ATTACHMENT B

DESCRIPTION AND SAFETY ANALYSIS OF
PROPOSED CHANGES TO
APPENDIX A TECHNICAL SPECIFICATIONS OF
FACILITY OPERATING LICENSES
NPF-37, NPF-66, NPF-72, AND NPF-77

BYRON UNITS 1 & 2	BRAIDWOOD UNITS 1 & 2
Revised Pages	Revised Pages
IV	IV
3/4 1-4	3/4 1-4
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3/4 1-5a	3/4 1-5a
B 3/4 1-1	B 3/4 1-1
B 3/4 1-2	B 3/4 1-2
6-22	6-22

k:nla\byrbwd\pmtcsup\11

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 ~~The moderator temperature coefficient (MTC) shall be:~~

INSERT A

a. ~~Less positive than 0 Δk/k/°F for the all rods withdrawn, hot zero THERMAL POWER condition, or~~

b. ~~Less negative than -4.1×10^{-4} Δk/k/°F for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.~~

~~APPLICABILITY: Specification 3.1.1.3a. - MODES 1 and 2* only#.~~
~~Specification 3.1.1.3b. - MODES 1, 2, and 3 only#.~~

ACTION:

a. With the MTC more positive than the BOL limit specified in the OLR, ~~of Specification 3.1.1.3a. above,~~ operation in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than the BOL limit specified in the OLR ~~0 Δk/k/°F~~ within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6;

2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition; and

3. A Special Report is prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

4. The provisions of Specification ~~3.0.4~~ are not applicable.

b. With the MTC more negative than the EOL limit specified in the OLR, ~~of Specification 3.1.1.3b. above,~~ be in HOT SHUTDOWN within 12 hours.

*With K_{eff} greater than or equal to 1.

#See Special Test Exceptions Specification 3.10.3.

INSERT A

3.1.1.3 The moderator temperature coefficient (MTC) shall be within the limits specified in the Operating Limits Report (OLR). The maximum upper limit shall be less than or equal to that shown in Figure 3.1-0.

<u>APPLICABILITY</u>: Beginning of Life (BOL) limit - MODES 1 and 2* only#
End of Life (EOL) limit - MODES 1, 2, and 3 only#.

k:nla\byrbwd\pmtcsup\12

SURVEILLANCE REQUIREMENTS

4.1.1.3 The MTC shall be determined to be within its limits during each fuel cycle as follows:

BOL

a. The MTC shall be measured and compared to the predicted MTC to establish administrative rod withdrawal limits, as necessary, to assure that the limit ~~of Specification 3.1.1.3a., above~~, is met throughout core life, prior to initial operation above ~~5%~~ of RATED THERMAL POWER, after each fuel loading, and

Specified in the OLR

b. The MTC shall be measured at any THERMAL POWER and compared to ~~-3.2×10^{-4} Δk/k/°F~~ (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than ~~-3.2×10^{-4} Δk/k/°F~~, the MTC shall be remeasured, and compared to the EOL MTC limit ~~of Specification 3.1.1.3b.~~, at least once per 14 EFPD during the remainder of the fuel cycle.

the 300 ppm surveillance limit specified in the OLR

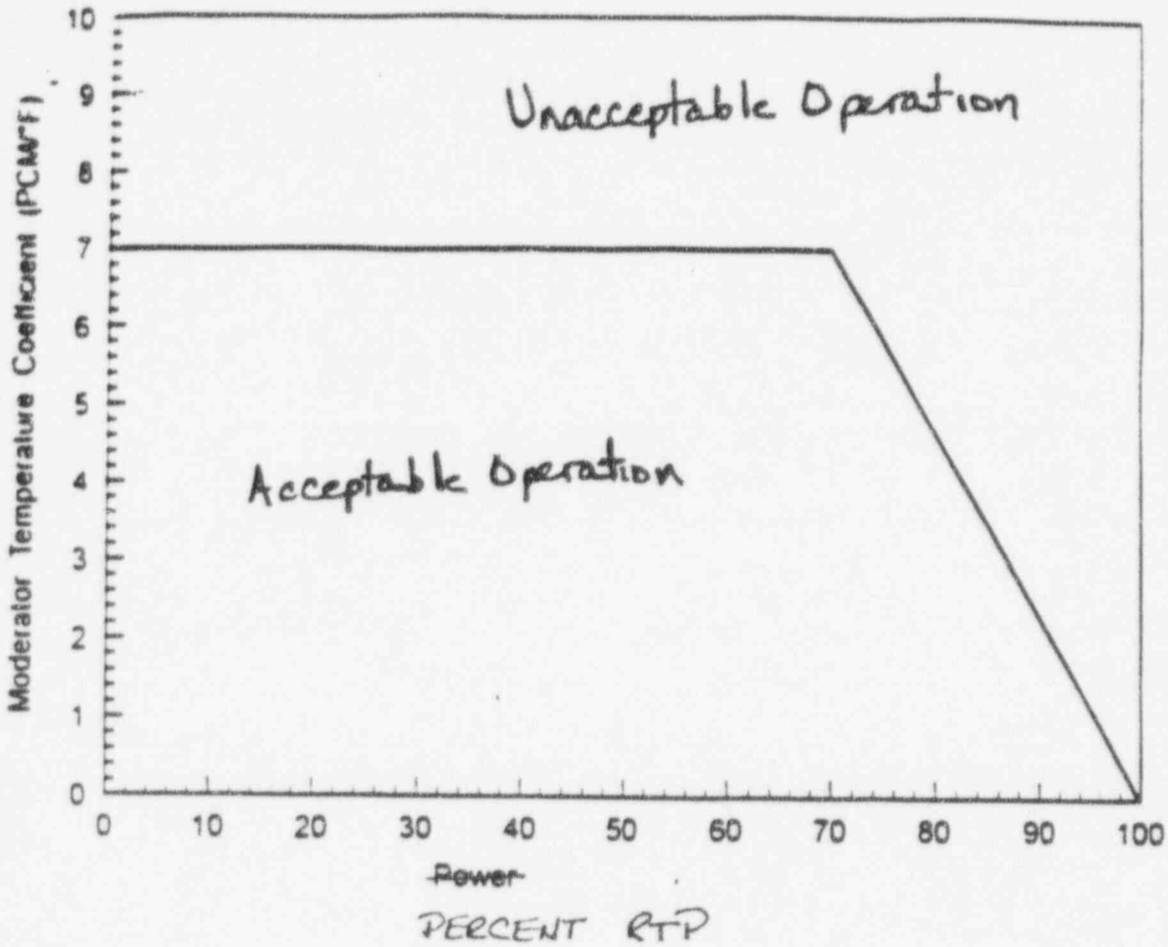

FIGURE 3.1-0

MODERATOR TEMPERATURE COEFFICIENT VS. POWER LEVEL

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that: (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.3% $\Delta k/k$ is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% $\Delta k/k$ SHUTDOWN MARGIN provides adequate protection provided that boration dilution paths are isolated. A 1.3% $\Delta k/k$ SHUTDOWN MARGIN is required to ensure the OPERABILITY of the automatic Boron Dilution Protection System.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within ~~the limiting condition~~ assumed in the ~~FSAR~~ UFSAR accident and transient analyses. INSERT B

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

INSERT B

The limitations on MTC also ensure that the Anticipated Transient Without Scram (ATWS) risk is acceptable. A cycle specific Unfavorable Exposure Time (UET) value will be calculated to ensure < 5% of the cycle operations occur when the reactivity feedback is not sufficient to prevent exceeding an ATWS overpressurization condition of $\geq$ 3200 psig in the RCS. This UET value will be updated for each core reload and appropriately considers the effects of changes in MTC, including any variations that are more adverse than those originally modeled in the analyses supporting the basis for the final ATWS rule.

MODERATOR TEMPERATURE COEFFICIENT (Continued)

The most negative MTC value equivalent to the most positive moderator UFSAR density coefficient (MDC), was obtained by incrementally correcting the MDC used in the ~~FSAR~~ analyses to nominal operating conditions. These corrections involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn EOL condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting MTC value ~~4.1×10^{-4} $\Delta k/k/°F$.~~ The ~~MTC~~ 300 ppm Surveillance limit ~~value of 3.2×10^{-4} $\Delta k/k/°F$~~ represents a conservative ~~value~~ (with corrections for burnup and soluble boron) ~~at a core condition of 300 ppm~~ equilibrium boron concentration ~~and is obtained~~ by making these corrections to the limiting MTC value ~~of 4.1×10^{-4} $\Delta k/k/°F$.~~ specified in the OLR. with an EOL

The Surveillance Requirements ~~for measurement~~ of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC can be maintained within its limits. The BOL MTC measurement, combined with the predicted MTC throughout core life, will be used to impose administrative limits on rod withdrawal, as required during core life to ensure that MTC will always ~~be less positive than 0 $\Delta k/k/°F$.~~ remain within the limits specified in the OLR. This coefficient changes slowly due principally to the reduction in RCS ~~boron concentration~~ associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE ~~FOR CRITICALITY~~

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 550°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the trip instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, (4) the reactor vessel is above its minimum RT_{NDT} temperature, and (5) the plant is above the cooldown steam dump permissive, P-12.

3/4.1.2 BORATION SYSTEMS

The Boron Injection System ensures that negative reactivity control is available during each MODE of facility operation. The components required to perform this function include: (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, and (5) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 350°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1.3% $\Delta k/k$ after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement is 13,487 (15,780*) gallons of 7000-ppm borated water from the boric acid storage tanks or 54,014 (70,450*) gallons of 2300-ppm (2000-ppm*) borated water from the refueling water storage tank. A Boric Acid Storage System level of 40% ensures that there is a volume of greater than or equal to 13,487 (15,780*) gallons available. A RWST level of 89% ensures that there is a volume of greater than or equal to 395,000 gallons available.

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*Not applicable to Unit 1. Applicable to Unit 2 until completion of cycle 5.

REPORTING REQUIREMENTS (Continued)

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.6 The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT**

6.9.1.7 The Semiannual Radioactive Effluent Release Report covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

MONTHLY OPERATING REPORT

6.9.1.8 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or RCS safety valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the NRC Regional Office, no later than the 15th of each month following the calendar month covered by the report.

OPERATING LIMITS REPORT

INSERT C

6.9.1.9 Operating limits shall be established and documented in the OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle. The analytical methods used to determine the operating limits shall be those previously reviewed and approved by the NRC in Topical Reports: ~~1) WCAP-9272-P-A "Westinghouse Reload Safety Evaluations Methodology" dated July 1985, 2) WCAP-8385 "Power Distribution Control and Load Following Procedures" dated September 1974, 3) NFSR-0016 "Benchmark of PWR Nuclear Design Methods" dated July 1983, and/or 4) NFSR-0081 "Benchmark of PWR Nuclear Design Methods Using the PHOENIX-P and ANC Computer Codes" dated July 1990~~. NEW PARAGRAPH The operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met. The OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

*A single submittal may be made for a multi-unit station.

**A single submittal may be made for a multi-unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

INSERT C

1. WCAP 9272-P-A, "Westinghouse Reload Safety Evaluations Methodology" dated July 1985.
2. WCAP-8385,"Power Distribution Control and Load Following Procedures-Topical Report" dated September 1974.
3. NFSR-0016, "Commonwealth Edison Company Topical Report on Benchmark of PWR Nuclear Design Methods" dated 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", dated 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 Controls Systems"

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

INSERT A

~~3.1.1.3 The moderator temperature coefficient (MTC) shall be:~~

~~a. Less positive than 0 $\Delta k/k/^{\circ}F$ for the all rods withdrawn, hot zero THERMAL POWER condition, or~~

~~b. Less negative than -4.1×10^{-4} $\Delta k/k/^{\circ}F$ for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.~~

APPLICABILITY: ~~Specification 3.1.1.3a. - MODES 1 and 2* only#.~~
~~Specification 3.1.1.3b. - MODES 1, 2, and 3 only#.~~

ACTION:

a. With the MTC more positive than the BOL limit ~~of Specification 3.1.1.3a. above,~~ specified in the COLR, operation in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than ~~0 $\Delta k/k/^{\circ}F$~~ the BOL limit specified in the COLR within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6;

2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition; and

3. A Special Report is prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

4. The provisions of Specification 3.0.4 are not applicable.

b. With the MTC more negative than the EOL limit ~~of Specification 3.1.1.3b. above,~~ specified in the COLR, be in HOT SHUTDOWN within 12 hours.

*With K_{eff} greater than or equal to 1.

#See Special Test Exceptions Specification 3.10.3.

INSERT A

3.1.1.3 The moderator temperature coefficient (MTC) shall be within the limits specified in the Operating Limits Report (OLR). The maximum upper limit shall be less than or equal to that shown in Figure 3.1-0.

APPLICABILITY: Beginning of Life (BOL) limit - MODES 1 and 2* only#
End of Life (EOL) limit - MODES 1, 2, and 3 only#.

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS

4.1.1.3 The MTC shall be determined to be within its limits during each fuel cycle as follows:

the BOL

a. The MTC shall be measured and compared to the BOL predicted MTC to establish administrative rod withdrawal limits, as necessary to assure that limit ~~of Specification 3.1.1.3a., above,~~ is met throughout core life, prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading, ~~and~~

specified in the OLR

b. The MTC shall be measured at any THERMAL POWER and compared to ~~$-3.2 \times 10^{-4}\ \Delta k/k/^{\circ}F$~~ (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than ~~$-3.2 \times 10^{-4}\ \Delta k/k/^{\circ}F$~~, the MTC shall be remeasured, and compared to the EOL MTC limit ~~of Specification 3.1.1.3b.~~, at least once per 14 EFPD during the remainder of the fuel cycle.

the 300 ppm surveillance limit specified in the OLR

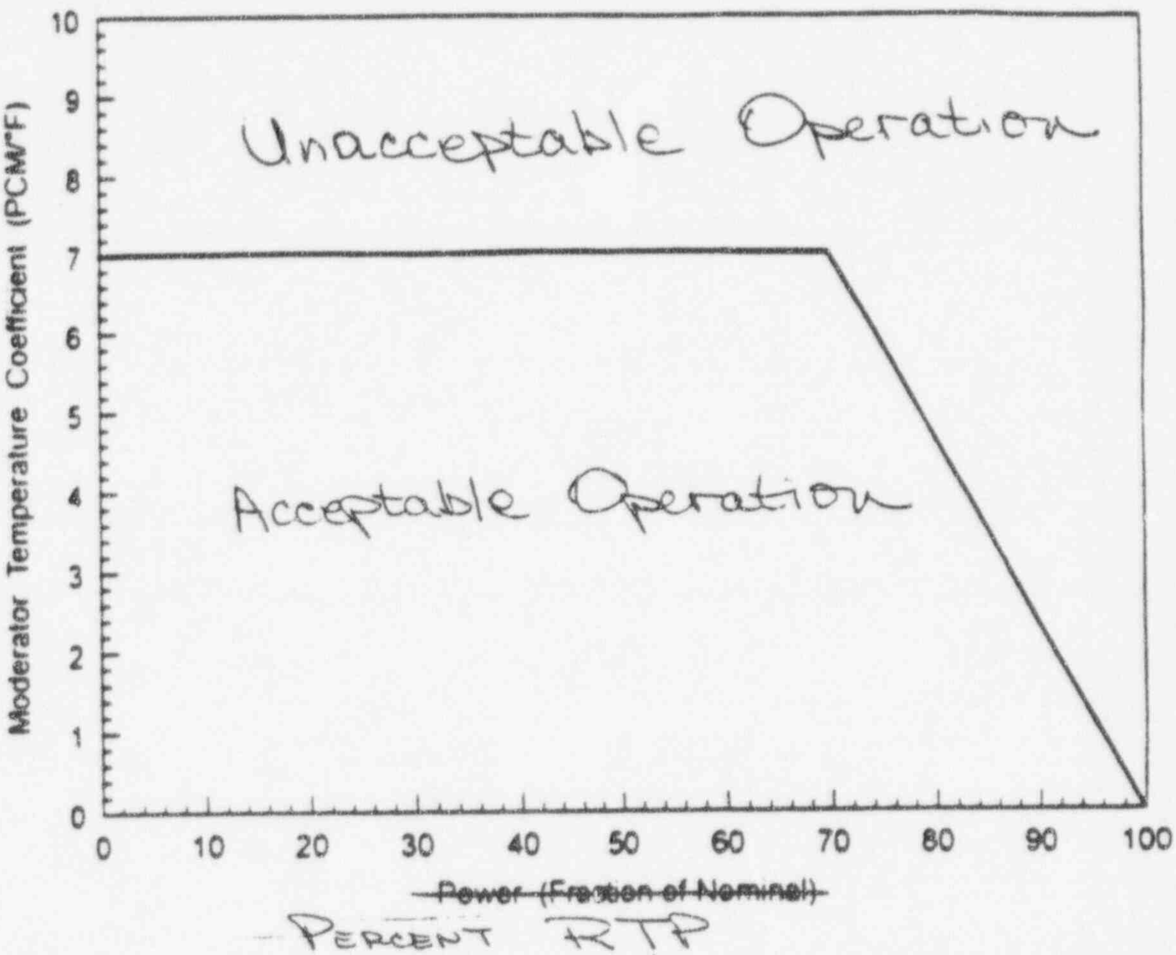

FIGURE 3.1-0

MODERATOR TEMPERATURE COEFFICIENT VS. POWER LEVEL

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that: (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.3% $\Delta k/k$ is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% $\Delta k/k$ SHUTDOWN MARGIN provides adequate protection provided that boration dilution paths are isolated. A 1.3% $\Delta k/k$ SHUTDOWN MARGIN is required to ensure the OPERABILITY of the automatic Boron Dilution Protection System.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the UFSAR ~~FSAR~~ accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

INSERT B

INSERT B

The limitations on MTC also ensure that the Anticipated Transient Without Scram (ATWS) risk is acceptable. A cycle specific Unfavorable Exposure Time (UET) value will be calculated to ensure < 5% of the cycle operations occur when the reactivity feedback is not sufficient to prevent exceeding an ATWS overpressurization condition of $\geq$ 3200 psig in the RCS. This UET value will be updated for each core reload and appropriately considers the effects of changes in MTC, including any variations that are more adverse than those originally modeled in the analyses supporting the basis for the final ATWS rule.

MODERATOR TEMPERATURE COEFFICIENT (Continued)

specified in the OLR

The most negative MTC value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the UFSAR ~~FSAR~~ analyses to nominal operating conditions. These corrections involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting EOL MTC value ~~-4.1×10^{-4} Δk/k/°F.~~ The ~~MTC value of -3.2×10^{-4} Δk/k/°F~~ 300 ppm surveillance limit represents a conservative value (with corrections for burnup and soluble boron) ~~at a core condition of 300 ppm~~ equilibrium boron concentration and is obtained by making with ARO these corrections to the limiting EOL MTC value. ~~of -4.1×10^{-4} Δk/k/°F.~~

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC can be maintained within its limits. The BOL MTC measurement combined with the predicted MTC with core burnup can be used to impose administrative limits on rod withdrawal to ensure that MTC will always ~~be less positive than 0 Δk/k/°F.~~ remain within the limits specified in the OLR. This coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 550°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the trip instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, (4) the reactor vessel is above its minimum RT_{NDT} temperature, and (5) the plant is above the cooldown steam dump permissive, P-12.

3/4.1.2 BORATION SYSTEMS

The Boron Injection System ensures that negative reactivity control is available during each MODE of facility operation. The components required to perform this function include: (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, and (5) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 350°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is 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 is 15,780 (13,487)* gallons of 7000-ppm borated water from the boric acid storage tanks or 70,450 (54,014)* gallons of 2000-ppm (2300-ppm)* borated water from the refueling water storage tank.

*Applicable to Unit 1 and Unit 2 starting with cycle 6.

REPORTING REQUIREMENTS (Continued)

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.6 The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT**

6.9.1.7 The Semiannual Radioactive Effluent Release Report covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

MONTHLY OPERATING REPORT

6.9.1.8 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or RCS safety valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the NRC Regional Office, no later than the 15th of each month following the calendar month covered by the report.

OPERATING LIMITS REPORT

6.9.1.9 Operating limits shall be established and documented in the OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle. The analytical methods used to determine the operating limits shall be those previously reviewed and approved by the NRC in Topical Reports: ~~1) WCAP 9272-P-A "Westinghouse Reload Safety Evaluations Methodology" dated July 1985~~, ~~2) WCAP-8385 "Power Distribution Control and Load Following Procedures" dated September 1974, 3) NFSR-0016 "Benchmark of PWR Nuclear Design Methods" dated July 1983, and/or 4) NFSR-0081 "Benchmark of PWR Nuclear Design Methods Using the PHOENIX-P and ANC Computer Codes" dated July 1990~~. The operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met. The OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

INSERT C

New paragraph

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*A single submittal may be made for a multi-unit station.

**A single submittal may be made for a multi-unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

INSERT C

1. WCAP 9272-P-A, "Westinghouse Reload Safety Evaluations Methodology" dated July 1985.
2. WCAP-8385,"Power Distribution Control and Load Following Procedures-Topical Report" dated September 1974.
3. NFSR-0016, "Commonwealth Edison Company Topical Report on Benchmark of PWR Nuclear Design Methods" dated 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", dated 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 Controls Systems"

ATTACHMENT C

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATIONS
FOR PROPOSED CHANGES TO
APPENDIX A TECHNICAL SPECIFICATIONS OF
FACILITY OPERATING LICENSES
NPF-37, NPF-66, NPF-72, and NPF-77

Commonwealth Edison (ComEd) has evaluated the proposed amendment and determined that it involves no significant hazards consideration. According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards 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.

A. INTRODUCTION

In a letter dated March 23, 1994 from J.A. Bauer to W.T. Russell, Commonwealth Edison Company (ComEd) requested a Technical Specification Amendment allowing use of a Positive Moderator Temperature Coefficient (PMTC) and Reduced Thermal Design Flow (RTDF). This Amendment request was later supplemented in a letter dated July 26, 1994 from J.A. Bauer to W.T. Russell to provide additional cycle specific implementation footnotes for clarification. An additional letter dated August 16, 1994, from D. Saccomando to W.T. Russell, transmitted responses to the NRC for an additional request for information regarding the effects a positive MTC would have on the results of the Anticipated Transient Without Scram (ATWS) analysis. ComEd subsequently received approval of the proposed Technical Specification Amendment on October 21, 1994 in a letter from G.F. Dick to D.L. Farrar. However, the approval specifically excluded the proposed PMTC Technical Specification change.

Based on subsequent meetings, discussions, and correspondence with the NRC, ComEd is resubmitting the Technical Specification Amendment request allowing use of a PMTC. The proposed changes are consistent with Standard Technical Specifications for Westinghouse Plants (NUREG-1431). That is, the Amendment request proposes that the MTC value be maintained within the limits specified in an

operating limits cycle specific report with a maximum upper limit specified in the Technical Specifications. Therefore, ComEd proposes to expand the current Operating Limits Report (OLR) to include a cycle specific MTC value.

The MTC change would allow a slightly positive MTC below 100 percent of rated full power. The principal benefit of this change is that it would facilitate the design of future reload cycles and yield significant fuel cost savings. The safety analyses for the Byron and Braidwood Updated Final Safety Analysis Report (UFSAR) transients were previously based on a maximum MTC being less than or equal to 0 pcm/°F at all times when the reactor is critical. The proposed change to the Technical Specifications would allow a maximum upper limit of +7 pcm/°F MTC for power levels up to 70 percent with a linear ramp to 0 pcm/°F at 100 percent power. However, the proposed Technical Specification would require that the cycle specific MTC value be maintained within the limits specified in the Operating Limits Report (OLR). The basis for the MTC limit is to ensure that the value of the coefficient remains within the limits assumed in the UFSAR accident and transient analyses. In keeping with this basis, the necessary accident and transient analyses were performed with the new MTC values to ensure that the results remain within all design and safety criteria. The analysis provides the basis for the proposed MTC Technical Specification change. This analysis, WCAP-13964 "Byron and Braidwood Units 1 and 2 Increased SGTP/Reduced TDF/PMTC Analysis Program Engineering/Licensing Report", was provided in Attachment 5 of the March 23, 1994 submittal.

To accommodate the positive MTC changes and the potential of lengthened reload fuel cycles due to increased energy requirements, Technical Specification changes are also required to meet shutdown margin requirements (SDM). To assure subcriticality requirements are met following a postulated loss-of-coolant accident (LOCA), the boron concentration is increased in the refueling water storage tank (RWST) and the safety injection accumulators. These changes have already been approved by the NRC in response to the original submittal.

Determination of a cycle specific MTC value will include an evaluation of Anticipated Transient Without Scram (ATWS) risk on a deterministic basis, since the ATWS rule was based in part on a given MTC value. The Unfavorable Exposure Time (UET) methodology will be used for evaluating ATWS risk on a deterministic basis. A UET value will be calculated each cycle. UET is defined as the amount of time during the operating cycle for which the reactivity feedback is not sufficient to prevent Reactor Coolant System (RCS) pressure from exceeding 3200 psig for a given plant configuration. The ATWS risk is considered acceptable for a UET of less than 5% of cycle length. This UET value will be updated for each core reload and appropriately considers the effects of changes in MTC, including any variations that are more adverse than those originally modeled in the analyses supporting the basis for the final ATWS rule.

B. 10 CFR 50.92 ANALYSIS

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

   An analysis program was pursued by Commonwealth Edison to justify a positive MTC, reduced reactor coolant system thermal design flow, and increased steam generator tube plugging levels. This analysis identified a need for corresponding increase in the boron concentration levels in the refueling water storage tank (RWST) and safety injection accumulators to assure subcriticality requirements are met following a postulated loss-of-coolant accident (LOCA). The increases in boron concentration are based on the maximum upper limit of the MTC. The corresponding Technical Specification changes required as a result of this analysis program were previously approved by the NRC, including the increases in boron concentration limits, with the exception of the positive MTC change. The safety analyses necessary to support this program are documented in WCAP-13964. The results were reviewed by Commonwealth Edison and found to be acceptable. All Departure from Nucleate Boiling Ratio (DNBR) design limits were determined such that there was a 95 percent probability at a 95 percent confidence level that DNB would not occur on the most limiting fuel rod for any Condition I or Condition II event. The present Technical Specification limit for Nuclear Enthalpy Rise Hot Channel Factor, $F^{N}_{\Delta H}$, of less than 1.65 ensures that the DNB design basis stated above would be met, thus fuel integrity will not be challenged.

   The accidents which are sensitive to MTC were analyzed as part of the overall program and the results were found to be acceptable. The safety functions of the evaluated systems and components remain unchanged. The analysis performed using the increased MTC value does not affect the integrity of the safety related systems and components such that their function to control radiological consequences is affected and all fission barriers will remain intact. The effects on offsite doses have been considered. The incorporation of a positive MTC, in conjunction with the previously approved reduction in reactor coolant system thermal design flow rate and increase in steam generator tube plugging levels, will result in a small increase in offsite doses; however, the total doses remain a small fraction of the 10CFR100 limits. As such, the accident analysis acceptance criteria continue to be satisfied.

   On a cycle-by-cycle basis, a deterministic evaluation of the impact on ATWS risk will be performed. An Unfavorable Exposure Time (UET) will be calculated, where UET is defined as the amount of time during the operating cycle for which the reactivity feedback is not sufficient to prevent Reactor Coolant System (RCS) pressure from exceeding 3200 psig for a given plant configuration. The UET methodology is consistent with the Westinghouse

Owner's Group methodology presented in WCAP 11992, "ATWS Rule Administration Process" and WCAP 11993, "Assessment of Compliance with ATWS Rule Basis for Westinghouse PWRs". Corrective actions will be taken, as necessary, to assure a UET of less than 5% of cycle length.

Therefore, implementation of a positive MTC will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed changes do not create the possibility of a new or different type of accident from any accident previously evaluated.

The methodology and manner of plant operation as a result of the proposed changes is unaffected. Implementation of a positive MTC does not impact the safe operation of the reactor provided that the Limiting Conditions for Operation (LCOs) and the associated action requirements are satisfied. The assumptions do not create any new failure modes that could adversely impact safety related equipment. The reload safety limits and LCOs in the plant Technical Specifications will be evaluated and satisfied for each future reload core design via the 10CFR50.59 process. All DNBR limits have been satisfied. Currently installed equipment will not be operated in a manner different than previously designed. No new credible limiting single failure has been created. No new or different accidents or failure modes have been identified for any systems or components important to safety.

Therefore, there is not a potential for creating the possibility of a new or different type of accident from any accident previously evaluated.

3. The proposed changes do not involve a significant reduction in a margin of safety.

The performance and integrity of the evaluated safety related systems and components are not affected by the proposed change to the MTC. The radiological consequences of all previously analyzed accidents remain within acceptable limits. The proposed change to the MTC will have no effect on the availability, operability, or performance of the evaluated safety related systems or components. The incorporation of a positive MTC, in conjunction with the previously approved reduction in reactor coolant system thermal design flow rate and increase in steam generator tube plugging levels, will result in a small increase in offsite doses; however, the total doses remain a small fraction of the 10CFR100 limits. The methodology, discussed in D. Saccomando letter to NRC dated December 21, 1994, describes the determination and use of the UET values in the calculation of the Primary Pressure Relief node for the ATWS event tree to determine an overall ATWS risk value. The methodology will be used by ComEd to ensure that a core designed with a positive MTC will not

result in an unacceptable risk to core damage frequency due to an ATWS event. The margin of safety associated with the licensing basis safety analysis is not significantly reduced by the proposed changes. All acceptance criteria for the specific UFSAR Chapter 15 safety analyses (non-LOCA and LOCA) have been satisfactorily evaluated and verified using NRC approved methodologies.

Therefore, there is no significant reduction in the margin of safety as defined in the bases of any Technical Specification.

Based on the above evaluation, Commonwealth Edison has concluded that implementation of a positive MTC does not involve a significant hazards consideration with respect to the provisions of 10CFR50.92.

ATTACHMENT D

ENVIRONMENTAL ASSESSMENT FOR
PROPOSED CHANGES TO
APPENDIX A TECHNICAL SPECIFICATIONS OF
FACILITY OPERATING LICENSES
NPF-37, NPF-66, NPF-72, and NPF-77

Commonwealth Edison has evaluated the proposed changes associated with implementing a positive MTC against the criteria for and identification of licensing and regulatory actions requiring environmental assessment in accordance with 10CFR 51.21. It has been determined that the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10CFR51.22(c). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10CFR50, it involves changes to a surveillance requirement, and the amendment meets the following specific criteria:

(i) the amendment involves no significant hazards consideration.

As demonstrated in Attachment C, this proposed amendment does not involve any significant hazards considerations.

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

The effects on offsite doses have been considered. The analysis performed to justify a positive MTC, a reduction in thermal design flow (TDF) and increased tube plugging levels demonstrates a resultant increase in offsite doses. However, the increases are small and the total doses are a small fraction of the 10CFR100 limits. As such, the acceptance criteria continue to be satisfied. The reduction in thermal design flow and the increase in tube plugging levels have already been approved and will be implemented on a cycle specific basis. The assumptions used in the analysis to justify a positive MTC do not change, degrade, or prevent the response of the evaluated safety related systems and components such that their function in the control of radiological consequences is affected.

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

This proposed change will not result in changes in the operation or configuration

of the facility. There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste; nor will the proposal result in any change in the normal radiation levels in the plant. Therefore, there will be no increase in individual or cumulative occupational radiation exposure resulting from this change.

Commonwealth Edison has evaluated the proposed amendment against the criteria and found the changes meet the categorical exclusion permitted by 10CFR51.22(c)(9).

ATTACHMENT E

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 Controls Systems"

Commonwealth Edison
1400 Opus Place
Downers Grove, Illinois 60515

December 21, 1994

Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attn: Document Control Desk

Subject: Additional Information Regarding Application for Amendment to Facility Operating Licenses-Reactivity Controls Systems

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

References:
1) Teleconference dated December 15, 1994, between Commonwealth Edison Company and the Nuclear Regulatory Commission regarding the Positive Moderator Temperature Coefficient

2) G. Dick letter to D. Farrar dated October 21, 1994, transmitting Safety Evaluation pertaining to the Reduced Thermal Design Flow

3) J. Bauer letter to W. Russell dated July 26, 1994, transmitting supplement to proposed Byron and Braidwood license amendment addressing "Positive Moderator Temperature Coefficient and Reduced Thermal Design Flow"

4) J. Bauer letter to W. Russell dated March 23, 1994, transmitting a proposed Byron and Braidwood license amendment addressing "Positive Moderator Temperature Coefficient and Reduced Thermal Design Flow"

Reference 4 transmitted Commonwealth Edison Company's (ComEd) request to amend the Technical Specifications for Braidwood and Byron Station addressing positive moderator temperature coefficient (PMTC) and reduced thermal design flow. This amendment request was subsequently supplemented in Reference 3. The Nuclear Regulatory Commission (NRC) issued the referenced Safety Evaluation for the reduced thermal design flow portion of the amendment request. Several conference calls were held between the NRC and ComEd concerning the PMTC portion of the amendment. During the reference teleconference, ComEd agreed to provide the NRC with a document that describes which sections of the previously submitted material are applicable to ComEd's application of the Unfavorable Exposure Time methodology. This document is attached.

Attachment

WCAP-11992/11993 UET Sections and

Explanation of ComEd Application of UET Methodology

This document provides guidance as to which sections of WCAPs 11992 and 11993 (References 1 and 2) are applicable to the Commonwealth Edison (ComEd) application of the Unfavorable Exposure Time (UET) methodology. The UET methodology is used for evaluating Anticipated Transients Without Scram (ATWS) risk on a deterministic basis. This document also provides details of how ComEd will apply the methodology to cycle-specific calculations for the Byron/Braidwood stations.

It should be noted that ComEd intends for this subset of the Westinghouse methodology to apply only until a review of the complete Westinghouse methodology previously submitted in Reference 3 can be performed. ComEd is working with the Westinghouse Owners Group (WOG) to determine an appropriate licensing approach to resolve this issue for the long term.

WCAP-11992/11993 Applicable Sections

WCAP-11992 describes the administration process developed by the WOG to determine ATWS risk throughout a plant's lifetime. WCAP-11993 provides results of the application of the process to show that Westinghouse plants were in compliance with the original ATWS rule basis at the time of the study.

Both WCAPs 11992 and 11993 discuss the historical perspective of the ATWS rule, the Probabilistic Risk Assessment (PRA) model used by the NRC in the original rulemaking, and the expanded PRA model developed by the WOG to calculate ATWS risk. These documents provide useful information in understanding the ATWS risk model; however, since the PRA overall methodology is not being applied here, only the sections which specifically refer to the critical trajectory and UET methodologies are discussed below. Since the discussion of UET is almost identical in both documents, the section numbers in the following discussion refer to WCAP-11992.

- Section 4.3.8 discusses the pressure relief node of the WOG ATWS event tree, and provides a definition for UET.
- Section 4.6.8 describes the assumptions and operating conditions at which the UET is evaluated. This section also discusses which sets of equipment availability are considered in the UET calculations.
- The UET methodology is described in detail in Appendix B, Section B.7.1, "ATWS Critical Power Trajectory Methodology." This section establishes the reference plant analysis from the previous 1979 ATWS Submittal (Reference 4) as the basis for the sensitivity studies performed to determine the critical power trajectories.

As stated in WCAP-11992, sensitivity studies were performed to determine the reactivity feedback conditions required to yield a peak RCS pressure equivalent to 3200 psig for various RCS pressure relief capacities (PORV and auxiliary feedwater availabilities). The transient model analysis used a point kinetics model which must be transformed into equivalent steady state reactor conditions to allow for comparison with steady state core evaluation models. This transformation is possible since the limiting ATWS transients are characterized by a relatively slow heatup and pressurization of the RCS and the conditions at important times during the transient have been demonstrated to be quasi-steady state.

With the transient reactivity feedback conditions transformed for various RCS pressure relief capacity conditions, the reactor heatup shutdown characteristics, or critical trajectories, are determined for each pressure relief capacity. These resulting critical trajectories (Figures B-1A through B-1F in WCAP-11992), which are in the form of steady state reactor power versus inlet coolant temperature, represent the locus of conditions (power vs. inlet temperature) that result in a peak RCS pressure of 3200 psig in the transient analysis of the limiting ATWS event for the reference plant configuration.

ComEd Application of Westinghouse UET Methodology

The WOG study defines the term UET as the time during cycle life when the core reactivity feedback is not sufficient to prevent RCS pressure from exceeding 3200 psig for a given plant configuration (initiating event, power level, manual rod insertion, auxiliary feedwater flow, and PORV availability). The WOG methodology provides for calculations of UET for different PORV availabilities, Auxiliary Feedwater (AFW) system capabilities, and whether or not manual control rod insertion has occurred. The methodology, as applied by ComEd, will employ the "base case" set of conditions from Reference 4 (100% PORV capacity, or both PORVs available, 100% AFW capacity, and no control rod insertion). The critical trajectory corresponding to this set of conditions is shown in Figure B-1A of WCAP-11992. The use of this (base case) set of conditions is based on a meeting between ComEd and the NRC staff on September 15, 1994. At this meeting, NRC staff personnel stated that this approach was equivalent to the original ATWS rule (base case) used to show the 3200 psig criterion was met for 95% of the cycle length.

Since the Moderator Temperature Coefficient (MTC) is a component of the total reactivity feedback, having a positive MTC has a direct impact on the UET value.

The reload specific analysis considers the following:

1. Initial conditions are assumed to be nominal equilibrium conditions (100% power, all rods out, equilibrium xenon)

2. A power search is performed for criticality conditions as a function of core inlet temperature assuming a pressure of 3200 psig; and

3. Calculations need only be done using the low end of the previous cycle burnup window, since this is the most reactive core configuration, i.e., the condition which yields the most positive current cycle MTC.

The resulting calculated critical state powers are compared to the critical trajectory established from the transient analysis. This comparison shows any core design conditions (in terms of power versus inlet temperature) that are greater than the transient conditions (e.g., those which would result in peak RCS pressure exceeding the stress criterion of 3200 psig). The UET is that time during the cycle that the cycle specific core design critical trajectory is greater than the transient critical trajectory.

Hence, the UET value (or the time during cycle life when the reactivity feedback is not sufficient to prevent exceeding 3200 psig) appropriately considers the effects of changes in MTC, including any variations that are more adverse than those originally modeled in the analyses supporting the basis for the final ATWS rule. Furthermore, by addressing changes in core reactivity feedback in this manner, compliance with the final ATWS rule is appropriately demonstrated for the base case set of conditions by showing that the UET for a given cycle core design will remain below 5%, or conversely, the ATWS overpressure acceptance criterion will be met for 95% of the cycle length.

References

1. "Joint Westinghouse Owners Group/Westinghouse Program: ATWS Rule Administration Process," WCAP-11992, December, 1988.

2. "Joint Westinghouse Owners Group/Westinghouse Program: Assessment of Compliance with ATWS Rule Basis for Westinghouse PWRs," WCAP-11993, December, 1988.

3. "Response to Request for Additional Information Regarding a Proposed License Amendment Addressing Positive Moderator Temperature Coefficients and Reduced Thermal Design Flow," Letter from D. M. Saccomando (ComEd) to W. T. Russell (NRC), August 16, 1994.

4. NS-TMA-2182, Letter from T. M. Anderson (Westinghouse) to Dr. S. H. Hanauer (NRC) dated December 30, 1979, "ATWS Submittal".

ATTACHMENT F

Example OLR to include Positive Moderator Temperature Coefficient limits

EXAMPLE OLR

STATION [Z] UNIT [X] CYCLE [Y]

The Operating Limits Report is provided in accordance with Specification 6.9.1.9 of the [Z] Station Technical Specifications. These limits have been developed using the NRC-approved methodologies specified in Specification 6.9.1.9.

<u>OPERATING LIMITS REPORT - Fxy PORTION</u>

The Fxy limits for RATED THERMAL POWER within specified core planes for Cycle [Y] shall be (values include missing support pin penalty):

a: For the lower core region from greater than or equal to 0% to less than or equal to 50%:

1) For all core planes containing bank"D" control rods:

$Fxy^{RTP} \leq [\quad]$

2) For all unrodded core planes:

$Fxy^{RTP} \leq [\quad]$

b: For the upper core region from greater than 50% to less than or equal to 100%:

1) For all core planes containing bank"D" control rods:

$Fxy^{RTP} \leq [\quad]$

2) For all unrodded core planes:

$Fxy^{RTP} \leq [\quad]$

These Fxy(z) limits were used to confirm that the heat flux hot channel factor FQ(z) will be limited to the Technical Specification values of

$Fq(z) \leq [2.50/P]\ [K(z)]$ for $P > 0.5$ and

$Fq(z) \leq [\ 5.00\ ]\ [K(z)]$ for $P \leq 0.5$

assuming the most limiting axial power distributions expected to result from the insertion and removal of control Banks C and D during operation, including the accompanying variations in the axial xenon and power distributions as described in the "Power Distribution Control and Load Following Procedures," WCAP-8403, September 1974. Therefore, these Fxy limits provide assurance that the initial conditions assumed in the LOCA analysis and the Emergency Core Cooling Systems (ECCS) acceptance criteria of 10 CFR 50.46 are met.

OPERATING LIMITS REPORT - MTC PORTION

a) The Moderator Temperature Coefficient (MTC) limits are:

1) The BOL/ARO/HZP-MTC shall be less positive than [ ] $\Delta k/k/^{\circ}F$.

2) The EOL/ARO/RTP-MTC shall be less negative than [-] $\Delta k/k/^{\circ}F$.

b) The MTC Surveillance limit is:

The 300 ppm/ARO/RTP-MTC should be less negative than or equal to [-] $\Delta k/k/^{\circ}F$.

where: BOL stands for Beginning of Cycle Life
ARO stands for All Rods Out
HZP stands for Hot Zero Thermal Power
EOL stands for End of Cycle Life
RTP stands for RATED THERMAL POWER