d)	2	H,I	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.15	≤ 1.42 E5 cps
	3(a), 4(a), 5(a)	2	I,J	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.15	≤ 1.42 E5 cps

(continued)

(a) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.

(b) Below the P-10 (Power Range Neutron Flux) interlock.

(c) Above the P-6 (Source Range Block Permissive) interlock.

(d) Below the P-6 (Source Range Block Permissive) interlock.

ATTACHMENT B-4

PROPOSED TS CHANGES INCORPORATED FOR BYRON STATION

TS PAGE

3.3.1-14

Table 3.3.1-1 (page 1 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.13	NA
	3(a), 4(a), 5(a)	2	C	SR 3.3.1.13	NA
2. Power Range Neutron Flux					
a. High	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.15	≤ 110.8% RTP
b. Low	1(b),2	4	E	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.15	≤ 27.0% RTP
3. Power Range Neutron Flux – High Positive Rate	1,2	4	E	SR 3.3.1.7 SR 3.3.1.11	≤ 6.2% RTP with time constant ≥ 2 sec
4. Intermediate Range Neutron Flux	1(b), 2(c)	2	F,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 30.0% RTP
5. Source Range Neutron Flux	2(d)	2	H,I	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.15	≤ 1.42 E5 cps
	3(a), 4(a), 5(a)	2	I,J	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.15	≤ 1.42 E5 cps

(continued)

(a) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.

(b) Below the P-10 (Power Range Neutron Flux) interlock.

(c) Above the P-6 (Source Range Block Permissive) interlock.

(d) Below the P-6 (Source Range Block Permissive) interlock.

ATTACHMENT B-5

PROPOSED TS BASES CHANGES INCORPORATED FOR BRAIDWOOD STATION
FOR INFORMATION ONLY

TS BASES PAGES

B 3.3.1-11
B 3.3.1-41

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

In MODE 1, below the Power Range Neutron Flux (P-10 setpoint), and in MODE 2, the Power Range Neutron Flux-Low trip must be OPERABLE. This Function may be manually blocked by the operator when two out of four power range channels are greater than approximately 10% RTP (P-10 setpoint). This Function is automatically unblocked when three out of four power range channels are below the P-10 setpoint. Above the P-10 setpoint, positive reactivity additions are mitigated by the Power Range Neutron Flux-High trip Function.

In MODE 3, 4, 5, or 6, the Power Range Neutron Flux-Low trip Function does not have to be OPERABLE because the reactor is shut down and the NIS power range detectors cannot detect neutron levels in this range. Other RTS trip Functions and administrative controls provide protection against positive reactivity additions or power excursions in MODE 3, 4, 5, or 6.

3. Power Range Neutron Flux-High Positive Rate

The Power Range Neutron Flux-High Positive Rate trip uses the same channels as discussed for Function 2 above.

The Power Range Neutron Flux-High Positive Rate trip Function ensures that protection is provided against rapid increases in neutron flux that are characteristic of an RCCA drive rod housing rupture and the accompanying ejection of the RCCA. This Function compliments the Power Range Neutron Flux-High and Low Setpoint trip Functions to ensure that the criteria are met for a rod ejection from the power range.

The LCO requires all four of the Power Range Neutron Flux-High Positive Rate channels to be OPERABLE.

BASES

ACTIONS (continued)

E.1 and E.2

Condition E applies to the following reactor trip Functions:

- Power Range Neutron Flux - Low;
- Overtemperature ΔT;
- Overpower ΔT;
- Power Range Neutron Flux - High Positive Rate;
- Pressurizer Pressure - High; and
- SG Water Level - Low Low.

A known inoperable channel must be placed in the tripped condition within 6 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one-out-of-three logic for actuation of the two-out-of-four trips. The 6 hours allowed to place the inoperable channel in the tripped condition is justified in Reference 7.

If the inoperable channel cannot be placed in the trip condition within the specified Completion Time, the unit must be placed in a MODE where these Functions are not required OPERABLE. An additional 6 hours is allowed to place the unit in MODE 3. Six hours is a reasonable time, based on operating experience, to place the unit in MODE 3 from full power in an orderly manner and without challenging plant systems.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The 4 hour time limit is justified in Reference 7.

ATTACHMENT B-6

PROPOSED TS BASES CHANGES INCORPORATED FOR BYRON STATION
FOR INFORMATION ONLY

TS BASES PAGES

B 3.3.1-11
B 3.3.1-41

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

In MODE 1, below the Power Range Neutron Flux (P-10 setpoint), and in MODE 2, the Power Range Neutron Flux-Low trip must be OPERABLE. This Function may be manually blocked by the operator when two out of four power range channels are greater than approximately 10% RTP (P-10 setpoint). This Function is automatically unblocked when three out of four power range channels are below the P-10 setpoint. Above the P-10 setpoint, positive reactivity additions are mitigated by the Power Range Neutron Flux-High trip Function.

In MODE 3, 4, 5, or 6, the Power Range Neutron Flux-Low trip Function does not have to be OPERABLE because the reactor is shut down and the NIS power range detectors cannot detect neutron levels in this range. Other RTS trip Functions and administrative controls provide protection against positive reactivity additions or power excursions in MODE 3, 4, 5, or 6.

3. <u>Power Range Neutron Flux-High Positive Rate</u>

The Power Range Neutron Flux-High Positive Rate trip uses the same channels as discussed for Function 2 above.

The Power Range Neutron Flux-High Positive Rate trip Function ensures that protection is provided against rapid increases in neutron flux that are characteristic of an RCCA drive rod housing rupture and the accompanying ejection of the RCCA. This Function compliments the Power Range Neutron Flux-High and Low Setpoint trip Functions to ensure that the criteria are met for a rod ejection from the power range.

The LCO requires all four of the Power Range Neutron Flux-High Positive Rate channels to be OPERABLE.

BASES

ACTIONS (continued)

E.1 and E.2

Condition E applies to the following reactor trip Functions:

- Power Range Neutron Flux – Low;
- Overtemperature ΔT;
- Overpower ΔT;
- Power Range Neutron Flux – High Positive Rate;
- Pressurizer Pressure – High; and
- SG Water Level – Low Low.

A known inoperable channel must be placed in the tripped condition within 6 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one-out-of-three logic for actuation of the two-out-of-four trips. The 6 hours allowed to place the inoperable channel in the tripped condition is justified in Reference 7.

If the inoperable channel cannot be placed in the trip condition within the specified Completion Time, the unit must be placed in a MODE where these Functions are not required OPERABLE. An additional 6 hours is allowed to place the unit in MODE 3. Six hours is a reasonable time, based on operating experience, to place the unit in MODE 3 from full power in an orderly manner and without challenging plant systems.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The 4 hour time limit is justified in Reference 7.

ATTACHMENT C

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:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

Commonwealth Edison (ComEd) Company is proposing changes to the Technical Specifications (TS) of Facility Operating License Nos. NPF-72, NPF-77, NPF-37 and NPF-66, for the Braidwood Station, Units 1 and 2, and the Byron Station, Units 1 and 2, respectively. The proposed changes revise TS 3.3.1,"Reactor Trip System (RTS) Instrumentation," to delete the "Power Range Neutron Flux High Negative Rate" Trip Function 3.b from Table 3.3.1-1, "Reactor Trip System Instrumentation." The proposed changes are consistent with the NRC approved methodology presented in the Westinghouse Topical Report WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event," dated January 1990.

Information supporting the determination that the criteria set forth in 10 CFR 50.92 are met for this amendment request is indicated below.

1. **Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The removal of the Power Range Neutron Flux High Negative Rate Trip (i.e., Negative Flux Rate Trip (NFRT)) Function does not increase the probability or consequences of reactor core damage accidents resulting from dropped Rod Cluster Control Assembly (RCCA) events previously analyzed. The safety functions of other safety related systems and components, which are related to mitigation of these events, have not been altered. All other primary Reactor Trip System (RTS) and Engineered Safety Features Actuation Systems (ESFAS) protection functions are not impacted by the elimination of the NFRT Function. The NFRT circuitry detects and responds to negative reactivity insertion due to RCCA misoperation events should they occur. Therefore, the NFRT Function is not assumed in the initiation of such events. Because the NFRT Function is being eliminated from the plant, it can no longer actuate and cause a transient. The consequences of accidents previously evaluated in the Updated Final Safety Analysis Report (UFSAR) are unaffected by the proposed changes because no change to any equipment response or accident mitigation scenario has resulted.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?**

The deletion of the NFRT Function does not create the possibility of a new or different kind of accident than any accident previously evaluated in the UFSAR. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. The proposed changes do not challenge the performance or integrity of any safety related systems. It has been demonstrated that the NFRT Function can be eliminated by the NRC approved methodology described in Westinghouse Topical Report WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event," dated January 1990. The Braidwood Station and the Byron Station cycle-specific analyses have confirmed that for a dropped RCCA(s) event, no direct reactor trip or automatic power reduction is required to meet the Departure From Nucleate Boiling (DNB) limits for this Condition II, "Faults of Moderate Frequency," event. The NFRT Function is not credited either as a primary or backup mitigation feature for any other UFSAR event. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Do the proposed changes involve a significant reduction in a margin of safety?**

The margin of safety associated with the licensing basis acceptance criteria for any postulated accident is unchanged. It has been demonstrated that the NFRT Function can be eliminated by the NRC approved methodology described in WCAP 11394-P-A. The Braidwood Station and the Byron Station cycle-specific analyses have confirmed that for a dropped RCCA(s) event, DNB limits are not exceeded with the proposed changes. Conformance to our licensing basis acceptance criteria for Design Basis Accidents (DBAs) and transients with the deletion of the NFRT Function is demonstrated, and DNB limits are not exceeded. The proposed changes will have no adverse effect on the availability, operability, or performance of the safety related systems and components assumed to actuate in the event of a DBA or transient. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

ATTACHMENT D

INFORMATION SUPPORTING AN ENVIRONMENTAL ASSESSMENT

Commonwealth Edison (ComEd) Company has evaluated the 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." ComEd has determined that the proposed changes meet the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9), "Criteria 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 which changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria.

(i) The amendment involves no significant hazards consideration.

As demonstrated in Attachment C, the proposed changes do not involve any significant hazards consideration.

(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 are limited to deletion of the Power Range Neutron Flux High Negative Rate Trip Function from Technical Specification 3.3.1, "Reactor Trip System (RTS) Instrumentation." No new radiological analyses are required. The proposed 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 by-products. Therefore, the proposed changes do not affect actual unit effluents. Therefore, the proposed changes do not change the types or increase the amounts of any effluents released offsite.

(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 of radioactive effluents or handling of solid radioactive waste, nor will the proposal 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 the proposed changes.