Commonwealth Edison
1400 Opus Place
Downers Grove, Illinois 60515

March 23, 1994

William T. Russell, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Document Control Desk

Subject: Application for Amendment to Facility Operating Licenses:

Byron Station Units 1 and 2
(NPF-37/66; NRC Docket Nos. 50-454/455)

Braidwood Station Units 1 and 2
(NPF-72/77; NRC Docket Nos. 50-456/457)

"Positive Moderator Temperature Coefficient and Reduced Thermal Design Flow"

References: 1. WCAP 13964 Revision 1, "Commonwelath Edison Company; Byron and Braidwood Units 1 and 2; Increased SGTP/Reduced TDF/PMTC Analysis Program; Engineering/Licensing Report"

2. Operating Parameter Uncertainties for the Byron/Braidwood Revised Thermal Design Procedure

Dear Mr. Russell:

Pursuant to 10 CFR 50.90, Commonwealth Edison Company (CECo) proposes to amend Appendix A, Technical Specifications of Facility Operating Licenses NPF-37, NPF-66, NPF-72, and NPF-77. The proposed amendment request addresses Technical Specification changes necessary tc support fuel cycles designed with a positive moderator temperature coefficient. The subject amendment request also proposes to change the core thermal design flow requirements to support an increase in the steam generator tube plugging limit to 15%.

A detailed description of the proposed changes is presented in Attachment 1. The revised Technical Specification pages are contained in Attachment 2.

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change : NRC PDR Ltr. Encl. 1 INP
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The proposed changes have been reviewed and approved by the On-site and Off-site Review Committees in accordance with CECo procedures. CECo has reviewed this proposed amendment in accordance with 10 CFR 50.92(c) and has determined that no significant hazards consideration exists as documented in Attachment 3. An Environmental Assessment has also been completed and is contained in Attachment 4.

Attachment 6 contains Westinghouse Proprietary Report, "Operating Parameter Uncertainties for the Byron/Braidwood Revised Thermal Design Procedure". Attachment 6 also contains the associated Westinghouse authorization letter, CAW-94-586, accompanying affidavit, and Proprietary Information Notice. As this report contains information proprietary to Westinghouse Electric Corporation, it is supported by the previously mentioned affidavit, signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.790 of the Commission's regulations. Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.790 of the Commission's regulations.

Correspondence with respect to the proprietary aspects of the items listed above or the supporting Westinghouse Affidavit should reference CAW-94-586 and should be addressed to N. J. Liparulo, Manager of Nuclear Safety & Regulatory Activities, Westinghouse Electric Corporation, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

CECo is notifying the State of Illinois of our application for these amendments by transmitting a copy of this letter and the associated attachments to the designated State Official.

Commonwealth Edison Company respectfully requests that this proposed amendment be reviewed and approved in time to support implementation of the changes during the next Byron Unit 1 refueling outage, scheduled to begin on September 9, 1994.

To the best of my knowledge and belief, the statements contained in this document are true and correct. In some respects these statements are not based on my personal knowledge, but on information furnished by other CECo employees, contractor employees, and/or consultants. Such information has been reviewed in accordance with company practice, and I believe it to be reliable.

Please address any comments or questions regarding this matter to this office.

Respectfully,

Joseph A. Bauer

Joseph A. Bauer
Nuclear Licensing Administrator

JAB/gp

Attachments

cc: G. F. Dick, Byron Project Manager - NRR
R. R. Assa, Braidwood Project Manager - NRR
H. Peterson, SRI - Byron
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Maryellen D. Long
3-23-94

OFFICIAL SEAL
MARYELLEN D LONG
NOTARY PUBLIC, STATE OF ILLINOIS
MY COMMISSION EXPIRES 09/01/97

PROPOSED LICENSE AMENDMENT

"POSITIVE MODERATOR TEMPERATURE COEFFICIENT
AND
REDUCED THERMAL DESIGN FLOW"

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Attachment 8	CECo I & C Engineering Letter

ATTACHMENT 1
DETAILED DESCRIPTION

1. Moderator Temperature Coefficient

Description of the Current Operating Requirement:

Technical Specification 3.1.1.3, Moderator Temperature Coefficient, requires the moderator temperature coefficient (MTC) to be less positive than 0 $\Delta k/k/^{\circ}F$ for the all rods withdrawn, hot zero thermal power condition or less negative than -4.1×10^{-4} $\Delta k/k/^{\circ}F$ for the all rods withdrawn, end of cycle life, rated thermal power condition.

Basis for the Current Operating Requirement:

The limits on MTC are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the Updated Final Safety Analysis Report (UFSAR) accident and transient analysis.

Description of the Need for Amending the OL Requirement:

This change would allow a core to be designed with a positive MTC. The primary benefits of a positive MTC are:

1) the reduced burnable poison rods required to control peaking during the early portion of each cycle,

2) reduced burnable poison handling requirements,

3) fewer problems associated with storage and disposal of spent burnable poisons,

4) a reduced probability of enforcing administrative control rod withdrawal limits at low power which improves operational flexibility,

5) increased fuel discharge burnups, and

6) significant fuel cost savings.

Description of the Amended OL Requirement:

The proposed change to Technical Specification 3.1.1.3.a allows a $+7.0 \times 10^{-5}$ $\Delta k/k/^{\circ}F$ MTC below 70% of rated thermal power, with a linear ramp to 0 $\Delta k/k/^{\circ}F$ at 100% rated thermal power. A new figure, Figure 3.1-0 "Moderator Temperature Coefficient vs. Power Level", also identifies the acceptable operating range for MTC.

An administrative change is requested to the MTC Technical Specification which changes the "or" to "and" in the Limiting Condition for Operation between Specifications 3.1.1.3a and 3.1.1.3b.

Basis for the Amended Operating Requirement:

Operation of the plant with a slightly positive MTC satisfies all design criteria and results in significant benefits. The basis for the MTC limit is to ensure that the value of the coefficient remains within the limits assumed in the Updated Final Safety Analysis Report (UFSAR) accident and transient analyses. In keeping with this basis, Westinghouse performed the necessary accident and transient analyses with the new MTC values to ensure that the results remain within all design and safety criteria. The Westinghouse analysis provides the basis for the proposed MTC Technical Specification change. This analysis, entitled "Byron and Braidwood Units 1 and 2 Increased SGTP/Reduced TDF/PMTC Analysis Engineering and Licensing Report" is presented in Attachment 5.

A power level dependent MTC was chosen to minimize the effects of the Technical Specification change on postulated accidents at higher power levels. Moreover, as the power level is raised, the average coolant temperature becomes higher as allowed by the programmed average temperature controller, tending to make the MTC more negative. Also, the boron concentration can be reduced as xenon builds into the core. Thus, there is less need to allow a positive MTC as full power is approached. As fuel burnup is achieved, boron is further reduced and the MTC will eventually become negative over the entire operating power range.

In addition to a positive reactivity feedback with increasing core average temperature as a result of a positive MTC, there is a negative reactivity feedback with increasing fuel temperature as a result of the fuel temperature coefficient which is always negative. The fuel average temperature increase over the operating power range is significantly greater than the core average temperature increase. Thus the cumulative reactivity feedback as core power approaches 100% is always negative.

While Anticipated Transient Without Scram (ATWS) is not a design basis accident, an evaluation will be done each fuel cycle by CECo's Nuclear Fuel Services department to verify that the risk associated with an ATWS while operating with a positive MTC is acceptable.

Revision of the "or" to "and" between Specification 3.1.1.3a and 3.1.1.3b is requested to ensure that it is clear that both Specifications are required to be met over the fuel cycle.

2. Shutdown Margin Requirements

Description of the Current Operating Requirement:

Technical Specification Sections 3.1, 3.5 and 3.9 contain boron concentration requirements for the Refueling Water Storage Tank (RWST), the Reactor Coolant System (RCS) accumulators and the refueling cavity. These requirements are as follows:

a. Technical Specification 3.1.2.5.b, Borated Water Source - Shutdown, currently requires a minimum boron concentration of 2000 ppm in the RWST during Modes 5 and 6.

b. Technical Specification 3.1.2.6.b, Borated Water Source - Operating, currently requires a minimum boron concentration of 2000 ppm in the RWST during Modes 1 through 4.

c. Technical Specification 3.5.1, Accumulators, currently requires an accumulator boron concentration of between 1900 and 2100 ppm during Modes 1 through 3.

d. Technical Specification 3.5.5, Refueling Water Storage Tank, currently requires a minimum RWST boron concentration of 2000 ppm during Modes 1 through 4.

e. Technical Specification 3.9.1, Boron Concentration, currently requires a minimum boron concentration in the refueling cavity of 2000 ppm during Mode 6.

f. Technical Specification Bases 3/4.1.2, Boration Systems, states that with the RCS average temperature above 350°F, the maximum expected boration capability requirement occurs at end of core life (EOL) from full power equilibrium xenon conditions and requires 15,780 gallons of 7000 ppm borated water from the boric acid storage tanks or 70,450 gallons of 2000 ppm borated water from the RWST. Also, the boron capability requirement below 200°F to provide sufficient shutdown margin is either 2,652 gallons of 7000 ppm borated water from the boric acid storage tanks or 11,840 gallons of 2000 ppm borated water from the RWST.

g. Technical Specification Bases 3/4.5.5, Refueling Water Storage Tank, states that the RWST pH will remain between 8.5 and 11.0 for the solution recirculated within containment after a Loss of Coolant Accident (LOCA).

h. Technical Specification Bases 3/4.6.2.2, Spray Additive System, states that the limits on the spray additive system ensures the pH will be between 8.5 and 11.0 for the solution recirculated within containment after a LOCA.

Basis for the Current Operating Requirement:

The Emergency Core Cooling System (ECCS) boration capabilities are supplied by the RWST and the accumulators. The boron supplied by these sources provides negative reactivity to shutdown the core for both LOCA and non-LOCA transients. In order to provide the necessary shutdown margin requirements, the RWST boron concentration is required to be a minimum of 2000 ppm.

<u>Boration Systems</u>
With the RCS average temperature above 350°F, the boration capability shall be sufficient to provide a shutdown margin from expected operating conditions of 1.3% Δk/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at end of core life (EOL) from full power equilibrium xenon conditions and requires 15,780 gallons of 7000 ppm borated water from the boric acid storage tanks or 70,450 gallons of 2000 ppm borated water from the RWST. Both boron injection flow paths are required to be operable.

With the RCS temperature below 350°F, one boron injection system (as discussed above) is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting core alterations and positive reactivity changes in the event the single boron injection system becomes inoperable.

The boron capability required below 200°F is sufficient to provide shutdown margin of 1% Δk/k after xenon decay and cooldown from 200°F to 140°F. This condition requires either 2,652 gallons of 7000 ppm borated water from the boric acid storage tanks or 11,840 gallons of 2000 ppm borated water from the RWST.

<u>Accumulators</u>
The operability of each RCS accumulator ensures that a sufficient volume of borated water will be immediately forced into the core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met. A contained borated water level between 31% and 63% ensures a volume of greater than or equal to 6995 gallons but less than or equal to 7217 gallons.

<u>RWST</u>
The operability of the RWST as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that: (1) sufficient

water is available within containment to permit recirculation cooling flow to the core, and (2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except of the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 11.0 for the solution recirculated within the containment after a LOCA. This pH band minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

Spray Additive System

The operability of the spray additive system ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH volume and concentration ensure a pH value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The contained solution volume limit includes an allowance for solution not usable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the safety analyses. A spray additive tank level of between 78.6% and 90.3% ensures a volume of greater than or equal to 4000 gallons but less than or equal to 4540 gallons.

Refueling Operations Boron Concentration

The limitations on reactivity conditions during refueling ensure that:
(1) the reactor will remain subcritical during core alterations, and
(2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel.

The limitation on K_{eff} of no greater than 0.95 is sufficient to prevent reactor criticality during refueling operations and includes a 1% $\Delta k/k$ conservative allowance for uncertainties. Similarly, the boron concentration value of 2000 ppm or greater includes a conservative uncertainty allowance of 50 ppm. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses.

Description of the Need for Amending the OL Requirement:

This Technical Specification change reflects the values used in the design and safety analysis to implement a positive MTC. The minimum RWST and refueling cavity boron concentrations were increased to 2300 ppm to maintain adequate shutdown margin throughout the cycle particularly due to increased cycle energies. The increased minimum accumulator boron concentration of 2200 also ensures adequate shutdown margin while allowing for small accumulator volume control operations using makeup water or small inadvertent dilutions. The maximum boron concentrations of 2500 ppm and 2400 ppm for the RWST and accumulators respectively were added to ensure that boron precipitation would not occur prior to hot leg switchover during a LOCA.

Description of the Amended OL Requirement:

The revision will change the minimum boron concentration of the RWST to 2300 ppm and will increase the accumulator boron concentration range to between 2200 and 2400 ppm. In addition, the refueling cavity boron concentration requirement will be increased to a minimum of 2300 ppm to ensure adequate shutdown margin.

With the increased boron concentration and the RCS average temperature above 350°F, the maximum expected boration capability requirement has changed to 13,487 gallons of 7000 ppm borated water from the boric acid storage tanks or 54,014 gallons of 2300 ppm borated water from the RWST. With the RCS below 200°F, the boration requirement has been reanalyzed to require 740 gallons of 7000 ppm borated water from the boric acid storage tanks or 2264 gallons of 2300 ppm borated water from the RWST to provide the adequate shutdown margin.

A change to the pH value for the solution recirculated within containment after a LOCA from between 8.5 and 11.0 to between 8.0 and 11.0 is requested to accommodate the increased boron concentrations in the RWST and accumulators.

Basis for the Amended Operating Requirement:

The accidents which are sensitive to positive or near zero MTC were reanalyzed to ensure all acceptance criteria were met. The increased boron concentrations were included in these events to ensure adequate shutdown margin is maintained at all times.

The maximum boron concentrations were verified to meet post-LOCA minimum sump pH requirements. A recalculation of the minimum sump pH was performed. LOCA radiological consequences and Post-LOCA hydrogen production and control were evaluated for the minimum pH of 8.0 and demonstrated that the Elemental Iodine Decontamination Factor of 100 would be met. The evaluation verified that the UFSAR and SER LOCA dose analyses remained valid. Although a very small increase in hydrogen production would exist post-LOCA, there would be no adverse impact on the time at which hydrogen recombiner operation is required or on the capability of the recombiner to control hydrogen.

3. Thermal Design Flow Reduction

Description of the Current Operating Requirement:

Technical Specification Sections 2.1, 2.2 and 3.2.3 contain requirements based on the thermal design flow of the RCS. These requirements are as follows:

a. Technical Specification 2.1, Safety Limits, states that the combination of thermal power, pressurizer pressure and the highest loop Tavg cannot exceed the limits shown in Figure 2.1-1. This graph is based on the reactor coolant system flow.

b. Technical Specification Table 2.2-1, Reactor Trip System Instrumentation Trip Setpoints, denotes the minimum measured RCS flow as being 97,600 gpm.

c. Technical Specification Table 2.2-1, Note 1, Overtemperature ΔT, specifies the constants used in the Overtemperature ΔT trip setpoint equation.

d. Technical Specification Table 2.2-1, Note 2 states the maximum deviation of the Overtemperature ΔT trip setpoint from its computed setpoint is 3.71% of ΔT span.

e. Technical Specification Table 2.2-1, Note 3, Overpower ΔT, specifies the constants used in the Overpower ΔT trip setpoint equation.

f. Technical Specification Table 2.2-1, Note 4 states the maximum deviation of the Overpower ΔT trip setpoint from is computed setpoint is 2.31% of ΔT span.

g. Technical Specification Bases 2.1.1 provides the design and safety analysis DNBR values for a typical cell and a thimble cell.

h. Technical Specification 3.2.3 requires the RCS total flowrate to be $\geq$ 390,400 gpm.

Basis for the Current Operating Requirement:

The current RCS flowrate requirement of $\geq$ 390,400 gpm is based on the loop minimum measured flow of 97,600 gpm which is used in the Improved Thermal Design Procedure described in UFSAR Sections 4.4.1 and 15.0.3.

The basis for the requirements of Table 2.2-1 is to prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature. Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation. A correlation exists between DNB and the combination of thermal power and reactor coolant temperature and pressure. This relation has been used to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, and is indicative of the margin to DNB.

In meeting this design basis, uncertainties in plant operating parameters, nuclear and thermal parameters and fuel fabrication parameters are considered statistically such that there is at least a 95% confidence that the minimum DNBR for the limiting rods is greater than or equal to the DNBR limit. The uncertainties in the above plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit, establishes a design DNBR value which must be met in plant safety analysis using values of the input parameters without uncertainties.

The design DNBR values are 1.34 and 1.32 for a typical cell and a thimble cell, respectively for OFA fuel, and 1.33 for a typical cell and 1.32 for a thimble cell for the VANTAGE 5 fuel. In addition, margin has been maintained in both designs by meeting safety analysis DNBR limits of 1.49 for a typical cell and 1.47 for a thimble cell for OFA fuel, and 1.67 and 1.65 for a typical cell and a thimble cell, respectively for the VANTAGE 5 fuel in performing safety analyses.

The curves of Figure 2.1-1 show the loci of points of thermal power, reactor coolant system pressure and average temperature for which the minimum design DNBR is no less than the design DNBR value, or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid. This curve is based on an assumption of reactor coolant system flow.

Both the Overtemperature and Overpower ΔT trip setpoint constants were calculated utilizing the improved thermal design procedure.

Description of the Need for Amending the OL Requirement:

The purpose of the reduction in the Technical Specification required RCS flow rate is to provide a greater margin between the actual flow rate and the minimum measured flow. This margin will then be able to offset any reduction in flow as a result of steam generator tube plugging.

Description of the Amended OL Requirement:

Figure 2.1-1 is being revised to reflect the lower RCS flow values used in the analysis. A new Figure 2.1-1 is provided in the marked up Technical Specifications (Attachment 2).

In Technical Specification 3.2.3, the RCS total flowrate is revised to 371,400 gpm. The Minimum Measured Flow is changed to 92,850 gpm per loop as listed in Table 2.2-1, Reactor Trip System Instrumentation Trip Setpoints.

The Overtemperature ΔT constants K_1, K_2 and K_3 are being revised to 1.3250, 0.0297/°F and 0.00181, respectively. The error allowance on Overtemperature ΔT is being changed from 3.71% of span to 1.16% of span. The $f_1(\Delta I)$ band is now -24% to +10% for a zero penalty with a positive slope of 4.11% and a negative slope of 3.35%. The Overpower ΔT constant K_6 is changed to 0.00245/°F. The error allowance on Overpower ΔT is being changed from 2.31% of span to 3.08% of span. The I & C Engineering letter on the justification of the change to the Overtemperature ΔT constant K_1 and the error allowance is included in Attachment 8.

The bases for Technical Specification 2.1.1 have been revised to reflect the DNBR values. For VANTAGE 5 fuel, the design values are 1.25 for both a typical cell and a thimble cell. The safety analysis DNBR values are 1.50 for a typical cell and a thimble cell.

Basis for the Amended Operating Requirement:

The core thermal limits have been revised to reflect the utilization of the Revised Thermal Design Procedure (RTDP) DNB methodology and the requested reduction in RCS flow. Commonwealth Edison Company performed the statistical calculations to determine the operating parameter uncertainty inputs utilized in the Revised Thermal Design Procedure. These uncertainty values were then used by Westinghouse to determine the DNBR values and the core limits using NRC approved codes as discussed in WCAP 13964, Section 7.2.

RTDP provides a methodology for accounting for instrument uncertainties to predict plant DNBR limits. The RTDP considers four operating parameters, pressurizer pressure, primary coolant average temperature (T_{avg}), reactor power and RCS flow in the uncertainty analysis of the thermal design procedure. In general, the RTDP methodology considers instrument uncertainties to be random, normal, two sided probability distributions. The methodology combines the error components for a channel using the square root of the sum of the squares for error terms considered statistically independent. Errors that are considered dependent are combined arithmetically into independent groups and then systematically combined. The instrument uncertainty methodology used by CECo is consistent with the methodology used in WCAP 12801, "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology for Commonwealth Edison Zion Units 1 and 2 Nuclear Power Station". WCAP 12801 was submitted as a part of Zion's license amendment to allow use of VANTAGE 5 fuel and conversion to the EAGLE-21 process protection system hardware. The NRC Staff found this methodology acceptable as documented in the NRC's Safety Evaluation related to Zion Station Amendments No. 139 (Unit 1) and No. 128 (Unit 2) dated June 26, 1992.

The RTDP instrument uncertainty methodology is also consistent with the methodology CECo used to perform the Byron/Braidwood protection system setpoint uncertainty calculations. The protection setpoint uncertainty methodology was based on the methodology presented in WCAP 12583, "Westinghouse Setpoint Methodology for Protection Systems, May 1990". This was submitted as a part of a Byron/ Braidwood license amendment addressing setpoint reconciliation of the Reactor Trip System and Engineered Safety Features Actuation System instrumentation. The NRC Staff found this methodology acceptable as documented in the Safety Evaluation related to Byron Station Amendment No. 53 and Braidwood Station Amendment No. 42, dated April 13, 1993. The proprietary and non-proprietary versions of the RTDP are provided in Attachments 6 and 7, respectively.

The use of the RTDP methodology allows the stations to gain margin to trip under normal operating conditions for the Overtemperature ΔT trip setpoint constants. The recalculations also required revision of the channel's maximum trip setpoint deviation from the computed trip setpoint for both Overtemperature ΔT and Overpower ΔT due to the recalculations. Non-LOCA transients were analyzed using the RTDP uncertainties and verified that adequate margin exists to the new DNBR design limits.

The new constants for the OTΔT and OPΔT are identified in Table 3.0-1, page 3-15 of WCAP 13964, Attachment 5, and in the Technical Specification Table 2.2-1 markups in Attachment 2. The uncertainties for the constant values were calculated by Commonwealth Edison's Nuclear Engineering and Technology Services I&C department. These constants continue to protect the core thermal limits and prevent fuel center-line melting. The values allowed by Technical Specification Table 2.2-1 are the nominal setpoint values that include an allowance for instrument uncertainties and are more conservative than the maximum values used in the safety analysis.

It is worthy to note that WCAP 13964 contains three discrepancies related to the new constants. Table 3.0-1 incorrectly identified the $K1_{NOM}$ parameter as $K1_{MAX}$ and $K4_{NOM}$ as the $K4_{MAX}$ parameter. This discrepancy is administrative in nature and in no way affects the results of the safety analysis. In addition, during the calculation of the revised OTΔT setpoint, the change in the f(ΔI) penalty was not properly accounted for. As a result, the K1 value of 1.37, noted in Table 3.0-1, is incorrect. The correct value for K1 is 1.325. Since the corrected $K1_{NOM}$ value remains more conservative than the $K1_{MAX}$ constant validated in the accident analysis, the OTΔT setpoint calculation continues to be bounded by the analysis. The calculation of these values are discussed in Attachment 8.

The implementation of the new setpoint constants will not be affected by the error in the WCAP. The actual setpoint installation is conducted under Station approved procedures, consistent with the Technical Specifications, and do not reference the WCAP.

The RCS flow reduction reflects a conservatively low RCS flowrate so that flow margin is available over that assumed in the analysis of the systems and components of the NSSS, the LOCA and non-LOCA transients and the fuel. This margin can be used to accommodate reductions in RCS flow such as with increased steam generator tube plugging. The safety analyses verified that the units can operate safely at a reduced RCS flow.

The maximum RCS flow measurement uncertainty for Byron and Braidwood Units 1 and 2 has been increased to 3.5% from the 2.2%. This value of flow measurement uncertainty, identified in the bases, is a conservative value that has been endorsed by the NRC for use in plants that do not have a plant-specific calculation of flow measurement uncertainty. This change is discussed in the RTDP instrument uncertainty methodology provided in Attachments 6 and 7.

There are administrative changes to index page IV and the bases page 3/4 1-3. The changes identify the addition of Figure 3.1-0 "Moderator Temperature Coefficient vs. Power Level" and in the bases page, the addition of the word "relief" between Suction and valve in the last sentence of the third paragraph for clarification.

Schedule Requirements:

Commonwealth Edison requests that the review and approval of the proposed amendment be completed as soon as possible, to support the implementation during the Byron Unit 1 refueling outage, scheduled to begin on September 9, 1994.