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10 CFR 50.90

RS-00-112

October 24, 2000

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

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

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

Subject: Request for a License Amendment for Plant Specific Use of Best Estimate Large Break Loss of Coolant Accident Analysis

Reference: Letter from R. M. Krich to the NRC, "Request for a License Amendment to Permit Uprated Power Operations at Byron and Braidwood Stations," dated July 5, 2000

In accordance with 10 CFR 50.90, Commonwealth Edison (ComEd) Company is requesting changes to Appendix A, Technical Specifications (TS) of Facility Operating License Nos. NPF-72, NPF-77, NPF-37 and NPF-66, for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, respectively. The requested amendment proposes to apply the Westinghouse generic Best Estimate Large Break Loss of Coolant Accident (LOCA) Analysis methodology using the WCOBRA/TRAC computer code and would modify the documents referenced in TS 5.6.5, "Core Operating Limits Report (COLR)," to reflect this change. WCOBRA/TRAC is an analysis tool that models the reactor core thermal hydraulic behavior for a spectrum of hypothetical loss of coolant accidents. This analysis is being performed specifically to support operation at uprated power conditions as described in the referenced letter.

A001

A plant specific analysis of the Byron Station and Braidwood Station has been performed using the NRC approved methodology specified in WCAP-12945-P-A, "Code Qualification Document For Best Estimate LOCA Analysis." The NRC approved this methodology in a letter from R. C. Jones, NRC, to N. J. Liparulo, Westinghouse Electric Corporation, "Acceptance for Referencing of the Topical Report WCAP-12945 (P), 'Westinghouse Code Qualification Document For Best Estimate Loss of Coolant Analysis,'" dated June 28, 1996. The WCOBRA/TRAC computer code is described in WCAP-12945-P-A. It has been determined that application of this methodology is appropriate for each unit at Byron Station and Braidwood Station. The TS changes are being made to incorporate the Westinghouse best estimate large break LOCA analysis methodology into the licensing basis for the Byron Station, Units 1 and 2, and the Braidwood Station, Units 1 and 2, consistent with 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," and Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," dated May 1989.

This amendment 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 provide the marked up TS pages with the proposed changes indicated for Byron Station and Braidwood Station. Attachments B-3 and B-4 provide the typed TS pages with the proposed changes incorporated. The associated Bases pages are also included for informational purposes.

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. We have determined that the proposed changes will not significantly increase the amount of any effluent which may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure.

We request that the NRC review and approve the proposed TS changes by May 7, 2001, to support a power uprate of Byron Station, Unit 1 and Unit 2, and Braidwood Station, Unit 1 and Unit 2, as described in the referenced letter.

These proposed changes have been reviewed and approved by the Byron and Braidwood Stations' Plant Operations Review Committees and the Nuclear Safety Review Boards in accordance with the requirements of the ComEd Quality Assurance Program.

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

Should you have any questions relative to this submittal, please contact Mr. J. A. Bauer at (630) 663-7287.

Respectfully,

R. M. Krich
Vice President – Regulatory Services

Attachments: Attachment A, Description and Safety Analysis for Proposed Changes
Attachment B-1, Marked-up Pages For Proposed Changes, Byron Station
Attachment B-2, Marked-up Pages For Proposed Changes, Braidwood Station
Attachment B-3, Incorporated Proposed Changes, Typed Pages, Byron Station
Attachment B-4, Incorporated Proposed Changes, Typed Pages, Braidwood Station
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

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

SUBJECT: **Request for a License Amendment for Plant Specific Use of Best Estimate Large Break Loss of Coolant Accident Analysis**

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

R. M. Krich
Vice President - Regulatory Services

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

for the State above named, this 24 day of

October, 20 00.

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

Jeffery A. Batara

Notary Public

ATTACHMENT A

BYRON STATION, UNITS 1 AND 2
BRAIDWOOD STATION, UNITS 1 AND 2

DESCRIPTION AND SAFETY ANALYSIS FOR PROPOSED CHANGES

A. SUMMARY OF PROPOSED CHANGES

In accordance with 10 CFR 50.90, Commonwealth Edison (ComEd) Company is requesting changes to Appendix A, Technical Specifications (TS) of Facility Operating License Nos. NPF-72, NPF-77, NPF-37 and NPF-66, for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, respectively. The proposed changes revise TS 5.6.5, "Core Operating Limits Report (COLR)," to reference the appropriate Westinghouse topical report that describes the Best Estimate Large Break Loss of Coolant Accident (LOCA) analysis methodology. This analysis was performed using the NRC approved Westinghouse Best Estimate LOCA model WCOBRA/TRAC. The NRC approved this methodology in a letter from R. C. Jones, NRC, to N. J. Liparulo, Westinghouse Electric Corporation, "Acceptance for Referencing of the Topical Report WCAP-12945 (P), 'Westinghouse Code Qualification Document For Best Estimate Loss of Coolant Analysis,'" dated June 28, 1996. WCOBRA/TRAC is an analysis tool that models the reactor core thermal hydraulic behavior for a spectrum of hypothetical loss of coolant accidents. The specific proposed changes to the TS are indicated in the marked up copy of the applicable TS page provided in Attachment B, "Marked-Up Changes for Proposed Changes." This analysis is being performed to support operation at uprated power conditions as noted in a letter from R. M. Krich (ComEd) to the NRC, "Request for a License Amendment to Permit Uprated Power Operations at Byron And Braidwood Stations," dated July 5, 2000.

B. DESCRIPTION OF THE CURRENT REQUIREMENTS

TS 5.6.5, "Core Operating Limits Report (COLR)," Items b.6, b.7, and b.8 currently provide references to the Westinghouse topical reports used for the Byron Station and Braidwood Station Large Break LOCA analysis. These references are as follows.

6. WCAP-9220-P-A, "Westinghouse ECCS Evaluation Model-1981 Version," February 1982.

7. WCAP-9561-P-A, Add. 3, "BART A-1: a Computer Code for Best Estimate Analysis of Reflood Transients – Special Report: Thimble Modeling in Westinghouse ECCS Evaluation Model," July 1986.

8. WCAP-10266-P-A, "The 1981 Version of Westinghouse Evaluation Model using BASH Code," March 1987, including Addendum 1 "Power Shape Sensitivity Studies," Revision 2-P-A, dated December 15, 1987, and Addendum 2 "BASH Methodology Improvements and Reliability Enhancements," Revision 2, dated May 1988.

C. BASES FOR THE CURRENT REQUIREMENTS

The current Large Break LOCA analysis utilizes the methodologies described in the references listed in Items b.6, b.7, and b.8 of TS 5.6.5. as noted above in Section B. These methodologies demonstrate that all acceptance criteria of 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," are satisfied.

D. NEED FOR REVISION OF THE REQUIREMENTS

In a letter from R. M. Krich (ComEd) to the NRC, "Request for a License Amendment to Permit Uprated Power Operations at Byron and Braidwood Stations," dated July 5, 2000, ComEd submitted a request to the NRC, with supporting analysis, to increase the licensed power level of the Byron Station and Braidwood Station units from 3411 Megawatts-thermal (MWt) to 3586.6 MWt. Analysis results of large break LOCA transients were not submitted with our July 5, 2000, submittal. Reanalysis of the large break LOCA transients, utilizing the NRC approved Westinghouse Best Estimate LOCA model WCOBRA/TRAC, was therefore performed to demonstrate that 10 CFR 50.46 acceptance criteria are satisfied at uprated power conditions.

E. DESCRIPTION OF THE PROPOSED CHANGES

TS 5.6.5.b identifies the applicable references for the analytical methods used to determine the core operating limits and specifies that they shall be previously reviewed and approved by the NRC. The current references in Items b.6, b.7, and b.8 list the Westinghouse topical reports that describe the large break LOCA analysis methodology. The large break LOCA analyses have been re-performed utilizing the NRC approved Westinghouse Best Estimate LOCA model WCOBRA/TRAC; therefore, the references in Items b.6, b.7, and b.8 will be deleted and replaced with Westinghouse topical report, WCAP-12945-P-A, Volume 1, Revision 2, and Volumes 2 through 5, Revision 1, "Code Qualification Document for Best Estimate LOCA Analysis," March 1998.

F. SAFETY ANALYSIS OF THE PROPOSED CHANGES

A best estimate large break LOCA analysis has been performed for Byron Station and Braidwood Station using the NRC approved Westinghouse best estimate large break LOCA methodology described in WCAP-12945-P-A. The NRC approved this methodology in a letter from R. C. Jones, NRC, to N. J. Liparulo, Westinghouse Electric Corporation, "Acceptance for Referencing of the Topical Report WCAP-12945 (P), 'Westinghouse Code Qualification Document For Best Estimate Loss of Coolant Analysis,'" dated June 28, 1996. All plant specific parameters used in the analysis are bounded by the models and correlations contained in the generic methodology. Therefore, the Byron Station and Braidwood Station specific analysis satisfies the acceptance criteria of 10 CFR 50.46, conforms to the requirements of 10 CFR 50, Appendix K, "ECCS Evaluation Models," Section II, "Required Documentation," and meets the guidance of Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," dated May 1989.

It has been determined that application of this methodology is appropriate for each unit at Byron Station and Braidwood Station. The Unit 1 and Unit 2 analyses were performed separately to reflect the different steam generator design of Unit 1 vice Unit 2. Note that the Unit 1 steam generators at Byron Station and Braidwood Station are of the same design; likewise, the Unit 2 steam generators at Byron Station and Braidwood Station are of the same design. The Byron Station and Braidwood Station reactor vessel designs are fundamentally identical for all units. There are; however, minor differences in the detailed design of certain components, which affect the metal mass and fluid volume in different regions of the vessel. These minor differences were addressed by use of the bounding plant design when there was a clear direction of conservatism (e.g., maximizing metal masses that contribute to lower plenum/downcomer heatup and boiling, and minimizing fluid volumes that contribute to core cooling). When the differences pertained to parameters judged to be inconsequential, a representative plant choice was used. The analysis input assumptions were confirmed to bound the actual "as operated" plant parameters. In addition, each unit's cycle-specific COLR will be updated and issued prior to implementation of the best estimate large break LOCA analysis methodology. The updated COLRs will update, as needed, any reactor core cycle-specific parameters and will reconfirm that all reactor core design criteria continue to be met.

In accordance with 10 CFR 50.46, the conclusions of the best estimate large break LOCA analysis are that there is a high level probability that the following criteria are met.

1) The calculated maximum fuel element cladding temperature (i.e., peak cladding temperature (PCT)) will not exceed 2200°F.

2) The calculated total oxidation of the cladding (i.e., maximum cladding oxidation) will nowhere exceed 0.17 times the total cladding thickness before oxidation.

3) The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam (i.e., maximum hydrogen generation) will not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

4) The calculated changes in core geometry are such that the core remains amenable to cooling.

   Note that criterion 4 has historically been satisfied by adherence to criteria 1 and 2, and by assuring that fuel deformation due to combined LOCA and seismic loads is specifically addressed. Criteria 1 and 2 are satisfied for best estimate large break LOCA applications. The approved methodology specifies that effects of LOCA and seismic loads on core geometry do not need to be considered unless grid crushing extends beyond the assemblies in the low power channel as defined in the WCOBRA/TRAC model. This situation has not been calculated to occur for the Byron Station or Braidwood Station. Therefore, acceptance criterion 4 is satisfied.

5) After successful initial operation of the Emergency Core Cooling System (ECCS), the core temperature will be maintained at an acceptably low value and decay heat will be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

   The approved best estimate large break LOCA methodology position on criterion 5 is that this requirement is satisfied if a coolable core geometry is maintained, and the core remains subcritical following the LOCA. This position is unaffected by the application of the best estimate large break LOCA methodology for Byron Station and Braidwood Station.

Table 1 lists the plant specific parameters used in the Byron Station and Braidwood Station plant specific analyses and the intended location (e.g., Updated Final Safety Analysis Report (UFSAR) or COLR) of the documentation of the values and ranges used for the specific parameters.

Table 2 presents the calculated 50th and 95th percentile PCT, maximum cladding oxidation, maximum hydrogen generation, and core cooling results.

Table 3 lists the calculated minimum injected ECCS flow into the three intact loops of the reactor coolant system (RCS).

Figure 1 presents the assumed axial power shape operating space envelope.

Figure 2 shows the minimum calculated containment backpressure boundary condition.

Based on the analysis, the Westinghouse Best Estimate Large Break LOCA methodology has shown that all related plant design criteria continue to be met and the acceptance criteria of 10 CFR 50.46 continue to be satisfied.

Table 1

Byron Station and Braidwood Station
Major Plant Parameter Assumptions
Used in the Best Estimate Large Break LOCA Analysis

Parameter	Value	Intended Location
Plant Physical Description		
Steam Generator Tube Plugging	$\leq$ 5% (U1), $\leq$ 10% (U2)	UFSAR Section 15.6
Plant Initial Operating Conditions		
Reactor Power	$\leq$ 102% of 3586.6* MWt with 2% Calorimetric Uncertainty	UFSAR Section 15.6
Peaking Factors	Fq=2.60, $F\Delta H$=1.70	COLR
Axial Power Distribution	Figure 1	UFSAR Section 15.6
Fluids Conditions		
RCS Average Temperature (T_{avg})	565 °F to 598 °F	UFSAR Section 15.6
Pressurizer Pressure	2207 to 2293 psia	UFSAR Section 15.6
Reactor Coolant Flow	$\geq$ 92000 gpm/loop	UFSAR Section 15.6
Accumulator Temperature	60 °F to 130 °F	UFSAR Section 15.6
Accumulator Pressure	602 to 677 psia	UFSAR Section 15.6
Accumulator Water Volume	920 to 980 ft^3	UFSAR Section 15.6
Accident Boundary Conditions		
Single Failure Assumptions	1 Train of ECCS Pumps	UFSAR Section 15.6
Safety Injection Flow	Table 3	UFSAR Section 15.6
Safety Injection Temperature	32 °F to 103 °F	UFSAR Section 15.6
Safety Injection Initiation Delay Time	$\leq$ 27 sec No-Loss of Off-Site Power (LOOP) $\leq$ 40 sec LOOP	UFSAR Section 15.6
Containment Pressure	Figure 2	UFSAR Section 15.6

* 3586.6 MWt conservatively chosen to bound operations at uprated power conditions.

Table 2

Byron Station and Braidwood Station
Best Estimate Large Break LOCA Results

Parameter	Value	Criteria
Calculated 50th Percentile PCT (°F) * (for time period of maximum 95th Percentile)		
Unit 1	1676	N/A
Unit 2	1756	N/A
Calculated 95th Percentile PCT (°F)*		
Unit 1	2044	2200
Unit 2	2088	2200
Maximum Cladding Oxidation (%)*	< 17	17
Maximum Hydrogen Generation (%)*	< 1	1
Coolable Geometry	Core Remains Coolable	Core Remains Coolable
Long Term Cooling	Core Remains Cool in Long Term	Core Remains Cool in Long Term

The Unit 1 and Unit 2 analyses were performed separately to reflect the different steam generator design of Unit 1 vice Unit 2. Note that the Unit 1 steam generators at Byron Station and Braidwood Station are of the same design; likewise, the Unit 2 steam generators at Byron Station and Braidwood Station are of the same design. The Byron Station and Braidwood Station reactor vessel designs are fundamentally identical for all units. There are, however, minor differences in the detailed design of certain components, which affect the metal mass and fluid volume in different regions of the vessel. These minor differences were addressed by use of the bounding plant design when there was a clear direction of conservatism (e.g., maximizing metal masses that contribute to lower plenum/downcomer heatup and boiling, and minimizing fluid volumes that contribute to core cooling). When the differences pertained to parameters judged to be inconsequential, a representative plant choice was used.

* To be documented in UFSAR Section 15.6, "Decrease in Reactor Coolant Inventory." These parameters have been calculated using the methodology in the following reference:

WCAP-12945-P-A, Volume 1 (Revision 2) and Volumes 2 through 5 (Revision 1), "Code Qualification Document for Best Estimate LOCA Analysis," March 1998.

Table 3

Byron Station and Braidwood Station
Best Estimate Large Break LOCA
Calculated Minimum Injected ECCS Flow into RCS
(Three Intact Loops)

RCS Pressure (psig)	Flow Rate (ft^3/sec)
0	7.292
20	6.758
40	6.010
60	4.827
80	3.335
105.3	1.504
200	1.429
300	1.352
400	1.274

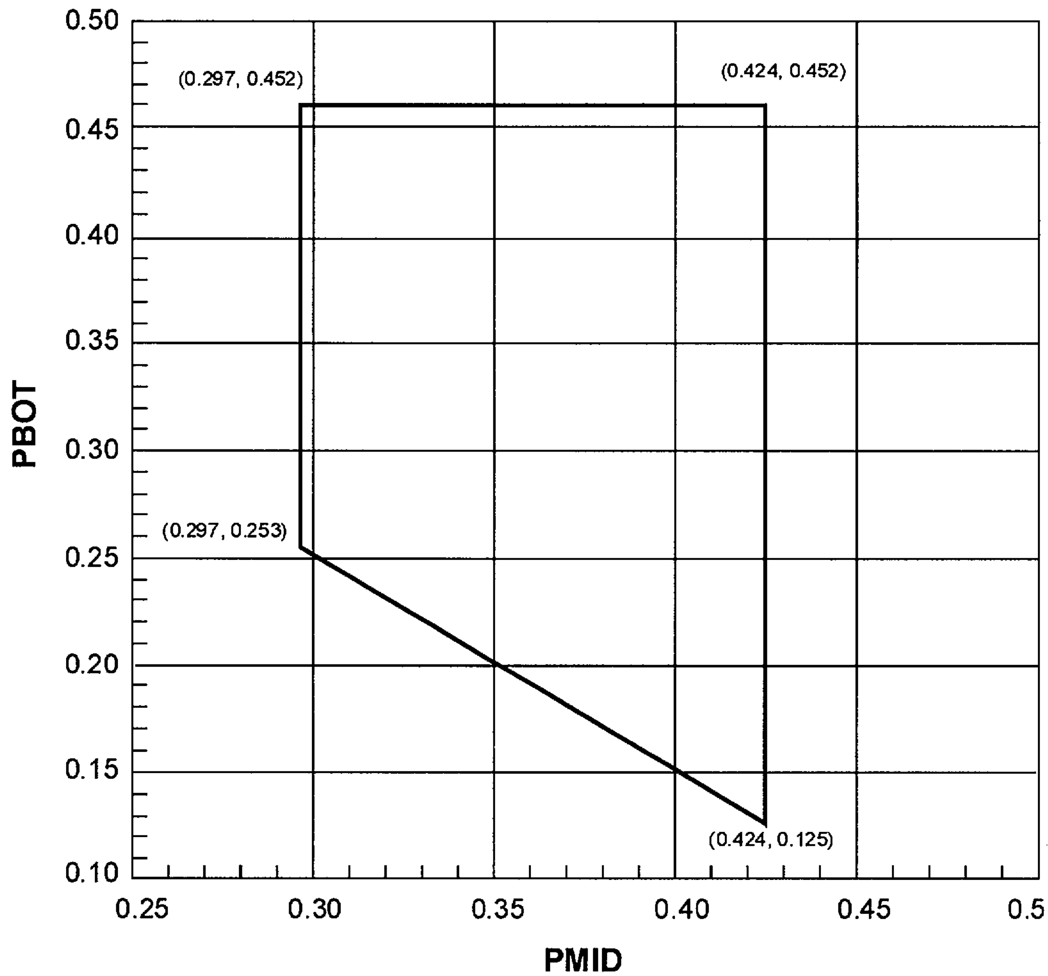
ATTACHMENT A

BYRON STATION, UNITS 1 AND 2
BRAIDWOOD STATION, UNITS 1 AND 2

DESCRIPTION AND SAFETY ANALYSIS FOR PROPOSED CHANGES

Figure 1

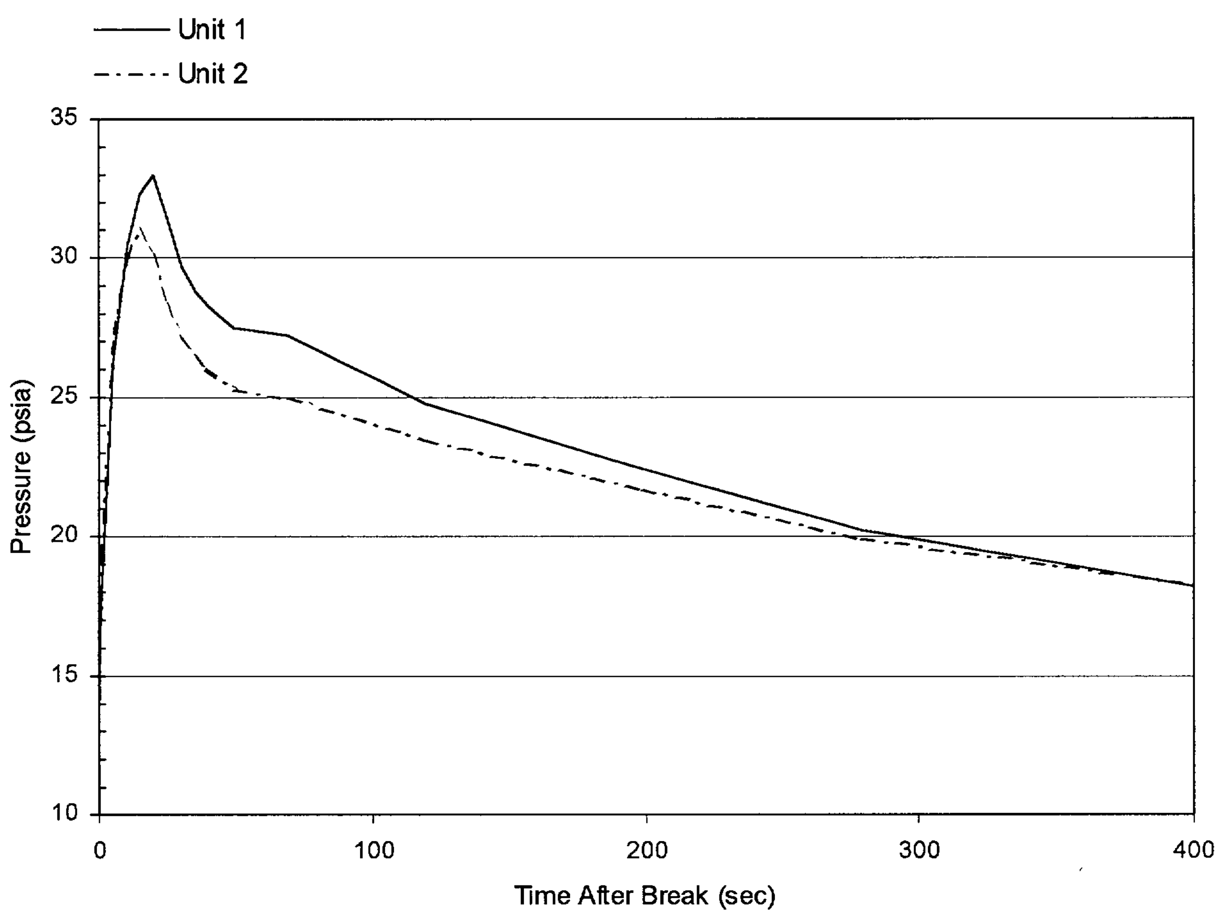
Byron Station and Braidwood Station
Best Estimate Large Break LOCA Analysis
Assumed Axial Power Shape Operating Space Envelope



PBOT – Power in the bottom one-third of the reactor core
PMID – Power in the middle one-third of the reactor core

Figure 2

Byron Station and Braidwood Station
Best Estimate Large Break LOCA Analysis
Minimum Calculated Containment Backpressure Boundary Condition



G. IMPACT ON PREVIOUS SUBMITTALS

All license amendment requests for the Byron Station and Braidwood Station, currently under review by the NRC, were evaluated to determine the potential impact from this submittal. Our July 5, 2000, submittal, "Request for a License Amendment to Permit Uprated Power Operations at Byron and Braidwood Stations," specified that the Large Break LOCA analysis in support of uprated power operations, would be submitted in a separate license amendment request; consequently, this submittal directly supports our July 5, 2000, license amendment request. No other pending license amendment requests are affected by this submittal.

H. SCHEDULE REQUIREMENTS

We request that the NRC review and approve the proposed TS changes by May 7, 2001, concurrent with our July 5, 2000, license amendment request, to support uprated power operations of Byron Station, Unit 1 and Unit 2, and Braidwood Station, Unit 1 and Unit 2, prior to the summer months of 2001.

ATTACHMENT B-1

MARKED-UP PAGES FOR PROPOSED CHANGES

BYRON STATION

<u>REVISED TS PAGES</u>

5.6-4

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

6. ~~WCAP-9220-P-A, "Westinghouse ECCS Evaluation Model-1981 Version," February 1982.~~

WCAP-12945-P-A, Volume 1, Revision 2, and Volumes 2 through 5, Revision 1, "Code Qualification Document for Best Estimate LOCA Analysis," March 1998.

~~7.~~ ~~WCAP-9561-P-A, Add. 3, "BART A-1: a Computer Code for Best Estimate Analysis of Reflood Transients - Special Report: Thimble Modeling in Westinghouse ECCS Evaluation Model," July 1986.~~

~~8.~~ ~~WCAP-10266-P-A, "The 1981 Version of Westinghouse Evaluation Model using BASH Code," March 1987, including Addendum 1 "Power Shape Sensitivity Studies," Revision 2-P-A, dated December 15, 1987, and Addendum 2 "BASH Methodology Improvements and Reliability Enhancements," Revision 2, Dated May 1988.~~

7. ~~9.~~ WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code," August 1985.

8. ~~10.~~ WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model using NOTRUMP Code," August 1985.

9. ~~11.~~ WCAP-10216-A, Revision 1, "Relaxation of Constant Axial Offset Control - F_Q Surveillance Technical Specification," February 1994.

10. ~~12.~~ WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986;

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met; and

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

ATTACHMENT B-2

MARKED-UP PAGES FOR PROPOSED CHANGES

BRAIDWOOD STATION

<u>REVISED TS PAGES</u>

5.6-4

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

6. ~~WCAP-9220-P-A, "Westinghouse ECCS Evaluation Model-1981 Version," February 1982.~~

~~7.~~ ~~WCAP-9561-P-A, Add. 3, "BART A-1: a Computer Code for Best Estimate Analysis of Reflood Transients - Special Report: Thimble Modeling in Westinghouse ECCS Evaluation Model," July 1986.~~

~~8.~~ ~~WCAP-10266-P-A, "The 1981 Version of Westinghouse Evaluation Model using BASH Code," March 1987, including Addendum 1 "Power Shape Sensitivity Studies," Revision 2-P-A, dated December 15, 1987, and Addendum 2 "BASH Methodology Improvements and Reliability Enhancements," Revision 2, Dated May 1988.~~

7. ~~9.~~ WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code," August 1985.

8. ~~10.~~ WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model using NOTRUMP Code," August 1985.

9. ~~11.~~ WCAP-10216-A, Revision 1, "Relaxation of Constant Axial Offset Control - F_Q Surveillance Technical Specification," February 1994.

10. ~~12.~~ WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986;

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met; and

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

WCAP-12945-P-A, Volume 1, Revision 2, and Volumes 2 through 5, Revision 1, "Code Qualification Document for Best Estimate LOCA Analysis," March 1998.

ATTACHMENT B-3

INCORPORATED PROPOSED CHANGES
TYPED PAGES

BYRON STATION

<u>REVISED TS PAGES</u>

5.6-4

<u>REVISED BASES PAGES</u>

B 3.2.1-2
B 3.2.1-4
B 3.2.2-2
B 3.2.4-1
B 3.5.1-4
B 3.5.1-5
B 3.5.1-6
B 3.5.1-7
B 3.5.1-9
B 3.5.2-4
B 3.5.2-5

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

6. WCAP-12945-P-A, Volume 1, Revision 2, and Volumes 2 through 5, Revision 1, "Code Qualification Document for Best Estimate LOCA Analysis," March 1998.

7. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code," August 1985.

8. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model using NOTRUMP Code," August 1985.

9. WCAP-10216-A, Revision 1, "Relaxation of Constant Axial Offset Control - F_Q Surveillance Technical Specification," February 1994.

10. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986;

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met; and

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

BASES

BACKGROUND (continued)

Core monitoring and control under non-equilibrium conditions are accomplished by operating the core within the limits of the appropriate LCOs, including the limits on AFD, QPTR, and control rod insertion.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a small break Loss Of Coolant Accident (LOCA) the peak cladding temperature must not exceed 2200°F and during a large break LOCA there must be a high level of probability that the peak cladding temperature does not exceed 2200°F (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 Departure from Nucleate Boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the prompt energy deposition to the fuel must not exceed 200 cal/gm (Ref. 2); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

Limits on $F_Q(Z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the peak cladding temperature is typically most limiting.

$F_Q(Z)$ limits assumed in the LOCA analysis are typically limiting relative to (i.e., lower than) the $F_Q(Z)$ limit assumed in safety analyses for other postulated accidents. Therefore, this LCO provides conservative limits for other postulated accidents.

$F_Q(Z)$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO (continued)

The expression for $F_Q^W(Z)$ is:

$$F_Q^W(Z) = F_Q^C(Z)\ W(Z)$$

where W(Z) is a cycle dependent function that accounts for power distribution transients encountered during normal operation. W(Z) is included in the COLR. The $F_Q^C(Z)$ is calculated at equilibrium conditions.

The $F_Q(Z)$ limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during a small break LOCA and assures with a high level of probability that the peak cladding temperature does not exceed 2200°F during a large break LOCA (Ref. 1).

This LCO requires operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA $F_Q(Z)$ limits. If $F_Q^C(Z)$ cannot be maintained within the LCO limits, reduction of the core power is required.

Violating the LCO limits for $F_Q(Z)$ may produce unacceptable consequences if a design basis event occurs while $F_Q(Z)$ is outside its specified limits.

APPLICABILITY

The $F_Q(Z)$ limits must be maintained in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.

BASES

BACKGROUND (continued)

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Limits on $F^N_{\Delta H}$ preclude core power distributions that exceed the following fuel design limits:

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

b. During a small break Loss Of Coolant Accident (LOCA) Peak Cladding Temperature (PCT) must not exceed 2200°F and during a large break LOCA there must be a high level of probability that PCT does not exceed 2200°F;

c. During an ejected rod accident, the prompt energy deposition to the fuel must not exceed 200 cal/gm (Ref. 1); and

d. Fuel design limits required by GDC 26 (Ref. 2) for the condition when control rods must be capable of shutting down the reactor with a minimum required Shutdown Margin with the highest worth control rod stuck fully withdrawn.

For transients that may be DNB limited, $F^N_{\Delta H}$ is a significant core parameter. The limits on $F^N_{\Delta H}$ ensure that the DNB design criterion is met for normal operation, operational transients, and any transients arising from events of moderate frequency. Refer to the Bases for LCO 3.4.1, "RCS Pressure, Temperature, and Flow DNB Limits," for a discussion of the applicable Departure from Nucleate Boiling Ratio (DNBR) limits.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE SAFETY ANALYSES

Limits on QPTR preclude core power distributions that violate the following fuel design criteria:

a. During a small break Loss Of Coolant Accident (LOCA) the Peak Cladding Temperature (PCT) must not exceed 2200°F and during a large break LOCA there must be a high level of probability that the PCT does not exceed 2200°F (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 Departure from Nucleate Boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the prompt energy deposition to the fuel must not exceed 200 cal/gm (Ref. 2); and

d. The control rods must be capable of shutting down the reactor with a minimum required Shutdown Margin with the highest worth control rod stuck fully withdrawn (Ref. 3).

BASES

APPLICABLE SAFETY ANALYSES (continued)

This LCO helps to ensure that the following acceptance criteria established for the ECCS by 10 CFR 50.46 (Ref. 4) will be met following a LOCA:

a. During a small break Loss Of Coolant Accident (LOCA) maximum fuel element cladding temperature is ≤ 2200°F and during a large break LOCA there must be a high level of probability that maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react; and

d. Core is maintained in a coolable geometry.

Since the accumulators discharge during the blowdown phase of a LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46.

For the small break LOCA analyses, a nominal contained accumulator water volume is used. For the large break LOCA analyses, a contained accumulator water volume range of 920 ft^3 - 980 ft^3 is used. The contained water volume is the same as the deliverable volume for the accumulators, since the accumulators are emptied, once discharged. For small breaks, the peak clad temperature is not sensitive to the accumulator water volume. For large breaks, there are two competing effects regarding accumulator water volume: the amount of water available for injection versus the injection rate. A higher water volume results in a larger total injection but at a slower injection rate. Conversely, a lower water volume results in a smaller total injection but at a faster infection rate.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Both the large and the small break LOCA analyses model the pipe water volume from the accumulator to the SI accumulator discharge header downstream cold leg injection check valve (SI8948). However, an evaluation was performed neglecting the pipe water volume between the SI accumulator discharge header upstream cold leg injection check valve (SI8956) to the SI accumulator discharge header downstream cold leg injection check valve (SI8948) to address gas accumulation. This evaluation determined that the impact on peak clad temperature was minimal for both the large break and the small break LOCA analyses. Since the range of the allowed accumulator volumes is relatively small and has a minimal effect on peak clad temperature, a nominal water volume is used in the small break LOCA analysis. The small break LOCA analysis assumes a nominal water volume of 7106 gallons based on the Technical Specification (TS) minimum and maximum limits of 6995 gallons (935 ft^3, 31% of indicated level) and 7217 gallons (965 ft^3, 63% of indicated level). The large break LOCA analysis assumes a water volume range of 6882 gallons (920 ft^3, 15% of indicated level) to 7331 gallons (980 ft^3, 79% of indicated level) which bounds the TS limits.

The minimum boron concentration setpoint is used in the post LOCA boron concentration calculation. The calculation is performed to assure reactor subcriticality in a post LOCA environment. Of particular interest is the large break LOCA, since no credit is taken for control rod assembly insertion. A reduction in the accumulator minimum boron concentration would produce a subsequent reduction in the available containment sump concentration for post LOCA shutdown and an increase in the maximum sump pH. The maximum boron concentration is used in determining the cold leg to hot leg recirculation injection switchover time and minimum sump pH.

The small break LOCA analyses are performed at the minimum nitrogen cover pressure, since sensitivity analyses have demonstrated that higher nitrogen cover pressure results in a computed peak clad temperature benefit. The large break LOCA analyses are performed at a nitrogen cover pressure range of 602 psia to 677 psia. The maximum nitrogen cover pressure limit prevents accumulator relief valve actuation, and ultimately preserves accumulator integrity.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The effects on containment mass and energy releases from the accumulators are accounted for in the appropriate analyses (Refs. 2 and 3).

The accumulators satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO establishes the minimum conditions required to ensure that the accumulators are available to accomplish their core cooling safety function following a LOCA. Four accumulators are required to ensure that 100% of the contents of three of the accumulators will reach the core during a LOCA. This is consistent with the assumption that the contents of one accumulator spill through the break. If less than three accumulators are injected during the blowdown phase of a LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 4) could be violated.

For an accumulator to be considered OPERABLE, the isolation valve must be fully open with power removed, a contained volume ≥ 31% and ≤ 63% (6995 gallons to 7217 gallons) with a boron concentration ≥ 2200 ppm and ≤ 2400 ppm, and a nitrogen cover pressure ≥ 602 and ≤ 647 psig, must be met.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS pressure > 1000 psig, the accumulator OPERABILITY requirements are based on full power operation. Although cooling requirements decrease as power decreases, the accumulators are still required to provide core cooling as long as elevated RCS pressures and temperatures exist.

This LCO is only applicable at pressures > 1000 psig. At pressures ≤ 1000 psig, the rate of RCS blowdown is such that the ECCS pumps can provide adequate injection to ensure that peak clad temperature remains below the 10 CFR 50.46 (Ref. 4) limit of 2200°F.

BASES

APPLICABILITY (continued)

In MODE 3, with RCS pressure ≤ 1000 psig, and in MODES 4, 5, and 6, the accumulator motor operated isolation valves are closed to isolate the accumulators from the RCS. This allows RCS cooldown and depressurization without discharging the accumulators into the RCS or requiring depressurization of the accumulators.

ACTIONS

A.1

If the boron concentration of one accumulator is not within limits, it must be returned to within the limits within 72 hours. In this Condition, ability to maintain subcriticality or minimum boron precipitation time may be reduced. The boron in the accumulators contributes to the assumption that the combined ECCS water in the partially recovered core during the early reflooding phase of a large break LOCA is sufficient to keep that portion of the core subcritical. One accumulator below the minimum boron concentration limit, however, will have no effect on available ECCS water and an insignificant effect on core subcriticality during reflood. Boiling of ECCS water in the core during reflood concentrates boron in the saturated liquid that remains in the core. In addition, current analysis demonstrates that the accumulators do not discharge following a large main steam line break. Thus, 72 hours is allowed to return the boron concentration to within limits.

B.1

If one accumulator is inoperable for a reason other than boron concentration, the accumulator must be returned to OPERABLE status within 1 hour. In this Condition, the required contents of three accumulators cannot be assumed to reach the core during a LOCA. Due to the severity of the consequences should a LOCA occur in these conditions, the 1 hour Completion Time to open the valve, remove power to the valve, or restore the proper water volume or nitrogen cover pressure ensures that prompt action will be taken to return the inoperable accumulator to OPERABLE status. The Completion Time minimizes the potential for exposure of the unit to a LOCA under these conditions.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.4

The boron concentration should be verified to be within required limits for each accumulator every 31 days since the static design of the accumulators limits the ways in which the concentration can be changed. The 31 day Frequency is adequate to identify changes that could occur from mechanisms such as stratification or inleakage.

SR 3.5.1.5

Sampling the affected accumulator within 6 hours after a 1% volume increase (nominally 70 gallons or 10% of indicated level) will identify whether inleakage has caused a reduction in boron concentration to below the required limit. It is not necessary to verify boron concentration of the accumulator after a 1% volume increase (10% indicated level increase) if the added water inventory is from the Refueling Water Storage Tank (RWST) and the boron concentration of the RWST is ≥ 2200 ppm and ≤ 2400 ppm. With the water contained in the RWST within the boron concentration requirements of the accumulators, any added inventory would not cause the accumulator's boron concentration to exceed the limits of this LCO.

With the only indication available to the operators in the control room being level indication in percent, a required accumulator volume increase of 1% or an increase of 10% of indicated level would require the accumulator to be sampled to verify the accumulator boron concentration is within the limits. The small break LOCA analysis assumes a nominal water volume of 7106 gallons based on the TS minimum and maximum limits of 6995 gallons (935 ft^3, 31% of indicated level) and 7217 gallons (965 ft^3, 63% of indicated level). These volumes are also indicated in the specific tank curves for the SI accumulators. The large break LOCA analysis assumes a water volume range of 6882 gallons (920 ft^3, 15% of indicated level) to 7331 gallons (980 ft^3, 79% of indicated level) which bounds the TS limits. The 10% indicated level increase is considered a conservative indication for a 70 gallon increase in the accumulator volume requiring an increase in the sampling requirement to verify accumulator boron concentration remains within the specified limits.

BASES

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 2), will be met following a LOCA:

a. During a small break LOCA maximum fuel element cladding temperature is ≤ 2200°F and during a large break LOCA there must be a high level of probability that maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

d. Core is maintained in a coolable geometry; and

e. Adequate long term core cooling capability is maintained.

The LCO also limits the potential for a post trip return to power following an MSLB event and ensures that containment temperature limits are met.

Each ECCS subsystem is taken credit for in a large break LOCA event at full power (Ref. 3). This event establishes the requirement for runout flow for the ECCS pumps, as well as the maximum response time for their actuation. The centrifugal charging pumps and SI pumps are credited in a small break LOCA event. This event establishes the flow and discharge head at the design point for the centrifugal charging pumps. The SGTR and MSLB events also credit the centrifugal charging pumps. The OPERABILITY requirements for the ECCS are based on the following LOCA analysis assumptions:

a. A large break LOCA event, with loss of offsite power and a single failure disabling one RHR pump (both emergency DG trains are assumed to operate due to requirements for modeling full active containment heat removal system operation); and

BASES

APPLICABLE SAFETY ANALYSES (continued)

b. A small break LOCA event, with a loss of offsite power and a single failure disabling one ECCS train.

During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding during large breaks or control rod insertion for small breaks. Following depressurization, emergency cooling water is injected into the cold legs, flows into the downcomer, fills the lower plenum, and refloods the core.

The effects on containment mass and energy releases are accounted for in appropriate analyses (Refs. 3 and 4). The LCO ensures that an ECCS train will deliver sufficient water to match boiloff rates soon enough to minimize the consequences of the core being uncovered following a large LOCA. It also ensures that the centrifugal charging and SI pumps will deliver sufficient water and boron during a small LOCA to maintain core subcriticality. For smaller LOCAs, the centrifugal charging pump delivers sufficient fluid to maintain RCS inventory. For a small break LOCA, the steam generators continue to serve as the heat sink, providing part of the required core cooling.

The ECCS trains satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

In MODES 1, 2, and 3, two independent (and redundant) ECCS trains are required to ensure that sufficient ECCS flow is available, assuming a single failure affecting either train. Additionally, individual components within the ECCS trains may be called upon to mitigate the consequences of other transients and accidents.

In MODES 1, 2, and 3, an ECCS train consists of a centrifugal charging subsystem, an SI subsystem, and an RHR subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an SI signal and automatically transferring suction to the containment sump.

ATTACHMENT B-4

INCORPORATED PROPOSED CHANGES
TYPED PAGES

BRAIDWOOD STATION

<u>REVISED TS PAGES</u>

5.6-4

<u>REVISED BASES PAGES</u>

B 3.2.1-2
B 3.2.1-4
B 3.2.2-2
B 3.2.4-1
B 3.5.1-4
B 3.5.1-5
B 3.5.1-6
B 3.5.1-7
B 3.5.1-9
B 3.5.2-4
B 3.5.2-5

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

6. WCAP-12945-P-A, Volume 1, Revision 2, and Volumes 2 through 5, Revision 1, "Code Qualification Document for Best Estimate LOCA Analysis," March 1998.

7. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code," August 1985.

8. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model using NOTRUMP Code," August 1985.

9. WCAP-10216-A, Revision 1, "Relaxation of Constant Axial Offset Control - F_Q Surveillance Technical Specification," February 1994.

10. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986;

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met; and

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

BASES

BACKGROUND (continued)

Core monitoring and control under non-equilibrium conditions are accomplished by operating the core within the limits of the appropriate LCOs, including the limits on AFD, QPTR, and control rod insertion.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a small break Loss Of Coolant Accident (LOCA) the peak cladding temperature must not exceed 2200°F and during a large break LOCA there must be a high level of probability that the peak cladding temperature does not exceed 2200°F (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 Departure from Nucleate Boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the prompt energy deposition to the fuel must not exceed 200 cal/gm (Ref. 2); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

Limits on $F_Q(Z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the peak cladding temperature is typically most limiting.

$F_Q(Z)$ limits assumed in the LOCA analysis are typically limiting relative to (i.e., lower than) the $F_Q(Z)$ limit assumed in safety analyses for other postulated accidents. Therefore, this LCO provides conservative limits for other postulated accidents.

$F_Q(Z)$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO (continued)

The expression for $F_Q^W(Z)$ is:

$$F_Q^W(Z) = F_Q^C(Z)\ W(Z)$$

where W(Z) is a cycle dependent function that accounts for power distribution transients encountered during normal operation. W(Z) is included in the COLR. The $F_Q^C(Z)$ is calculated at equilibrium conditions.

The $F_Q(Z)$ limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during a small break LOCA and assures with a high level of probability that the peak cladding temperature does not exceed 2200°F during a large break LOCA (Ref. 1).

This LCO requires operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA $F_Q(Z)$ limits. If $F_Q^C(Z)$ cannot be maintained within the LCO limits, reduction of the core power is required.

Violating the LCO limits for $F_Q(Z)$ may produce unacceptable consequences if a design basis event occurs while $F_Q(Z)$ is outside its specified limits.

APPLICABILITY

The $F_Q(Z)$ limits must be maintained in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.

BASES

BACKGROUND (continued)

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Limits on $F^N_{\Delta H}$ preclude core power distributions that exceed the following fuel design limits:

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

b. During a small break Loss Of Coolant Accident (LOCA) Peak Cladding Temperature (PCT) must not exceed 2200°F and during a large break LOCA there must be a high level of probability that PCT does not exceed 2200°F;

c. During an ejected rod accident, the prompt energy deposition to the fuel must not exceed 200 cal/gm (Ref. 1); and

d. Fuel design limits required by GDC 26 (Ref. 2) for the condition when control rods must be capable of shutting down the reactor with a minimum required Shutdown Margin with the highest worth control rod stuck fully withdrawn.

For transients that may be DNB limited, $F^N_{\Delta H}$ is a significant core parameter. The limits on $F^N_{\Delta H}$ ensure that the DNB design criterion is met for normal operation, operational transients, and any transients arising from events of moderate frequency. Refer to the Bases for LCO 3.4.1, "RCS Pressure, Temperature, and Flow DNB Limits," for a discussion of the applicable Departure from Nucleate Boiling Ratio (DNBR) limits.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE SAFETY ANALYSES

Limits on QPTR preclude core power distributions that violate the following fuel design criteria:

a. During a small break Loss Of Coolant Accident (LOCA) the Peak Cladding Temperature (PCT) must not exceed 2200°F and during a large break LOCA there must be a high level of probability that the PCT does not exceed 2200°F (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 Departure from Nucleate Boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the prompt energy deposition to the fuel must not exceed 200 cal/gm (Ref. 2); and

d. The control rods must be capable of shutting down the reactor with a minimum required Shutdown Margin with the highest worth control rod stuck fully withdrawn (Ref. 3).

BASES

APPLICABLE SAFETY ANALYSES (continued)

This LCO helps to ensure that the following acceptance criteria established for the ECCS by 10 CFR 50.46 (Ref. 4) will be met following a LOCA:

a. During a small break Loss Of Coolant Accident (LOCA) maximum fuel element cladding temperature is ≤ 2200°F and during a large break LOCA there must be a high level of probability that maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react; and

d. Core is maintained in a coolable geometry.

Since the accumulators discharge during the blowdown phase of a LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46.

For the small break LOCA analyses, a nominal contained accumulator water volume is used. For the large break LOCA analyses, a contained accumulator water volume range of 920 ft^3 - 980 ft^3 is used. The contained water volume is the same as the deliverable volume for the accumulators, since the accumulators are emptied, once discharged. For small breaks, the peak clad temperature is not sensitive to the accumulator water volume. For large breaks, there are two competing effects regarding accumulator water volume: the amount of water available for injection versus the injection rate. A higher water volume results in a larger total injection but at a slower injection rate. Conversely, a lower water volume results in a smaller total injection but at a faster infection rate.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Both the large and the small break LOCA analyses model the pipe water volume from the accumulator to the SI accumulator discharge header downstream cold leg injection check valve (SI8948). However, an evaluation was performed neglecting the pipe water volume between the SI accumulator discharge header upstream cold leg injection check valve (SI8956) to the SI accumulator discharge header downstream cold leg injection check valve (SI8948) to address gas accumulation. This evaluation determined that the impact on peak clad temperature was minimal for both the large break and the small break LOCA analyses. Since the range of the allowed accumulator volumes is relatively small and has a minimal effect on peak clad temperature, a nominal water volume is used in the small break LOCA analysis. The small break LOCA analysis assumes a nominal water volume of 7106 gallons based on the Technical Specification (TS) minimum and maximum limits of 6995 gallons (935 ft^3, 31% of indicated level) and 7217 gallons (965 ft^3, 63% of indicated level). The large break LOCA analysis assumes a water volume range of 6882 gallons (920 ft^3, 15% of indicated level) to 7331 gallons (980 ft^3, 79% of indicated level) which bounds the TS limits.

The minimum boron concentration setpoint is used in the post LOCA boron concentration calculation. The calculation is performed to assure reactor subcriticality in a post LOCA environment. Of particular interest is the large break LOCA, since no credit is taken for control rod assembly insertion. A reduction in the accumulator minimum boron concentration would produce a subsequent reduction in the available containment sump concentration for post LOCA shutdown and an increase in the maximum sump pH. The maximum boron concentration is used in determining the cold leg to hot leg recirculation injection switchover time and minimum sump pH.

The small break LOCA analyses are performed at the minimum nitrogen cover pressure, since sensitivity analyses have demonstrated that higher nitrogen cover pressure results in a computed peak clad temperature benefit. The large break LOCA analyses are performed at a nitrogen cover pressure range of 602 psia to 677 psia. The maximum nitrogen cover pressure limit prevents accumulator relief valve actuation, and ultimately preserves accumulator integrity.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The effects on containment mass and energy releases from the accumulators are accounted for in the appropriate analyses (Refs. 2 and 3).

The accumulators satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO establishes the minimum conditions required to ensure that the accumulators are available to accomplish their core cooling safety function following a LOCA. Four accumulators are required to ensure that 100% of the contents of three of the accumulators will reach the core during a LOCA. This is consistent with the assumption that the contents of one accumulator spill through the break. If less than three accumulators are injected during the blowdown phase of a LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 4) could be violated.

For an accumulator to be considered OPERABLE, the isolation valve must be fully open with power removed, a contained volume ≥ 31% and ≤ 63% (6995 gallons to 7217 gallons) with a boron concentration ≥ 2200 ppm and ≤ 2400 ppm, and a nitrogen cover pressure ≥ 602 and ≤ 647 psig, must be met.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS pressure > 1000 psig, the accumulator OPERABILITY requirements are based on full power operation. Although cooling requirements decrease as power decreases, the accumulators are still required to provide core cooling as long as elevated RCS pressures and temperatures exist.

This LCO is only applicable at pressures > 1000 psig. At pressures ≤ 1000 psig, the rate of RCS blowdown is such that the ECCS pumps can provide adequate injection to ensure that peak clad temperature remains below the 10 CFR 50.46 (Ref. 4) limit of 2200°F.

BASES

APPLICABILITY (continued)

In MODE 3, with RCS pressure ≤ 1000 psig, and in MODES 4, 5, and 6, the accumulator motor operated isolation valves are closed to isolate the accumulators from the RCS. This allows RCS cooldown and depressurization without discharging the accumulators into the RCS or requiring depressurization of the accumulators.

ACTIONS

A.1

If the boron concentration of one accumulator is not within limits, it must be returned to within the limits within 72 hours. In this Condition, ability to maintain subcriticality or minimum boron precipitation time may be reduced. The boron in the accumulators contributes to the assumption that the combined ECCS water in the partially recovered core during the early reflooding phase of a large break LOCA is sufficient to keep that portion of the core subcritical. One accumulator below the minimum boron concentration limit, however, will have no effect on available ECCS water and an insignificant effect on core subcriticality during reflood. Boiling of ECCS water in the core during reflood concentrates boron in the saturated liquid that remains in the core. In addition, current analysis demonstrates that the accumulators do not discharge following a large main steam line break. Thus, 72 hours is allowed to return the boron concentration to within limits.

B.1

If one accumulator is inoperable for a reason other than boron concentration, the accumulator must be returned to OPERABLE status within 1 hour. In this Condition, the required contents of three accumulators cannot be assumed to reach the core during a LOCA. Due to the severity of the consequences should a LOCA occur in these conditions, the 1 hour Completion Time to open the valve, remove power to the valve, or restore the proper water volume or nitrogen cover pressure ensures that prompt action will be taken to return the inoperable accumulator to OPERABLE status. The Completion Time minimizes the potential for exposure of the unit to a LOCA under these conditions.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.4

The boron concentration should be verified to be within required limits for each accumulator every 31 days since the static design of the accumulators limits the ways in which the concentration can be changed. The 31 day Frequency is adequate to identify changes that could occur from mechanisms such as stratification or inleakage.

SR 3.5.1.5

Sampling the affected accumulator within 6 hours after a 1% volume increase (nominally 70 gallons or 10% of indicated level) will identify whether inleakage has caused a reduction in boron concentration to below the required limit. It is not necessary to verify boron concentration of the accumulator after a 1% volume increase (10% indicated level increase) if the added water inventory is from the Refueling Water Storage Tank (RWST) and the boron concentration of the RWST is ≥ 2200 ppm and ≤ 2400 ppm. With the water contained in the RWST within the boron concentration requirements of the accumulators, any added inventory would not cause the accumulator's boron concentration to exceed the limits of this LCO.

With the only indication available to the operators in the control room being level indication in percent, a required accumulator volume increase of 1% or an increase of 10% of indicated level would require the accumulator to be sampled to verify the accumulator boron concentration is within the limits. The small break LOCA analysis assumes a nominal water volume of 7106 gallons based on the TS minimum and maximum limits of 6995 gallons (935 ft^3, 31% of indicated level) and 7217 gallons (965 ft^3, 63% of indicated level). These volumes are also indicated in the specific tank curves for the SI accumulators. The large break LOCA analysis assumes a water volume range of 6882 gallons (920 ft^3, 15% of indicated level) to 7331 gallons (980 ft^3, 79% of indicated level) which bounds the TS limits. The 10% indicated level increase is considered a conservative indication for a 70 gallon increase in the accumulator volume requiring an increase in the sampling requirement to verify accumulator boron concentration remains within the specified limits.

BASES

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 2), will be met following a LOCA:

a. During a small break LOCA maximum fuel element cladding temperature is ≤ 2200°F and during a large break LOCA there must be a high level of probability that maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

d. Core is maintained in a coolable geometry; and

e. Adequate long term core cooling capability is maintained.

The LCO also limits the potential for a post trip return to power following an MSLB event and ensures that containment temperature limits are met.

Each ECCS subsystem is taken credit for in a large break LOCA event at full power (Ref. 3). This event establishes the requirement for runout flow for the ECCS pumps, as well as the maximum response time for their actuation. The centrifugal charging pumps and SI pumps are credited in a small break LOCA event. This event establishes the flow and discharge head at the design point for the centrifugal charging pumps. The SGTR and MSLB events also credit the centrifugal charging pumps. The OPERABILITY requirements for the ECCS are based on the following LOCA analysis assumptions:

a. A large break LOCA event, with loss of offsite power and a single failure disabling one RHR pump (both emergency DG trains are assumed to operate due to requirements for modeling full active containment heat removal system operation); and

BASES

APPLICABLE SAFETY ANALYSES (continued)

b. A small break LOCA event, with a loss of offsite power and a single failure disabling one ECCS train.

During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding during large breaks or control rod insertion for small breaks. Following depressurization, emergency cooling water is injected into the cold legs, flows into the downcomer, fills the lower plenum, and refloods the core.

The effects on containment mass and energy releases are accounted for in appropriate analyses (Refs. 3 and 4). The LCO ensures that an ECCS train will deliver sufficient water to match boiloff rates soon enough to minimize the consequences of the core being uncovered following a large LOCA. It also ensures that the centrifugal charging and SI pumps will deliver sufficient water and boron during a small LOCA to maintain core subcriticality. For smaller LOCAs, the centrifugal charging pump delivers sufficient fluid to maintain RCS inventory. For a small break LOCA, the steam generators continue to serve as the heat sink, providing part of the required core cooling.

The ECCS trains satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

In MODES 1, 2, and 3, two independent (and redundant) ECCS trains are required to ensure that sufficient ECCS flow is available, assuming a single failure affecting either train. Additionally, individual components within the ECCS trains may be called upon to mitigate the consequences of other transients and accidents.

In MODES 1, 2, and 3, an ECCS train consists of a centrifugal charging subsystem, an SI subsystem, and an RHR subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an SI signal and automatically transferring suction to the containment sump.

ATTACHMENT C

INFORMATION SUPPORTING A FINDING OF NO SIGNIFICANT HAZARDS CONSIDERATION

According to 10CFR 50.92 (c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

> Involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated;
>
> Create the possibility of a new or different kind of accident from any previously analyzed, or;
>
> Involve a significant reduction in the margin of safety.

In support of this determination, an evaluation of each of the three criteria set forth in 10 CFR 50.92 is provided below regarding the proposed license amendment.

Overview

Commonwealth Edison (ComEd) Company is requesting changes to Appendix A, Technical Specifications (TS) of Facility Operating License Nos. NPF-72, NPF-77, NPF-37 and NPF-66 for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, respectively. The proposed changes revise Technical Specifications (TS) 5.6.5, "Core Operating Limits Report (COLR)," to reference the appropriate Westinghouse topical report that describes the Best Estimate Large Break Loss of Coolant Accident (LOCA) analysis methodology. It has been determined that application of this methodology is appropriate for each unit at Byron Station and Braidwood Station. This analysis was performed using the NRC approved Westinghouse Best Estimate LOCA model WCOBRA/TRAC. The NRC approved this methodology in a letter from R. C. Jones, NRC, to N. J. Liparulo, Westinghouse Electric Corporation, "Acceptance for Referencing of the Topical Report WCAP-12945 (P), 'Westinghouse Code Qualification Document For Best Estimate Loss of Coolant Analysis,'" dated June 28, 1996. WCOBRA/TRAC is an analysis tool that models the reactor core thermal hydraulic behavior for a spectrum of hypothetical loss of coolant accidents. This analysis is being performed to support operation at uprated power conditions as noted in a letter from R. M. Krich (ComEd) to the NRC, "Request for a License Amendment to Permit Uprated Power Operations at Byron And Braidwood Stations," dated July 5, 2000. This amendment request proposed to increase the licensed reactor power from 3411 Megawatts-thermal (MWt) to 3586.6 MWt for Units 1 and 2 at each station.

The proposed Technical Specifications (TS) changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

No physical plant changes are being made as a result of using the Westinghouse Best Estimate Large Break LOCA analysis methodology. The proposed TS changes simply involve updating the references in TS 5.6.5.b, "Core Operating Limits Report (COLR)," to reference the Westinghouse Best Estimate Large Break LOCA analysis methodology (i.e., Westinghouse topical report, WCAP-12945-P-A, Volume 1, Revision 2, and Volumes 2 through 5, Revision 1, "Code Qualification Document for Best Estimate LOCA Analysis," March 1998). The plant conditions assumed in the analysis are bounded by the design conditions for all equipment in the plant; therefore, there will be no increase in the probability of a LOCA. The consequences of a LOCA are not being increased, since the analysis has shown that the Emergency Core Cooling System (ECCS) is designed such that its calculated cooling performance conforms to the criteria contained in 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors." Furthermore, the re-performance of the Large Break LOCA analysis has no effect on the performance of the ECCS equipment. No other accident consequence is potentially affected by this change.

All systems will continue to be operated in accordance with current design requirements under the new analysis, therefore no new components or system interactions have been identified that could lead to an increase in the probability of any accident previously evaluated in the Updated Final Safety Analysis Report (UFSAR). No changes were required to the Reactor Protection System (RPS) or Engineered Safety Features (ESF) setpoints because of the new analysis methodology.

Based on the analysis, it is concluded that the proposed TS changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

There are no physical changes being made to the plant as a result of using the Westinghouse Best Estimate Large Break LOCA analysis methodology. No new modes of plant operation are being introduced. The configuration, operation and accident response of the Byron Station and the Braidwood Station systems, structures or components are unchanged by utilization of the new analysis methodology. Analyses of transient events have confirmed that no transient event results in a new sequence of events that could lead to a new accident scenario. The parameters assumed in the analysis are within the design limits of existing plant equipment.

In addition, employing the Westinghouse Best Estimate Large Break LOCA analysis methodology does not create any new failure modes that could lead to a different kind of accident. The design of all systems remains unchanged and no new equipment or systems have been installed which could potentially introduce new failure modes or accident sequences. No changes have been made to any RPS or ESF actuation setpoints.

Based on this review, it is concluded that no new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed changes. Therefore, the proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

It has been shown that the analytic technique used in the Westinghouse Best Estimate Large Break LOCA analysis methodology realistically describes the expected behavior of the Byron Station and Braidwood Station reactor system during a postulated LOCA. Uncertainties have been accounted for as required by 10 CFR 50.46. A sufficient number of

LOCAs with different break sizes, different locations, and other variations in properties have been considered to provide assurance that the most severe postulated LOCAs have been evaluated. The analysis has demonstrated that there is a high probability that all acceptance criteria contained in 10 CFR 50.46 paragraph b continue to be satisfied.

Based on this review, the proposed TS changes do not involve a significant reduction in the margin of safety.

Overall Conclusion

Based upon the above analyses and evaluations, we have concluded that the proposed changes to the TS involve no significant hazards consideration.

ATTACHMENT D

INFORMATION SUPPORTING AN ENVIRONMENTAL ASSESSMENT

Commonwealth Edison (ComEd) Company is requesting changes to Appendix A, Technical Specifications (TS) of Facility Operating License Nos. NPF-72, NPF-77, NPF-37 and NPF-66 for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, respectively. The requested change proposes to revise Technical Specification 5.6.5, "Core Operating Limits Report (COLR)," to reference the appropriate Westinghouse topical report that describes the Best Estimate Large Break Loss of Coolant Accident (LOCA) analysis methodology. It has been determined that application of this methodology is appropriate for each unit at Byron Station and Braidwood Station. This analysis was performed using the NRC approved Westinghouse Best Estimate LOCA model WCOBRA/TRAC. The NRC approved this methodology in a letter from R. C. Jones, NRC, to N. J. Liparulo, Westinghouse Electric Corporation, "Acceptance for Referencing of the Topical Report WCAP-12945 (P), 'Westinghouse Code Qualification Document For Best Estimate Loss of Coolant Analysis,'" dated June 28, 1996. WCOBRA/TRAC is an analysis tool that models the reactor core thermal hydraulic behavior for a spectrum of hypothetical loss of coolant accidents. This analysis is being performed to support operation at uprated power conditions as noted in a letter from R. M. Krich (ComEd) to the NRC, "Request for a License Amendment to Permit Uprated Power Operations at Byron And Braidwood Stations," dated July 5, 2000. This amendment request proposed to increase the licensed reactor power from 3411 Megawatts-thermal (MWt) to 3586.6 MWt for Units 1 and 2 at each station.

ComEd has evaluated this proposed operating license amendment consistent with 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 change meets the criteria for a categorical exclusion set forth in paragraph (c)(9) of 10 CFR 51.22, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," and as such, has determined that no irreversible consequences exist in accordance with paragraph (b) of 10 CFR 50.92, "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," that changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, "Standards for Protection Against Radiation," or that changes an inspection or a surveillance requirement, and the proposed amendment meets the following specific criteria.

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

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

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

Non-Radiological Effluent Releases

Utilization of the Westinghouse Best Estimate Large Break LOCA analysis methodology will have no effect on the type or amount of non-radiological effluent releases and will have no effect on effluent discharge permit limitations or other conditions associated with the operation of the plant. None of the data contained in the Environmental Report or in the latest National Pollutant Discharge Elimination System (NPDES) Permits will be affected. The new methodology is an analysis tool that models the reactor core thermal hydraulic behavior for a spectrum of hypothetical loss of coolant accidents; consequently, there is no significant change in the types or a significant increase in the amounts of non-radiological effluents that may be released offsite.

Radiological Effluent Releases

Utilization of the Westinghouse Best Estimate Large Break LOCA analysis methodology will have no physical effect on the type or amount of liquid, solid or gaseous radiological effluent releases of the plant. The new methodology is an analysis tool that models the reactor core thermal hydraulic behavior for a spectrum of hypothetical loss of coolant accidents. The analysis results have indicated no change from those previously analyzed; consequently, there is no significant change in the types or a significant increase in the amounts of radiological effluents that may be released offsite.

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

Utilization of the Westinghouse Best Estimate Large Break LOCA analysis methodology will have no effect on the level of controls or methodology used for personnel radiation protection, the processing of radioactive effluents or handling of solid radioactive waste. In addition, employing the new methodology does not result in a significant change in the normal radiation levels within the plant. It is therefore concluded that there will be no significant increase in individual or cumulative occupational radiation exposure resulting from this change.

Overall Conclusion

Based on this review, it has been determined that the proposed change meets the criteria for a categorical exclusion set forth in paragraph (c)(9) of 10 CFR 51.22, and as such, has determined that no irreversible consequences exist in accordance with paragraph (b) of 10 CFR 50.92.