ATTACHMENT B

BRAIDWOOD STATION

Proposed Changes to Appendix A
Technical Specifications of facility
Operating Licenses NPF-72 and NPF-77

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2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB. This relation has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, and is indicative of the margin to DNB.

The DNBR thermal design criterion is that the probability that DNB will not occur on the most limiting rod is at least 95% (at a 95% confidence level) for any Condition I or II event.

In meeting this design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered. As described in the UFSAR, the effects of these uncertainties have been statistically combined with the correlation uncertainty. Design limit DNBR values have been determined that satisfy the DNB design criterion.

~~The design DNBR values are 1.34 and 1.32 for a typical cell and a thimble cell, respectively for OFA* fuel, and 1.33 for a typical cell and 1.32 for a thimble cell for the VANTAGE 5 fuel (1.25 for the typical and thimble cells)**. In addition, margin has been maintained in both designs by meeting safety analysis DNBR limits of 1.49 for a typical cell and 1.47 for a thimble cell for OFA fuel, and 1.67 and 1.65 for a typical cell and a thimble cell, respectively for the VANTAGE 5 fuel (1.50 for the typical and thimble cells)** in performing safety analyses.~~

The curves of Figure 2.1-1 (Figure 2.1-1a)** show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum design DNBR is no less than the design DNBR value, or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

Margin is maintained by meeting safety analysis DNBR limits.

*Optimized Fuel Assemblies

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

SAFETY LIMITS

BASES

REACTOR CORE (Continued)

These curves are based on ~~an~~ the value of the limiting enthalpy hot channel factor, $F^N_{\Delta H}$, with an ~~of 1.49 for OFA fuel and 1.59 for VANTAGE 5 fuel.~~ An allowance ~~is~~ included to ~~for an~~ increase ~~in~~ $F^N_{\Delta H}$ at reduced power~~, based on the expression:~~

~~$F^N_{\Delta H}$ = 1.49 [1+ 0.3 (1-P)] for OFA fuel~~

~~$F^N_{\Delta H}$ = 1.59 [1+ 0.3 (1-P)] for VANTAGE 5 fuel~~

~~Where P is the fraction of RATED THERMAL POWER.~~

The~~se~~ limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance affect on the Overtemperature ΔT trips will reduce the Setpoints to provide protection consistent with core Safety Limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System (RCS) from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor vessel, pressurizer, and the RCS piping, valves, and fittings are designed to Section III of the ASME Code for Nuclear Power Plants which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated Code requirements.

The entire RCS is hydrotested at 3110 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

c. The control rod insertion limits of Specification 3.1.3.6 are maintained, and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided the Conditions a. through d. above are maintained. The combination of the RCS flow requirement 390,400 gpm (371,400 gpm)* and the requirement on $F^N_{\Delta H}$ guarantee that the DNBR used in the safety analysis will be met.

Margin between the safety analysis limit DNBRs ~~[1.49 and 1.47 for the OFA fuel typical and thimble cells, respectively and 1.67 and 1.65 for the VANTAGE 5 typical and thimble cells (1.50 for the typical and thimble cells)*]~~ and the design limit DNBRs ~~[1.34 and 1.32 for the OFA fuel typical and thimble cells, and 1.33 and 1.32 for the VANTAGE 5 fuel typical and thimble cells, respectively (1.25 for the typical and thimble cells)*]~~ is maintained.

A fraction of this margin is utilized to accommodate ~~the transition core DNBR penalty (maximum of 12.5%) and~~ the appropriate fuel rod bow DNBR penalty (less than 1.5% per WCAP-8691, Revision 1). The rest of the margin between design and safety analysis DNBR limits can be used for plant design flexibility.

The RCS flow requirement is based on the loop minimum measured flow rate of 97,600 gpm (92,850 gpm)* which is used in the ~~Improved Thermal Design Procedure~~ (Revised Thermal Design Procedure)* described in UFSAR 4.4.1 and 15.0.3. A precision heat balance is performed once each cycle and is used to calibrate the RCS flow rate indicators. Potential fouling of the feedwater venturi, which might not be detected, could bias the results from the precision heat balance in a non-conservative manner. Therefore, a penalty of 0.1% is assessed for potential feedwater venturi fouling. A maximum measurement uncertainty of 2.2% (3.5%)* has been included in the loop minimum measured flow rate to account for potential undetected feedwater venturi fouling and the use of the RCS flow indicators for flow rate verification. Any fouling which might bias the RCS flow rate measurement greater than 0.1% can be detected by monitoring and trending various plant performance parameters. If detected, action shall be taken, before performing subsequent precision heat balance measurements, i.e., either the effect of fouling shall be quantified and compensated for in the RCS flow rate measurement, or the venturi shall be cleaned to eliminate the fouling.

Surveillance Requirement 4.2.3.4 provides adequate monitoring to detect possible flow reductions due to any rapid core crud buildup.

Surveillance Requirement 4.2.3.5 specifies that the measurement instrumentation shall be calibrated within seven days prior to the performance of the calorimetric flow measurement. This requirement is due to the fact that the drift effects of this instrumentation are not included in the flow measurement uncertainty analysis. This requirement does not apply for the instrumentation whose drift effects have been included in the uncertainty analysis.

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

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.6 The Annual Radiological Environmental Operating Report covering the operation of the facility 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 A Radioactive Effluent Release Report covering the operation of the facility during the previous year shall be submitted prior to May 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 facility. 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 B

~~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.~~

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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.

OPERATING LIMITS REPORT (Continued)

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."

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.

ATTACHMENT B

BYRON STATION

Proposed Changes to Appendix A
Technical Specifications of facility
Operating Licenses NPF-37 and NPF-66

Revised Pages:

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2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB. This relation has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, and is indicative of the margin to DNB.

The DNBR thermal design criterion is that the probability that DNB will not occur on the most limiting rod is at least 95% (at a 95% confidence level) for any Condition I or II event.

In meeting this design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered. As described in the UFSAR, the effects of these uncertainties have been statistically combined with the correlation uncertainty. Design limit DNBR values have been determined that satisfy the DNB design criterion.

~~The design DNBR values are 1.25 for the typical and thimble cells (1.34 and 1.32 for a typical cell and a thimble cell, respectively for OFA** fuel, and 1.33 for a typical cell and 1.32 for a thimble cell for the VANTAGE 5 fuel*). In addition, margin has been maintained in both designs by meeting safety analysis DNBR limits of 1.50 for the typical and thimble cells (1.49 for a typical cell and 1.47 for a thimble cell for OFA fuel, and 1.67 and 1.65 for a typical cell and a thimble cell, respectively for the VANTAGE 5 fuel*) in performing safety analyses.~~

The curves of Figure 2.1-1 (Figure 2.1-1a*) show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum design DNBR is no less than the design DNBR value, or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

Margin is maintained by meeting safety analysis DNBR limits

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*Not applicable to Unit 1. Applicable to Unit 2 until completion of cycle 5.
~~**Optimized fuel assembly.~~

REACTOR CORE (Continued)

These curves are based on ~~an~~ the value of the limiting enthalpy hot channel factor, $F^N_{\Delta H}$, with an ~~of 1.49 for OFA fuel and 1.59 for VANTAGE 5 fuel. An~~ allowance ~~is~~ included ~~for an~~ to increase ~~in~~ $F^N_{\Delta H}$ at reduced power ~~based on the expression:~~

$$\cancel{F^N_{\Delta H} = 1.49\ [1 + 0.3\ (1-P)]\quad \text{for OFA fuel}}$$

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~~Where P is the fraction of RATED THERMAL POWER.~~

The~~se~~ limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the Setpoints to provide protection consistent with core Safety Limits.

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HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

c. The control rod insertion limits of Specification 3.1.3.6 are maintained, and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided the Conditions a. through d. above are maintained. The combination of the RCS flow requirement [371,400 gpm (390,400 gpm*)] and the requirement on $F^N_{\Delta H}$ guarantee that the DNBR used in the safety analysis will be met.

Margin between the safety analysis limit DNBRs ~~[1.50 for the typical and thimble cells (1.49 and 1.47 for the OFA fuel typical and thimble cells, respectively and 1.67 and 1.65 for the VANTAGE 5 typical and thimble cells*)]~~ and the design limit DNBRs ~~[1.25 for the typical and thimble cells (1.34 and 1.32 for the OFA fuel typical and thimble cells, and 1.33 and 1.32 for the VANTAGE 5 fuel typical and thimble cells, respectively*)]~~ is maintained.

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The RCS flow requirement is based on the loop minimum measured flow rate of 92,850 gpm (97,600 gpm*) which is used in the Revised Thermal Design Procedure ~~(Improved Thermal Design Procedure~~ described in UFSAR 4.4.1 and 15.0.3~~)~~. A precision heat balance is performed once each cycle and is used to calibrate the RCS flow rate indicators. Potential fouling of the feedwater venturi, which might not be detected, could bias the results from the precision heat balance in a non-conservative manner. Therefore, a penalty of 0.1% is assessed for potential feedwater venturi fouling. A maximum measurement uncertainty of 3.5% (2.2%*) has been included in the loop minimum measured flow rate to account for potential undetected feedwater venturi fouling and the use of the RCS flow indicators for flow rate verification. Any fouling which might bias the RCS flow rate measurement greater than 0.1% can be detected by monitoring and trending various plant performance parameters. If detected, action shall be taken, before performing subsequent precision heat balance measurements, i.e., either the effect of fouling shall be quantified and compensated for in the RCS flow rate measurement, or the venturi shall be cleaned to eliminate the fouling.

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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*

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Insert B

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:~~

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OPERATING LIMITS REPORT (Continued)

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