Duke Power
A Duke Energy Company

W. R. McCollum, Jr.
Vice President

Duke Power
Oconee Nuclear Site
7800 Rochester Highway
Seneca, SC 29672
(864) 885-3107 OFFICE
(864) 885-3564 FAX

April 13, 2000

U. S. Nuclear Regulatory Commission
Washington, D. C. 20555-001

Attention: Document Control Desk

Subject: Oconee Nuclear Site, Units 1, 2, and 3
Docket Numbers 50-269, 50-270, and 50-287
License Amendment Request (TSC 2000-03),
Implementation of Mark-B11 Fuel with M5 Cladding.

References:

1) Letter, M. S. Tuckman. (Duke) to U. S. NRC Document Control Desk, "Oconee Nuclear Station Transition to Framatome MkB11 Fuel," December 2, 1997.

2) Letter, M. S. Tuckman. (Duke) to U. S. NRC Document Control Desk, "Transition to Mark-B11 Fuel with M5 Cladding," January 31, 2000.

3) Letter, D. E. LaBarge (NRC) to W. R. McCollum Jr. (Duke), "Exemption from Fuel Cladding Requirements (TAC Nosm MA6466, MA6467, and MA6468)," March 23,.2000.

4) Letter, S. A. Richards, (NRC) to T. A. Coleman (FCF), "Acceptance for Referencing of Licensing Topical Report BAW-10199P, Addendum 1, The BWU Critical Heat Flux Correlations (TAC No. M96728)," April 6, 2000.

5) Letter, S. N. Bailey (NRC) to T. A. Coleman (FCF), "Safety Evaluation for Topical Report BAW-10229P, Mark-B11 Fuel Assembly Design Topical Report, September 1997 (TAC No. M99904)," October 26, 1999.

A001

Pursuant to 10 CFR 50.90, Duke Energy Corporation (Duke) is submitting this license amendment request (LAR) for Facility Operating License Nos. DPR-38, DPR-47, and DPR-55 for Oconee Nuclear Station Units 1, 2, and 3 respectively. The LAR proposes to change the technical specifications and the bases to implement Mark-B11 fuel with M5 cladding. This LAR is required for the operation of Oconee Unit 1 Cycle 20 (O1C20). The associated design related information and the draft Core Operating Limits Report (COLR) for O1C20 are also provided as background information only.

On December 7, 1997, Duke informed the NRC of its intention to transition all three units at the Oconee Nuclear Station to Framatome Cogema Fuels Mark-B11 fuel (Reference 1). This letter identified the licensing submittals that were required to support implementation. On January 31, 2000, Duke updated the status of this plan, which included the implementation of M5 cladding (Reference 2). This letter provided the status of all of the supporting submittals and identified the remaining approvals required for Mark-B11 implementation. This LAR was identified as the remaining submittal required for implementation of Mark-B11 fuel with M5 cladding. The January 31, 2000 letter identified two NRC reviews that were in process and required for implementation of Mark-B11 fuel with M5 cladding. On March 23, 2000, the NRC granted the exemption request related to the use of M5 cladding (Reference 3). On April 6, 2000, the NRC approved the topical report on the BWU CHF correlation (Reference 4). As such, this LAR is not dependent on any additional NRC reviews or approvals.

The Oconee Nuclear Station will not be the first reactor to implement M5 cladding in the United States. The Davis Besse Nuclear Power Station is scheduled to load a full batch of fuel with M5 cladding this spring. This should establish a licensing precedence for this cladding material. O1C20 is; however, the first core to use the Mark-B11 fuel assembly design. This design has been reviewed and approved by the NRC (Reference 5).

This submittal contains the following attachments:

- Attachment I provides a marked up copy of the existing Oconee technical specifications and bases.

- Attachment II provides the reprinted Oconee technical specifications and bases and the instructions for updating the technical specifications and bases.

- Attachment III provides a description of the proposed changes and technical justification for the proposed changes.

- Attachment IV documents the determination that this LAR contains No Significant Hazards pursuant to 10 CFR 50.92.

- Attachment V provides the basis for categorical exclusion from performing an Environmental Assessment / Impact Statement pursuant to 10 CFR 51.22 (c) (9).

- Attachment VI provides the draft O1C20 COLR.

- Attachment VII provides design information relative to O1C20.

The UFSAR changes resulting from the changes proposed for Oconee 1 Cycle 20 are not included in this submittal since the NRC has indicated that these changes are not reviewed. The update of the Oconee UFSAR will be performed under 10 CFR 50.59 and implemented per the requirements of 10 CFR 50.71(e).

In accordance with Duke administrative procedures and the Duke Energy Quality Assurance Program Topical Report, the changes contained in this LAR have been reviewed and approved by the Oconee Plant Operations Review Committee and the Duke Corporate Nuclear Safety Review Board.

Pursuant to 10 CFR 50.91, a copy of this LAR is being sent to the State of South Carolina.

Duke is requesting that the NRC approve this LAR by November 15, 2000 to support the current Unit 1 outage schedule for the end of Cycle 19. Please address any comments or questions regarding this matter to Edwin D. Price Jr. at (864) 885-4388.

Very truly yours,

W. R. McCollum, Jr., Site Vice President
Oconee Nuclear Site

Attachments

xc: L. A. Reyes, Regional Administrator,
U. S. Nuclear Regulatory Commission, Region II
Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, GA 30303

D. E. Labarge, NRC Senior Project Manager (ONS)
U. S. Nuclear Regulatory Commission
Mail Stop O-8 H12
Washington, DC 20555-0001

M. C. Shannon
Senior Resident Inspector
Oconee Nuclear Station

V. R. Autry, Director
Division of Radioactive Waste Management
Bureau of Land and Waste Management
South Carolina Department of Health and
Environmental Control
2600 Bull Street
Columbia, SC 29201

AFFIDAVIT

W. R. McCollum, Jr., being duly sworn, states that he is the Site Vice President of Duke Energy Corporation: that he is authorized on the part of said Corporation to sign and file with the Nuclear Regulatory Commission revisions to the Oconee Nuclear Station Facility Operating Licenses No. DPR-38, DPR-47, and DPR-55; and that all the statements and matters set forth herein are true and correct to the best of his knowledge.

W. R. McCollum, Jr., Site Vice President

Subscribed and sworn to be before me this 13th day of

April, 2000.

Connie M. Breazeale
Notary Public

My Commission Expires:

2-13-2003

SEAL

ATTACHMENT I

MARKUPS OF THE TECHNICAL SPECIFICATIONS AND BASES

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, the maximum local fuel pin centerline temperature shall be ≤ 4642 - (5.8 x 10^{-3} x (Burnup, MWD/MTU))° F. Operation within this limit is ensured by compliance with the Axial Power Imbalance Protective Limits as specified in the Core Operating Limits Report.

2.1.1.2 In MODES 1 and 2, the departure from nucleate boiling ratio shall be maintained greater than the limit of 1.18 for the BWC correlation and 1.19 for the BWU correlation. Operation within this limit is ensured by compliance with the Axial Power Imbalance Protective Limits and RCS Variable Low Pressure Protective Limits as specified in the Core Operating Limits Report.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2750 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed:

2.2.1 In MODE 1 or 2, if SL 2.1.1.1 or SL 2.1.1.2 is violated, be in MODE 3 within 1 hour.

2.2.2 In MODE 1 or 2, if SL 2.1.2 is violated, restore compliance within limits and be in MODE 3 within 1 hour.

2.2.3 In MODES 3, 4, and 5, if SL 2.1.2 is violated, restore RCS pressure to ≤ 2750 psig within 5 minutes.

4.0 DESIGN FEATURES

4.1 Site Location

The Oconee Nuclear Station is approximately eight miles northeast of Seneca, South Carolina. The minimum distance from the reactor center line to the boundary of the exclusion area and to the outer boundary of the low population zone, as defined in 10 CFR 100.3, shall be one mile and six miles respectively.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 177 fuel assemblies. Each assembly shall consist of a matrix of zirconium alloy or M5 clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy, M5, or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 Control Assemblies

The reactor core shall contain 61 full-length CONTROL ROD Assemblies (CRAs) and 8 APSR assemblies. The full-length CRAs and APSR assemblies shall conform to the design described in the UFSAR or reload report.

4.3 Fuel Storage

4.3.1 Criticality

The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum nominal U-235 enrichment of 5.0 weight percent;

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

(7) DPC-NE-3000P-A, Thermal Hydraulic Transient Analysis Methodology, Rev. 2, (SER dated October 14, 1998);

(8) DPC-NE-2005P-A, Thermal Hydraulic Statistical Core Design Methodology, Rev. ~~1~~ 2, (SER dated ~~November 7, 1996~~ June 8, 1999); and

(9) DPC-NE-3005-PA, UFSAR Chapter 15 Transient Analysis Methodology, Rev. 1, (SER dated May 25, 1999).

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 System (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

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

5.6.6 Post Accident Monitoring (PAM) and Main Feeder Bus Monitor Panel (MFPMP) Report

When a report is required by Condition B or G of LCO 3.3.8, "Post Accident Monitoring (PAM) Instrumentation" or Condition D of LCO 3.3.23, "Main Feeder Bus Monitor Panel," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring (PAM only), the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.

(10) BAW-10227-PA, Evaluation of Advanced Cladding and Structural Material (MS) in PWR Reactor Fuel, (SER dated December 14, 1999).

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

ONS Design Criteria (Ref. 1) require that reactor core SLs ensure specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated transients. This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (95/95 DNB criterion) that DNB will not occur and by requiring that the fuel centerline temperature stays below the melting temperature.

and the BWU (Ref. 4)

DNB is not a directly measurable parameter during operation, but neutron power and Reactor Coolant System (RCS) temperature, flow and pressure can be related to DNB using a critical heat flux (CHF) correlation. The BWC (Ref. 2) CHF correlation has been developed to predict DNB for axially uniform and non-uniform heat flux distributions. The BWC correlation applies to Mark-BZ fuel. The local DNB heat flux ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual local heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.18 (BWC).

The BWU correlation applies to the Mark-B11 fuel.

and 1.19 (BWU)

The restrictions of this SL prevent overheating of the fuel and cladding and possible cladding perforation that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. 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.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film,

BASES (continued)

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs.

2.2.1

If SL 2.1.1.1 or SL 2.1.1.2 is violated, the requirement to go to MODE 3 places the unit in a MODE in which these SLs are not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where these SLs are not applicable and reduces the probability of fuel damage.

REFERENCES

1. UFSAR, Section 3.1.

2. BAW-10143P-A, "BWC Correlation of Critical Heat Flux," April 1995.

3. UFSAR, Chapter 15.

4. BAW-10199P, "The BWU Critical Heat Flux Correlations," Addendum 1, April 2000

BASES

ACTIONS (continued)

A.2.4

0.18% RTP, 0.36% RTP

The existing CONTROL ROD configuration must not cause an ejected rod to exceed the limit of ~~0.2%~~ Δk/k at ~~RPT, 0.4%~~ Δk/k at 80% ~~RPT~~, or ~~0.8%~~ Δk/k at zero power. This evaluation may require a computer calculation of the maximum ejected rod worth based on nonstandard configurations of the CONTROL ROD groups. The evaluation must determine the ejected rod worth for the duration of time that operation is expected to continue with a misaligned rod. Should fuel cycle conditions at some later time become more bounding than those at the time of the rod misalignment, additional evaluation will be required to verify the continued acceptability of operation. The required Completion Time of 72 hours is acceptable because LHRs are limited by the THERMAL POWER reduction and sufficient time is provided to perform the required evaluation.

0.7%

B.1

If the Required Actions and associated Completion Times for Condition A are not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, for reaching MODE 3 from RTP in an orderly manner and without challenging unit systems.

C.1.1

More than one trippable CONTROL ROD becoming inoperable or misaligned, or both inoperable but trippable and misaligned from their group average position, is not expected and may violate the minimum SDM requirement. Therefore, SDM must be evaluated. Ensuring the SDM meets the minimum requirement within 1 hour allows the operator adequate time to determine the SDM.

C.1.2

If the SDM is less than the limit, then the restoration of the required SDM requires increasing the RCS boron concentration to provide negative reactivity. RCS boration must occur as described in Bases Section 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated transients assume initial conditions within the normal steady state envelope. The limits placed on DNB related parameters ensure that these parameters will not be less conservative than were assumed in the analyses and thereby provide assurance that the minimum departure from nucleate boiling ratio (DNBR) will meet the required criteria for each of the transients analyzed.

The LCO for minimum RCS pressure is consistent with operation within the nominal operating envelope and is above that used as the initial pressure in the analyses. A pressure greater than the minimum specified will produce a higher minimum DNBR. A pressure lower than the minimum specified will cause the unit to approach the DNB limit.

The LCO for maximum RCS coolant loop average temperature is consistent with full power operation within the nominal operating envelope and is lower than the initial loop average temperature in the analyses. A loop average temperature lower than that specified will produce a higher minimum DNBR. A loop average temperature higher than that specified will cause the unit to approach the DNB limit.

The RCS flow rate is not expected to vary during operation with all pumps running. The LCO for the minimum RCS flow rate corresponds to that assumed for the DNBR analyses. A higher RCS flow rate will produce a higher DNBR. A lower RCS flow will cause the unit to approach the DNB limit.

APPLICABLE SAFETY ANALYSES

The requirements of LCO 3.4.1 represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will meet the DNBR criterion of ≥ 1.18 for BWC correlation, ≥ 1.19 for BWU correlation, or an equally valid limit when the statistical DNBR limit is employed (SCD methodology). This is the acceptance limit for the RCS DNBR parameters.

ATTACHMENT II

RETYPED TECHNICAL SPECIFICATIONS AND BASES

INSTRUCTIONS FOR UPDATING THE
TECHNICAL SPECIFICATIONS AND BASES

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, the maximum local fuel pin centerline temperature shall be $\leq$ 4642 - (5.8 x 10^{-3} x (Burnup, MWD/MTU))° F. Operation within this limit is ensured by compliance with the Axial Power Imbalance Protective Limits as specified in the Core Operating Limits Report.

2.1.1.2 In MODES 1 and 2, the departure from nucleate boiling ratio shall be maintained greater than the limit of 1.18 for the BWC correlation and 1.19 for the BWU correlation. Operation within this limit is ensured by compliance with the Axial Power Imbalance Protective Limits and RCS Variable Low Pressure Protective Limits as specified in the Core Operating Limits Report.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained $\leq$ 2750 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed:

2.2.1 In MODE 1 or 2, if SL 2.1.1.1 or SL 2.1.1.2 is violated, be in MODE 3 within 1 hour.

2.2.2 In MODE 1 or 2, if SL 2.1.2 is violated, restore compliance within limits and be in MODE 3 within 1 hour.

2.2.3 In MODES 3, 4, and 5, if SL 2.1.2 is violated, restore RCS pressure to $\leq$ 2750 psig within 5 minutes.

4.0 DESIGN FEATURES

4.1 Site Location

The Oconee Nuclear Station is approximately eight miles northeast of Seneca, South Carolina. The minimum distance from the reactor center line to the boundary of the exclusion area and to the outer boundary of the low population zone, as defined in 10 CFR 100.3, shall be one mile and six miles respectively.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 177 fuel assemblies. Each assembly shall consist of a matrix of zirconium alloy or M5 clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy, M5, or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 Control Assemblies

The reactor core shall contain 61 full-length CONTROL ROD Assemblies (CRAs) and 8 APSR assemblies. The full-length CRAs and APSR assemblies shall conform to the design described in the UFSAR or reload report.

4.3 Fuel Storage

4.3.1 Criticality

The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum nominal U-235 enrichment of 5.0 weight percent;

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

(7) DPC-NE-3000P-A, Thermal Hydraulic Transient Analysis Methodology, Rev. 2, (SER dated October 14, 1998);

(8) DPC-NE-2005P-A, Thermal Hydraulic Statistical Core Design Methodology, Rev. 2, (SER dated June 8, 1999); and

(9) DPC-NE-3005-PA, UFSAR Chapter 15 Transient Analysis Methodology, Rev. 1, (SER dated May 25, 1999).

(10) BAW-10227-PA, Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel, (SER dated December 14, 1999).

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 System (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

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

5.6.6 Post Accident Monitoring (PAM) and Main Feeder Bus Monitor Panel (MFPMP) Report

When a report is required by Condition B or G of LCO 3.3.8, "Post Accident Monitoring (PAM) Instrumentation" or Condition D of LCO 3.3.23, "Main Feeder Bus Monitor Panel," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring (PAM only), the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

ONS Design Criteria (Ref. 1) require that reactor core SLs ensure specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated transients. This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (95/95 DNB criterion) that DNB will not occur and by requiring that the fuel centerline temperature stays below the melting temperature.

DNB is not a directly measurable parameter during operation, but neutron power and Reactor Coolant System (RCS) temperature, flow and pressure can be related to DNB using a critical heat flux (CHF) correlation. The BWC (Ref. 2) and the BWU (Ref. 4) CHF correlations have been developed to predict DNB for axially uniform and non-uniform heat flux distributions. The BWC correlation applies to Mark-BZ fuel. The BWU correlation applies to the Mark-B11 fuel. The local DNB heat flux ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual local heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.18 (BWC) and 1.19 (BWU).

The restrictions of this SL prevent overheating of the fuel and cladding and possible cladding perforation that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. 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.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film,

BASES (continued)

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs.

2.2.1

If SL 2.1.1.1 or SL 2.1.1.2 is violated, the requirement to go to MODE 3 places the unit in a MODE in which these SLs are not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where these SLs are not applicable and reduces the probability of fuel damage.

REFERENCES

1. UFSAR, Section 3.1.

2. BAW-10143P-A, "BWC Correlation of Critical Heat Flux," April 1995.

3. UFSAR, Chapter 15.

4. BAW-10199P, "The BWU Critical Heat Flux Correlations," Addendum 1, April 2000

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated transients assume initial conditions within the normal steady state envelope. The limits placed on DNB related parameters ensure that these parameters will not be less conservative than were assumed in the analyses and thereby provide assurance that the minimum departure from nucleate boiling ratio (DNBR) will meet the required criteria for each of the transients analyzed.

The LCO for minimum RCS pressure is consistent with operation within the nominal operating envelope and is above that used as the initial pressure in the analyses. A pressure greater than the minimum specified will produce a higher minimum DNBR. A pressure lower than the minimum specified will cause the unit to approach the DNB limit.

The LCO for maximum RCS coolant loop average temperature is consistent with full power operation within the nominal operating envelope and is lower than the initial loop average temperature in the analyses. A loop average temperature lower than that specified will produce a higher minimum DNBR. A loop average temperature higher than that specified will cause the unit to approach the DNB limit.

The RCS flow rate is not expected to vary during operation with all pumps running. The LCO for the minimum RCS flow rate corresponds to that assumed for the DNBR analyses. A higher RCS flow rate will produce a higher DNBR. A lower RCS flow will cause the unit to approach the DNB limit.

APPLICABLE SAFETY ANALYSES

The requirements of LCO 3.4.1 represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will meet the DNBR criterion of $\geq$ 1.18 for BWC correlation, $\geq$ 1.19 for BWU correlation, or an equally valid limit when the statistical DNBR limit is employed (SCD methodology). This is the acceptance limit for the RCS DNBR parameters.

BASES

ACTIONS (continued)

A.2.4

The existing CONTROL ROD configuration must not cause an ejected rod to exceed the limit of 0.18% $\Delta k/k$ at RTP, 0.36% $\Delta k/k$ at 80% RTP, or 0.7% $\Delta k/k$ at zero power. This evaluation may require a computer calculation of the maximum ejected rod worth based on nonstandard configurations of the CONTROL ROD groups. The evaluation must determine the ejected rod worth for the duration of time that operation is expected to continue with a misaligned rod. Should fuel cycle conditions at some later time become more bounding than those at the time of the rod misalignment, additional evaluation will be required to verify the continued acceptability of operation. The required Completion Time of 72 hours is acceptable because LHRs are limited by the THERMAL POWER reduction and sufficient time is provided to perform the required evaluation.

B.1

If the Required Actions and associated Completion Times for Condition A are not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, for reaching MODE 3 from RTP in an orderly manner and without challenging unit systems.

C.1.1

More than one trippable CONTROL ROD becoming inoperable or misaligned, or both inoperable but trippable and misaligned from their group average position, is not expected and may violate the minimum SDM requirement. Therefore, SDM must be evaluated. Ensuring the SDM meets the minimum requirement within 1 hour allows the operator adequate time to determine the SDM.

C.1.2

If the SDM is less than the limit, then the restoration of the required SDM requires increasing the RCS boron concentration to provide negative reactivity. RCS boration must occur as described in Bases Section 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to

Attachment III
Description of the Proposed Changes and
Technical Justifications for the Proposed Changes

Background

This license amendment request proposes changes to the Oconee technical specifications and associated Bases to implement Mark-B11 fuel with M5 cladding. These changes are mostly administrative in nature with the technical justification provided by the referenced topical reports. The Mark-B11 fuel design has been reviewed and approved by the NRC (BAW-10229P). The M5 cladding material has also been reviewed and approved by the NRC (BAW-10227P). Lead test assemblies using the Mark-B11 design have been irradiated at Oconee for two cycles and have shown acceptable performance. Lead test assemblies using M5 cladding have been irradiated at a number of facilities in the United States and in Europe, including Duke's McGuire Nuclear Station, and acceptable performance has been demonstrated. Thus, there are no outstanding issues related to the design or performance of the fuel or the cladding material.

O1C20 will be the first Oconee core to utilize Mark-B11 fuel with M5 cladding. Therefore, the O1C20 reload analyses addressed the differences in the fuel design on the safety analysis. Attachment VII provides the results of this reanalysis of the UFSAR Chapter 15 events, which shows that all of the acceptance criteria are satisfied for O1C20 using Mark-B11 with M5 cladding.

Description of Proposed Changes

This licensing amendment request provides the changes necessary to incorporate Mark-B11 fuel with M5 cladding. The Mark-B11 fuel assembly incorporates mixing vane grids and a smaller rod diameter. To analyze the thermal-hydraulic performance, a new critical heat flux (CHF) correlation is used (BWU correlation). Changes to the technical specifications and bases are necessary to include the use of this correlation.

The M5 cladding material is an advanced alloy designed to reduce fuel rod corrosion and irradiation induced growth. The chemical composition of M5 differs from the specifications of Zircaloy. An administrative change to the technical specification is needed to add M5 to the description of the fuel assemblies. The difference in the composition of M5 compared to Zircaloy yields different high-temperature creep and deformation characteristics. As a result, revised cladding ballooning and flow blockage models are used in the LOCA evaluation model. A change to the technical specification (list of approved analytical methods) is necessary to include this revision to the LOCA evaluation model.

<u>Proposed Changes to TS 2.1 and Associated Bases</u>

Duke proposes to modify Safety Limit 2.1.1.2 to include the BWU correlation with its departure from nucleate boiling ratio (DNBR) limit of 1.19 to analyze Mark-B11 fuel. In addition, Duke proposes to revise the bases discussions for Safety Limit 2.1.1 to reflect this change.

The reactor core safety limits ensure that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated transients. One of these fuel design limits is to ensure that DNB will not occur at a 95% probability level at a 95% confidence level. Since DNB is not a measurable parameter during operation, DNB can be related to measurable parameters using a critical heat flux (CHF) correlation. Oconee currently uses non-mixing vane fuel with the BWC CHF correlation. The Mark-B11 fuel uses mixing vanes, and to accurately characterize the DNB response, a new CHF correlation is required. The BWU correlation was developed for application to fuel assemblies with mixing vanes. Topical report BAW-10199P justifies the use of this correlation.

Proposed Change to TS 4.2

Duke proposes to add M5 alloy as one of the materials to the description of the fuel rod cladding and filler rods. M5 is an advanced fuel rod cladding material designed to accommodate higher fuel rod burnups and to increase the performance margins with respect to fuel rod corrosion and fuel rod growth.

The BAW-10227P-A topical report provides justification for the use of M5 cladding along with the analysis changes necessary to model this material. The NRC has issued an SER for this topical report indicating that M5 is an acceptable cladding material for use in light-water nuclear reactors. However, 10 CFR 50.44, 10 CFR 50.46, and 10 CFR Part 50, Appendix K specifically contemplates the use of Zircaloy or Zirlo™ as the fuel rod cladding material. Therefore, Duke requested and was granted an exemption from the specific wording of these regulations, allowing the use of M5 cladding material.

Proposed Change to TS 5.6

Duke proposes to modify the list of previously reviewed and approved analytical methods used to determine the core operating limits. Reporting Requirements 5.6.5.b will be modified to add the topical report on M5 cladding (BAW-10227P-A), and to update the thermal hydraulic statistical core design topical report to the latest revision (DPC-NE-2005P-A Rev. 2).

The BAW-10227P-A topical report describes the LOCA evaluation model changes necessary to analyze M5 cladding. These models have been reviewed and approved by the NRC and together with BAW-10192-PA define the LOCA evaluation model that will be used to determine the LOCA related core operating limits. Revision 2 of topical report DPC-NE-2005P-A describes the methods that will be used to analyze the thermal hydraulic response of the Mark-B11 fuel design. This topical report has also been reviewed and approved by the NRC.

Proposed Changes to the Bases for TS 3.1.4.

Duke proposes to modify the bases of the Required Action A.2.4 for the control rod group alignment limits to be representative of the rod ejection analysis. The bases provides the assumed ejected rod worths for rated thermal power (RTP), 80% RTP, and zero power. The proposed values of the assumed rod worths are as follows:

0.18% $\Delta k/k$ at RTP
0.36% $\Delta k/k$ at 80% RTP
0.7% $\Delta k/k$ at zero power

The required action is to verify the potential ejected rod worth is within the assumptions of the rod ejection analysis. In analyzing the Mark-B11 fuel, these values were reduced by approximately 10%. Therefore, the bases of the control rod alignment limits is modified to provide the values that are assumed in the rod ejection analysis.

Proposed Change to the Bases for TS 3.4.1

Duke proposes to modify the bases of TS 3.4.1 in the applicable safety analysis section to include the BWU correlation along with its DNBR limit of 1.19.

The requirements of LCO 3.4.1 represent the initial conditions for DNB limited transients analyzed in the plant safety analyses. The bases section states that this LCO will meet the DNBR criterion of the applicable CHF correlation. To analyze Mark-B11 fuel, the BWU CHF correlation is used with a DNBR limit of 1.19. Therefore, the BWU CHF correlation is added to the applicable safety analysis section along with its DNBR limit of 1.19.

Conclusion

This LAR is administrative in nature. The topical reports related to the fuel design (BAW-10229P) and the cladding material (BAW-10227P) have been reviewed and approved by the NRC. Lead test assemblies have demonstrated the satisfactory performance of both the fuel design and cladding material. The safety analysis for Oconee 1 Cycle 20 has been performed using approved methods and has demonstrated that all of the acceptance criteria have been met. Therefore, Duke concludes that: 1) implementation of this amendment request will not endanger the health and safety of the public, 2) the amendment is in compliance with the Commission's regulations, and 3) the issuance of the amendment will not be adverse to the common defense and security or to the health and safety of the public.

ATTACHMENT IV

NO SIGNIFICANT HAZARDS CONSIDERATION

Pursuant to 10 CFR 50.91, Duke has made the determination that this license amendment involves no significant hazards by applying the standards established by the NRC regulations in 10 CFR 50.92. This ensures that operation of the facility in accordance with the proposed amendment would not:

1. Involve a significant increase in the probability or consequence of an accident previously evaluated?

   No. The proposed change to the technical specifications and bases incorporate the use of Mark-B11 fuel assemblies with M5 cladding. The analyzed events are initiated by the failure of specific plant structures, systems, or components. The change in fuel assembly design or cladding material does not impact the condition or performance of those structures, system, or components. Therefore, the proposed changes will not increase the probability of an accident previously evaluated.

   The accident analyses have been evaluated to address the changes in the fuel design and cladding material. The results of this evaluation demonstrate that the applicable acceptance criteria are met. Thus, the proposed changes will not increase the consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated?

   No. The proposed changes to the technical specifications are to support implementation of Mark-B11 fuel assemblies with M5 cladding. The changes in fuel design and cladding material do not alter the operating characteristics of the plant. In addition, the fuel handling equipment is compatible with the Mark-B11 fuel assembly design. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Involve a significant reduction in a margin of safety?

   No. The margin of safety is established through the design of the plant systems, structures, components, and the parameters within which the plant is operated. The proposed change does not involve any significant physical change to the plant. The primary design changes, which enhance nuclear, thermal-hydraulic and mechanical performance, include the following:

   1. Reduced diameter fuel rod,
   2. Flow mixing vanes on five of the six intermediate spacer grids,
   3. Improved grid restraint system, and
   4. M5 fuel rod cladding

The changes in fuel design and cladding material have been evaluated which demonstrates that all of the applicable acceptance criteria are met. Based on this, the proposed changes do not involve a significant reduction in a margin of safety.

Duke has concluded based on the above that there are no significant hazards considerations involved in this request.

ATTACHMENT V

ENVIRONMENTAL ASSESSMENT

Pursuant to 10 CFR 51.22 (b), an evaluation of the proposed amendment has been performed to determine whether or not it meets the criteria for categorical exclusion set forth in 10 CFR 51.22 (c) (9) of the regulations. The proposed amendment does not involve:

1) A significant hazards consideration

This conclusion is supported by the No Significant Hazards Consideration evaluation, which is provided in Attachment IV.

2) A significant change in the type or significant increase in the amount of any effluent that may be released offsite.

3) A significant increase in the individual or cumulative occupational radiation exposure.

The evaluation of the dose analysis performed for the Mark-B11 fuel indicate that all of the applicable radiological and environmental acceptance criteria will continue to be met. Therefore, this LAR meets the criteria set forth in 10 CFR 51.22 (c) (9) of the regulations for categorical exclusion from performing an environmental assessment / impact statement

ATTACHMENT VI

CORE OPERATING LIMITS REPORT FOR
OCONEE UNIT 1 CYCLE 20

Duke Power Company

Oconee 1 Cycle 20

Core Operating Limits Report

QA Condition 1

Not Reviewed or Approved by CFAM 3.13

Prepared By : _____ Date : _____

Checked By : _____ Date : _____

CDR By : _____ Date : _____

Approved By : _____ Date : _____

Oconee 1 Cycle 20

Core Operating Limits Report

Insertion Sheet for Revision Draft

This revision is not valid until the end of operation for Oconee 1 Cycle 19.

Remove these revision 13 pages	Insert these revision Draft pages
1 - 31	1 - 31

Revision Log

Revision	Effective Date	Pages Revised	Pages Added	Pages Deleted	Total Effective Pages
Oconee 1 Cycle 20 revisions below					
Draft	Mar-00	1 - 31	-	31	31
Oconee 1 Cycle 19 revisions below					
13	Feb-00	1,2,3,4	-	-	31
12	Jul-99	1, 2, 3, 8, 10, 13, 31	-	-	31
11	May-99	1 - 31	-	1-31	31
Oconee 1 Cycle 18 revisions below					
10	Mar-99	1 - 31	-	32-38	31
9	Feb-98	1,2,3,5,13, 16,17,32,36	-	-	38
8	Nov-97	1,2,3,5,10, 32	37	-	38
7	Aug-97	1 - 38	-	-	38

Oconee 1 Cycle 20

1.0 Error Adjusted Core Operating Limits

The Core Operating Limits Report for O1C20 has been prepared in accordance with the requirements of ITS 5.6.5. The core operating limits within this report have been developed using NRC approved methodology identified in references 1, 2, 3, 4, 5, 6, 7, 8, 9 and 10. The RPS protective limits and maximum allowable setpoints are documented in references 11 and 12. These limits are validated for use in O1C20 by references 13, 14, and 15. The O1C20 analyses assume a design flow of 107.5% of 88,000 gpm per RCS pump, radial local peaking (FΔh) of 1.714, and axial peaking factor (Fz) of 1.5.

The error adjusted core operating limits included in section 1 of the report incorporate all necessary uncertainties and margins required for operation of the O1C20 reload core.

1.1 References

1. Nuclear Design Methodology Using CASMO-3 / SIMULATE-3P, DPC-NE-1004A, Revision 0, SER dated November 23, 1992.

2. Oconee Nuclear Station Reload Design Methodology II, DPC-NE-1002A, Revision 1, SER dated October 1, 1985.

3. Oconee Nuclear Station Reload Design Methodology, NFS-1001A, Revision 4, SER dated July 29, 1981.

4. ONS Core Thermal Hydraulic Methodology Using VIPRE-01, DPC-NE-2003A, SER dated July 19, 1989.

5. Thermal Hydraulic Statistical Core Design Methodology, DPC-NE-2005P-A, Revision 2, SER dated June 8, 1999.

6. Fuel Mechanical Reload Analysis Methodology Using TACO3, DPC-NE-2008P-A, SER dated April 3, 1995.

7. UFSAR Chapter 15 Transient Analysis Methodology, DPC-NE-3005-PA, Revision 1, SER dated May 25, 1999.

8. DPC-NE-3000P-A, Thermal Hydraulic Transient Analysis Methodology, Rev. 2, SER dated October 14, 1998.

9. BAW-10192P-A, BWNT LOCA - BWNT Loss of Coolant Accident Evaluation Model for Once-Through Steam Generator Plants, SER dated February 18, 1997.

10. BAW-10227-PA, Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel, SER dated February 4, 2000.

11. Variable Low Pressure Safety Limit, OSC-4048, Revision 3, July 1998.

12. Power Imbalance Safety Limits and Tech Spec Setpoints Using Error Adjusted Flux-Flow Ratio of 1.094, OSC-5604, Revision 1, November 1998.

13. O1C20 Maneuvering Analysis, OSC-7528, Revision 0, March 2000.

14. O1C20 Specific DNB Analysis, OSC-7536, Revision 0, March 2000.

15. O1C20 Reload Safety Evaluation, OSC-7537, Revision 0, March 2000.

Oconee 1 Cycle 20

Miscellaneous Setpoints

BWST boron concentration shall be greater than 2220 ppm and less than 3000 ppm.
Referred to by ITS 3.5.4.

Spent fuel pool boron concentration shall be greater than 2220 ppm and less than 3000 ppm.
Referred to by ITS 3.7.12.

The equivalent of at least 1100 cubic feet of 11,000 ppm boron shall be maintained in the CBAST.
Referred to by ITS SLC 16.5.13.

CFT boron concentration shall be greater than 1835 ppm. The average boron concentration in the CFT's shall be less than 4000 ppm. Referred to by ITS 3.5.1.

RCS and Refueling canal boron concentration shall be greater than 2220 ppm.
Referred to by ITS 3.9.1.

Shutdown Margin (SDM) shall be greater than 1% Δk/k.
Referred to by ITS 3.1.1.

Moderator Temperature Coefficient (MTC) shall be less than:
Linear interpolation is valid within the table provided.
Referred to by ITS 3.1.3.

MTC x 10-4 Δρ / °F	% FP
+0.70	0
+0.40	15
0.00	80
-0.125	100
-0.25	120

Departure from Nucleate Boiling (DNB) parameter for RCS loop pressure shall be
Referred to by ITS 3.4.1.

4 RCP: measured hot leg pressure ≥ 2125 psig
3 RCP: measured hot leg pressure ≥ 2125 psig

DNB parameter for RCS loop average temperature shall be:
Referred to by ITS 3.4.1.

Max Loop Tavg Incl 2°F unc	ΔTc, °F
581.00	0

The measured Tavg must be less than the temperature specified by an amount equal to the uncertainty corresponding to the instrument from which it is read.
ΔTc is the setpoint value selected by the operators.

DNB parameter for RCS loop total flow shall be:
Referred to by ITS 3.4.1.

4 RCP: Measured ≥ 108.5 %df
3 RCP: Measured ≥ 74.7 % of 4 RCP min flow

Regulating rod groups shall be withdrawn in sequence starting with group 5, group 6, and finally group 7.
Referred to by ITS 3.2.1.

Regulating rod group overlap shall be 25% ± 5% between two sequential groups.
Referred to by ITS 3.2.1.

Oconee 1 Cycle 20

Steady State Operating Band

	Rod Index		APSR %WD	
EFPD	Min	Max	Min	Max
0 to 430	292 ± 5	300	30	40
430 to EOC	292 ± 5	300	100	100

Quandrant Power Tilt Setpoints

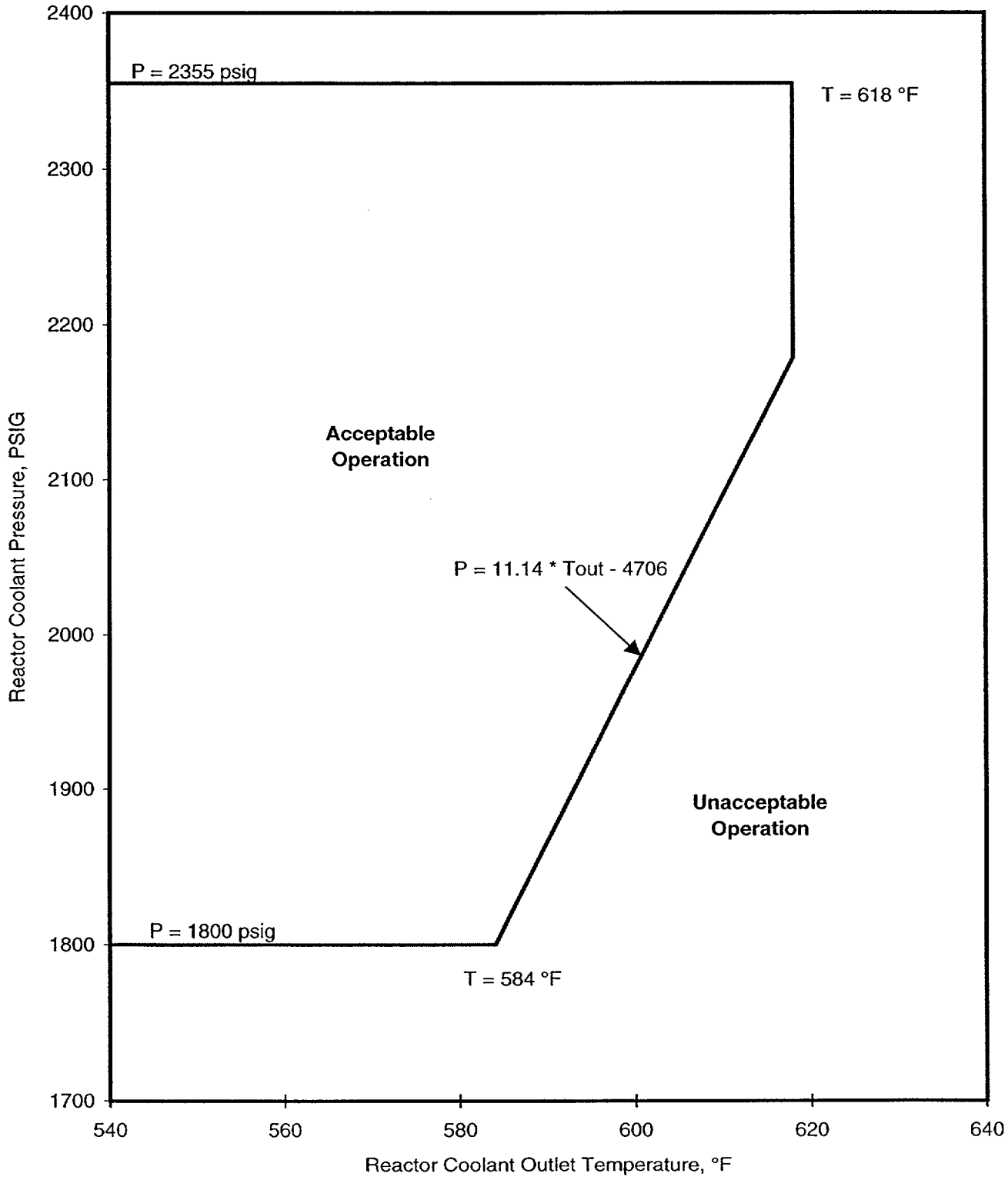
	Steady State		Transient		Maximum
Core Power Level, %FP	30 - 100	0 - 30	30 - 100	0 - 30	0 - 100
Full Incore	3.50	7.94	7.44	9.73	16.88
Out of Core	2.05	6.09	5.63	7.72	14.22
Backup Incore	2.16	3.87	3.63	4.81	10.07

Referred to by ITS 3.2.3.

Oconee 1 Cycle 20

Variable Low RCS Pressure RPS Setpoints

Referred to by ITS 3.3.1



Oconee 1 Cycle 20

RPS Power Imbalance Setpoints

	% FP	% Imbalance
4 Pumps	0	-33.0
	90.4	-33.0
	107.9	-14.4
	107.9	14.4
	90.4	33.0
	0	33.0
3 Pumps	0	-33.0
	63.1	-33.0
	80.6	-14.4
	80.6	14.4
	63.1	33.0
	0	33.0

Oconee 1 Cycle 20

Operational Power Imbalance Setpoints

	%FP	Full Incore	Backup Incore	Out of Core
4 Pumps	0	-30.7	-30.1	-30.7
	80	-30.7	-30.1	-30.7
	90	-27.6	-27.1	-27.6
	100	-19.1	-18.7	-19.1
	102	-17.0	-17.0	-17.0
	102	17.0	16.5	17.0
	100	19.1	18.7	19.1
	90	29.3	28.6	29.3
	80	29.3	28.6	29.3
	0	29.3	28.6	29.3
3 Pumps	0.0	-30.7	-30.1	-30.7
	64.06	-30.7	-	-30.7
	64.62	-	-30.1	-
	77.0	-17.0	-17.0	-17.0
	77.0	17.0	16.5	17.0
	66.04	-	28.6	-
	65.38	29.3	-	29.3
	0.0	29.3	28.6	29.3

Oconee 1 Cycle 20

Operational Power Imbalance Setpoints

Operation with 4 RCS Pumps, BOC to EOC

% FP	RPS Trip		Full Incore Alarm		Out of Core Alarm	
107.9	-14.40	14.40				
107	-15.36	15.36				
106	-16.42	16.42				
105	-17.48	17.48				
104	-18.55	18.55				
103	-19.61	19.61				
102	-20.67	20.67	-17.00	17.00	-17.00	17.00
101	-21.73	21.73	-18.05	18.05	-18.05	18.05
100	-22.80	22.80	-19.10	19.10	-19.10	19.10
99	-23.86	23.86	-19.95	20.12	-19.95	20.12
98	-24.92	24.92	-20.80	21.14	-20.80	21.14
97	-25.99	25.99	-21.65	22.16	-21.65	22.16
96	-27.05	27.05	-22.50	23.18	-22.50	23.18
95	-28.11	28.11	-23.35	24.20	-23.35	24.20
94	-29.17	29.17	-24.20	25.22	-24.20	25.22
93	-30.24	30.24	-25.05	26.24	-25.05	26.24
92	-31.30	31.30	-25.90	27.26	-25.90	27.26
91	-32.36	32.36	-26.75	28.28	-26.75	28.28
90.4	-33.00	33.00	-27.26	28.89	-27.26	28.89
90	-33.00	33.00	-27.60	29.30	-27.60	29.30
89	-33.00	33.00	-27.91	29.30	-27.91	29.30
88	-33.00	33.00	-28.22	29.30	-28.22	29.30
87	-33.00	33.00	-28.53	29.30	-28.53	29.30
86	-33.00	33.00	-28.84	29.30	-28.84	29.30
85	-33.00	33.00	-29.15	29.30	-29.15	29.30
84	-33.00	33.00	-29.46	29.30	-29.46	29.30
83	-33.00	33.00	-29.77	29.30	-29.77	29.30
82	-33.00	33.00	-30.08	29.30	-30.08	29.30
81	-33.00	33.00	-30.39	29.30	-30.39	29.30
80	-33.00	33.00	-30.70	29.30	-30.70	29.30
0	-33.00	33.00	-30.70	29.30	-30.70	29.30
% FP	RPS Trip		Full Incore Alarm		Out of Core Alarm	

Oconee 1 Cycle 20

Operational Power Imbalance Setpoints

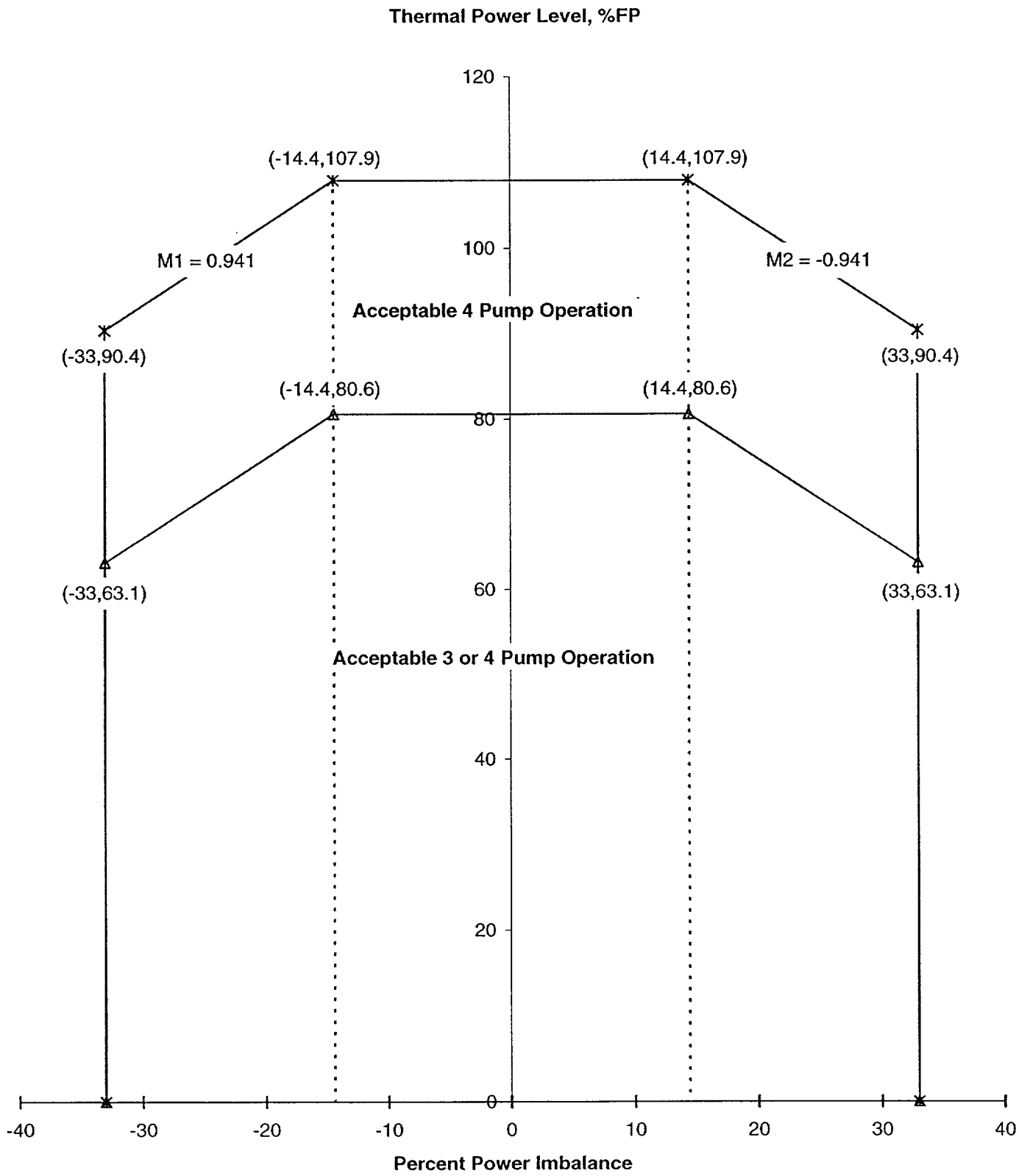
Operation with 3 RCS Pumps, BOC to EOC

% FP	RPS Trip		Full Incore Alarm		Out of Core Alarm	
80.6	-14.40	14.40				
80	-15.04	15.04				
79	-16.10	16.10				
78	-17.16	17.16				
77.0	-18.23	18.23	-17.00	17.00	-17.00	17.00
76	-19.29	19.29	-18.06	18.06	-18.06	18.06
75	-20.35	20.35	-19.12	19.12	-19.12	19.12
74	-21.41	21.41	-20.18	20.18	-20.18	20.18
73	-22.48	22.48	-21.23	21.23	-21.23	21.23
72	-23.54	23.54	-22.29	22.29	-22.29	22.29
71	-24.60	24.60	-23.35	23.35	-23.35	23.35
70	-25.67	25.67	-24.41	24.41	-24.41	24.41
69	-26.73	26.73	-25.47	25.47	-25.47	25.47
68	-27.79	27.79	-26.53	26.53	-26.53	26.53
67	-28.85	28.85	-27.58	27.58	-27.58	27.58
66	-29.92	29.92	-28.64	28.64	-28.64	28.64
65.4	-30.58	30.58	-29.30	29.30	-29.30	29.30
65	-30.98	30.98	-29.70	29.30	-29.70	29.30
64	-32.04	32.04	-30.76	29.30	-30.76	29.30
64.1	-31.98	31.98	-30.70	29.30	-30.70	29.30
63.1	-33.00	33.00	-30.70	29.30	-30.70	29.30
63	-33.00	33.00	-30.70	29.30	-30.70	29.30
62	-33.00	33.00	-30.70	29.30	-30.70	29.30
61	-33.00	33.00	-30.70	29.30	-30.70	29.30
60	-33.00	33.00	-30.70	29.30	-30.70	29.30
0	-33.00	33.00	-30.70	29.30	-30.70	29.30
% FP	RPS Trip		Full Incore Alarm		Out of Core Alarm	

Oconee 1 Cycle 20

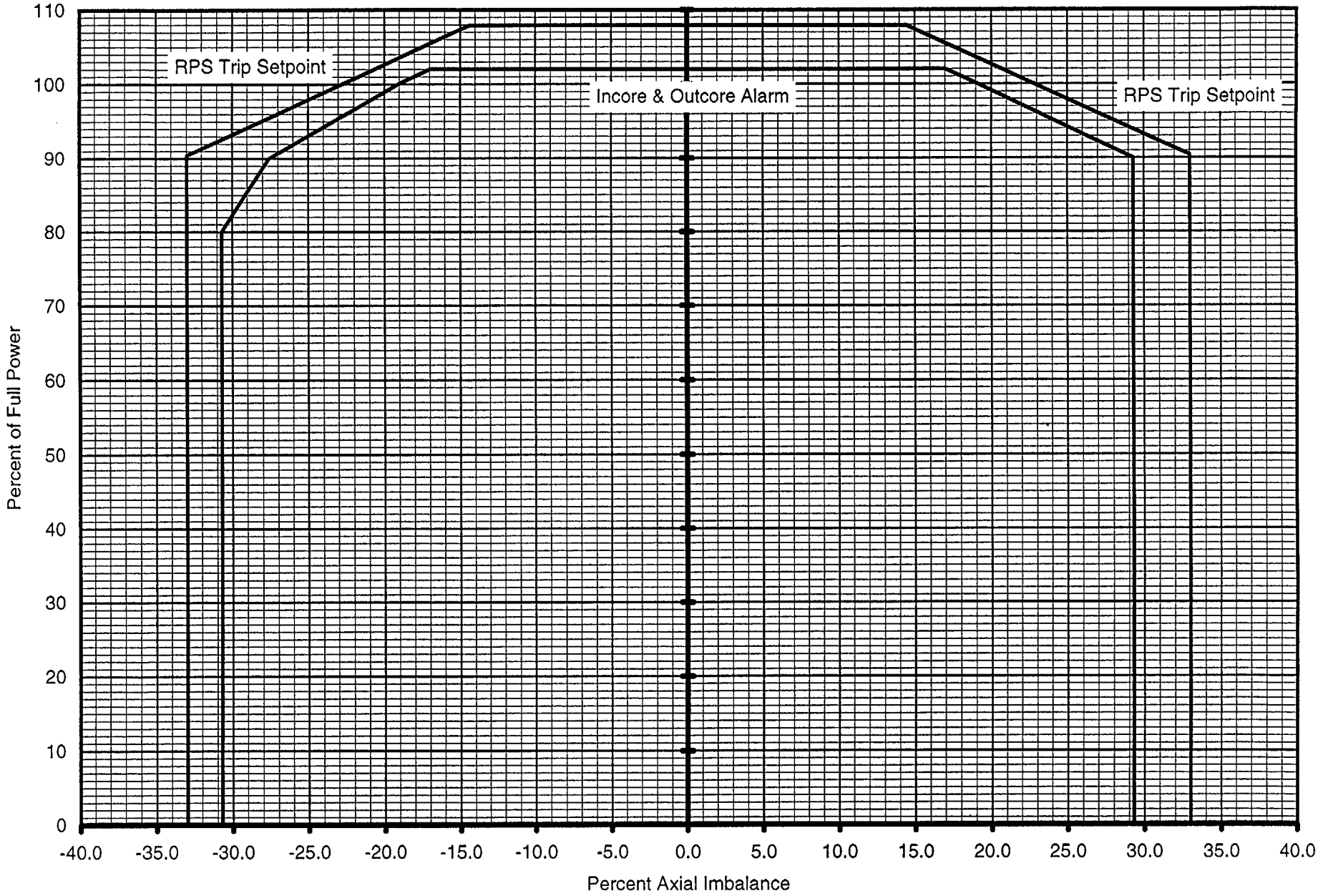
RPS Power Imbalance Setpoints

Referred to by ITS 3.3.1



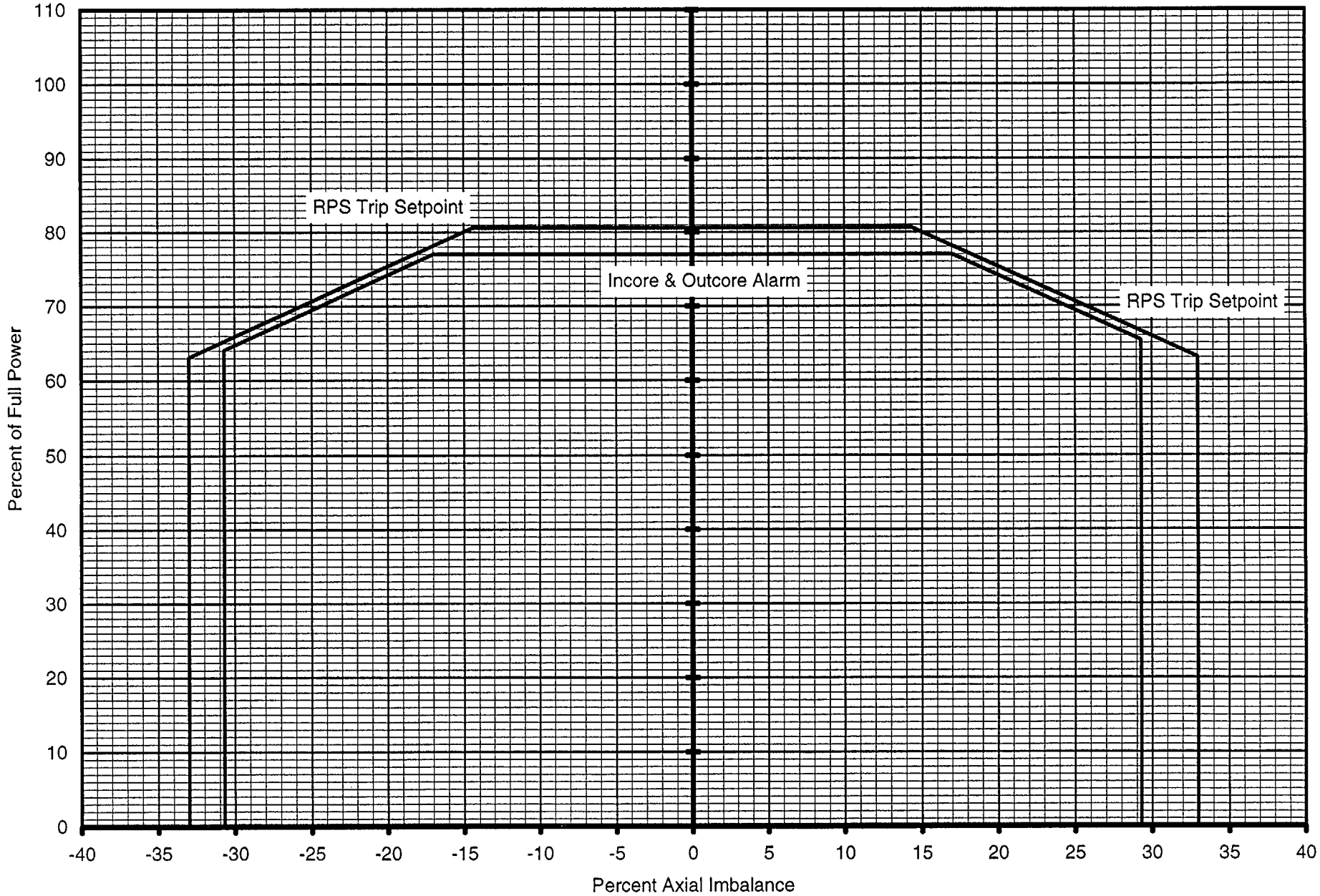
Oconee 1 Cycle 20

Imbalance Setpoints for 4 Pump Operation, BOC to EOC



Oconee 1 Cycle 20

Imbalance Setpoints for 3 Pump Operation, BOC to EOC



Oconee 1 Cycle 20

Operational Rod Index Setpoints

	%FP	RI Insertion Setpoint No Inop Rod	RI Insertion Setpoint 1 Inop Rod	RI Withdrawal Setpoint
4 Pumps	102.0	263.5	283.4	300
	100.0	261.5	281.5	300
	90.0	251.5	271.9	300
	80.0	241.5	262.3	300
	50.0	201.5	233.4	300
	48.0	195.2	231.5	300
	15.0	91.5	165.5	300
	13.0	76.5	161.5	300
	5.0	16.5	93.5	300
	3.0	1.5	76.5	300
	2.8	0.0	74.8	300
	0.0	0.0	51.0	300
3 Pumps	77.0	237.5	285.2	300
	75.0	234.8	281.5	300
	50.0	201.5	235.2	300
	48.0	195.2	231.5	300
	15.0	91.5	165.5	300
	13.0	76.5	161.5	300
	5.0	16.5	93.5	300
	3.0	1.5	76.5	300
	2.8	0.0	74.8	300
	0.0	0.0	51.0	300

Oconee 1 Cycle 20

Shutdown Margin Rod Index Setpoints

	%FP	RI Insertion Setpoint No Inop Rod	RI Insertion Setpoint 1 Inop Rod	RI Withdrawal Setpoint
4 Pumps	102.0	224.6	283.4	300
	100.0	221.5	281.5	300
	48.0	141.5	231.5	300
	13.0	76.5	161.5	300
	3.0	1.5	76.5	300
	2.8	0.0	74.8	300
	0.0	0.0	51.0	300
3 Pumps	77.0	227.4	285.2	300
	75.0	221.5	281.5	300
	48.0	141.5	231.5	300
	13.0	76.5	161.5	300
	3.0	1.5	76.5	300
	2.8	0.0	74.8	300
	0.0	0.0	51.0	300

Oconee 1 Cycle 20
Rod Index Setpoints
4 Pump Operation, No Inoperable Rods, BOC to EOC

% FP	Shutdown Margin Setpoint			Operational Alarm Setpoint		
	CRGP 5	CRGP 6	CRGP 7	CRGP 5	CRGP 6	CRGP 7
102	100	99.8	24.8	100	100	63.5
101	100	99.0	24.0	100	100	62.5
100	100	98.2	23.2	100	100	61.5
99	100	97.5	22.5	100	100	60.5
98	100	96.7	21.7	100	100	59.5
97	100	95.9	20.9	100	100	58.5
96	100	95.2	20.2	100	100	57.5
95	100	94.4	19.4	100	100	56.5
94	100	93.6	18.6	100	100	55.5
93	100	92.9	17.9	100	100	54.5
92	100	92.1	17.1	100	100	53.5
91	100	91.3	16.3	100	100	52.5
90	100	90.6	15.6	100	100	51.5
89	100	89.8	14.8	100	100	50.5
88	100	89.0	14.0	100	100	49.5
87	100	88.2	13.2	100	100	48.5
86	100	87.5	12.5	100	100	47.5
85	100	86.7	11.7	100	100	46.5
84	100	85.9	10.9	100	100	45.5
83	100	85.2	10.2	100	100	44.5
82	100	84.4	9.4	100	100	43.5
81	100	83.6	8.6	100	100	42.5
80	100	82.9	7.9	100	100	41.5
79	100	82.1	7.1	100	100	40.2
78	100	81.3	6.3	100	100	38.8
77	100	80.6	5.6	100	100	37.5
76	100	79.8	4.8	100	100	36.2
75	100	79.0	4.0	100	100	34.8
74	100	78.2	3.2	100	100	33.5
73	100	77.5	2.5	100	100	32.2
72	100	76.7	1.7	100	100	30.8
71	100	75.9	0.9	100	100	29.5
70	100	75.2	0.2	100	100	28.2
69.8	100	75.0	0	100	100	27.9
69	100	73.8	0	100	100	26.8
68	100	72.3	0	100	100	25.5
67.6	100	71.7	0	100	100	25.0
67	100	70.7	0	100	99.6	24.6
66	100	69.2	0	100	98.9	23.9
65	100	67.7	0	100	98.2	23.2
64	100	66.1	0	100	97.6	22.6
63	100	64.6	0	100	96.9	21.9
62	100	63.0	0	100	96.2	21.2
61	100	61.5	0	100	95.6	20.6
60	100	60.0	0	100	94.9	19.9
59	100	58.4	0	100	94.2	19.2
58	100	56.9	0	100	93.6	18.6
57	100	55.3	0	100	92.9	17.9
56	100	53.8	0	100	92.2	17.2
55	100	52.3	0	100	91.6	16.6
54	100	50.7	0	100	90.9	15.9
53	100	49.2	0	100	90.2	15.2
52	100	47.7	0	100	89.6	14.6
51	100	46.1	0	100	88.9	13.9
50	100	44.6	0	100	88.2	13.2
% FP	CRGP 5	CRGP 6	CRGP 7	CRGP 5	CRGP 6	CRGP 7
	Shutdown Margin Setpoint			Operational Alarm Setpoint		

RI = 300 is withdrawal limit at all power levels.

Continued on next page.

Oconee 1 Cycle 20
Rod Index Setpoints
4 Pump Operation, No Inoperable Rods, BOC to EOC

% FP	Shutdown Margin Setpoint			Operational Alarm Setpoint		
	CRGP 5	CRGP 6	CRGP 7	CRGP 5	CRGP 6	CRGP 7
49	100	43.0	0	100	86.7	11.7
48	100	41.5	0	100	85.1	10.1
47	100	39.6	0	100	83.5	8.5
46	100	37.8	0	100	82.0	7.0
45	100	35.9	0	100	80.4	5.4
44	100	34.1	0	100	78.8	3.8
43	100	32.2	0	100	77.2	2.2
42	100	30.4	0	100	75.7	0.7
41.6	100	29.6	0	100	75.0	0
41	100	28.5	0	100	73.2	0
40	100	26.6	0	100	70.1	0
39.1	100	25.0	0	100	67.3	0
39	99.9	24.9	0	100	66.9	0
38	99.0	24.0	0	100	63.8	0
37	98.0	23.0	0	100	60.6	0
36	97.1	22.1	0	100	57.5	0
35	96.2	21.2	0	100	54.3	0
34	95.2	20.2	0	100	51.2	0
33	94.3	19.3	0	100	48.1	0
32	93.4	18.4	0	100	44.9	0
31	92.5	17.5	0	100	41.8	0
30	91.5	16.5	0	100	38.6	0
29	90.6	15.6	0	100	35.5	0
28	89.7	14.7	0	100	32.4	0
27	88.8	13.8	0	100	29.2	0
26	87.8	12.8	0	100	26.1	0
25.7	87.5	12.5	0	100	25.0	0
25	86.9	11.9	0	99.0	24.0	0
24	86.0	11.0	0	97.4	22.4	0
23	85.0	10.0	0	95.8	20.8	0
22	84.1	9.1	0	94.2	19.2	0
21	83.2	8.2	0	92.7	17.7	0
20	82.2	7.2	0	91.1	16.1	0
19	81.3	6.3	0	89.5	14.5	0
18	80.4	5.4	0	88.0	13.0	0
17	79.5	4.5	0	86.4	11.4	0
16	78.5	3.5	0	84.8	9.8	0
15	77.6	2.6	0	83.2	8.2	0
14	76.7	1.7	0	79.5	4.5	0
13	75.8	0.8	0	75.8	0.8	0
12.8	75.0	0	0	75.0	0	0
12	69.0	0	0	69.0	0	0
11	61.5	0	0	61.5	0	0
10	54.0	0	0	54.0	0	0
9	46.5	0	0	46.5	0	0
8	39.0	0	0	39.0	0	0
7	31.5	0	0	31.5	0	0
6	24.0	0	0	24.0	0	0
5	16.5	0	0	16.5	0	0
4	9.0	0	0	9.0	0	0
3	1.5	0	0	1.5	0	0
2.8	0	0	0	0	0	0
2	0	0	0	0	0	0
1	0	0	0	0	0	0
0	0	0	0	0	0	0
% FP	CRGP 5	CRGP 6	CRGP 7	CRGP 5	CRGP 6	CRGP 7
	Shutdown Margin Setpoint			Operational Alarm Setpoint		

RI = 300 is withdrawal limit at all power levels.

Oconee 1 Cycle 20
Rod Index Setpoints
3 Pump Operation, No Inoperable Rods, BOC to EOC

% FP	Shutdown Margin Setpoint			Operational Alarm Setpoint		
	CRGP 5	CRGP 6	CRGP 7	CRGP 5	CRGP 6	CRGP 7
77	100	100	27.4	100	100	37.5
76.2	100	100	25.0	100	100	36.4
76	100	99.7	24.7	100	100	36.1
75	100	98.2	23.2	100	100	34.8
74	100	96.8	21.8	100	100	33.5
73	100	95.3	20.3	100	100	32.1
72	100	93.8	18.8	100	100	30.8
71	100	92.3	17.3	100	100	29.5
70	100	90.8	15.8	100	100	28.1
69	100	89.4	14.4	100	100	26.8
68	100	87.9	12.9	100	100	25.5
67.6	100	87.4	12.4	100	100	25.0
67	100	86.4	11.4	100	99.6	24.6
66	100	84.9	9.9	100	98.9	23.9
65	100	83.4	8.4	100	98.2	23.2
64	100	82.0	7.0	100	97.6	22.6
63	100	80.5	5.5	100	96.9	21.9
62	100	79.0	4.0	100	96.2	21.2
61	100	77.5	2.5	100	95.6	20.6
60	100	76.0	1.0	100	94.9	19.9
59.3	100	75.0	0	100	94.4	19.4
59	100	74.1	0	100	94.2	19.2
58	100	71.1	0	100	93.6	18.6
57	100	68.2	0	100	92.9	17.9
56	100	65.2	0	100	92.2	17.2
55	100	62.2	0	100	91.6	16.6
54	100	59.3	0	100	90.9	15.9
53	100	56.3	0	100	90.2	15.2
52	100	53.4	0	100	89.6	14.6
51	100	50.4	0	100	88.9	13.9
50	100	47.4	0	100	88.2	13.2
49	100	44.5	0	100	86.7	11.7
48	100	41.5	0	100	85.1	10.1
47	100	39.6	0	100	83.5	8.5
46	100	37.8	0	100	82.0	7.0
45	100	35.9	0	100	80.4	5.4
44	100	34.1	0	100	78.8	3.8
43	100	32.2	0	100	77.2	2.2
42	100	30.4	0	100	75.7	0.7
41.6	100	29.6	0	100	75.0	0
41	100	28.5	0	100	73.2	0
40	100	26.6	0	100	70.1	0
39.1	100	25.0	0	100	67.3	0
39	99.9	24.9	0	100	66.9	0
38	99	24.0	0	100	63.8	0
37	98	23.0	0	100	60.6	0
36	97.1	22.1	0	100	57.5	0
35	96.2	21.2	0	100	54.3	0
34	95.2	20.2	0	100	51.2	0
33	94.3	19.3	0	100	48.1	0
32	93.4	18.4	0	100	44.9	0
31	92.5	17.5	0	100	41.8	0
30	91.5	16.5	0	100	38.6	0
29	90.6	15.6	0	100	35.5	0
28	89.7	14.7	0	100	32.4	0
% FP	CRGP 5	CRGP 6	CRGP 7	CRGP 5	CRGP 6	CRGP 7
	Shutdown Margin Setpoint			Operational Alarm Setpoint		

RI = 300 is withdrawal limit at all power levels.

Continued on next page.

Oconee 1 Cycle 20
Rod Index Setpoints
3 Pump Operation, No Inoperable Rods, BOC to EOC

% FP	Shutdown Margin Setpoint			Operational Alarm Setpoint		
	CRGP 5	CRGP 6	CRGP 7	CRGP 5	CRGP 6	CRGP 7
27	88.8	13.8	0	100	29.2	0
26	87.8	12.8	0	100	26.1	0
25.7	87.5	12.5	0	100	25.0	0
25	86.9	11.9	0	99.0	24.0	0
24	86.0	11.0	0	97.4	22.4	0
23	85.0	10.0	0	95.8	20.8	0
22	84.1	9.1	0	94.2	19.2	0
21	83.2	8.2	0	92.7	17.7	0
20	82.2	7.2	0	91.1	16.1	0
19	81.3	6.3	0	89.5	14.5	0
18	80.4	5.4	0	88.0	13.0	0
17	79.5	4.5	0	86.4	11.4	0
16	78.5	3.5	0	84.8	9.8	0
15	77.6	2.6	0	83.2	8.2	0
14	76.7	1.7	0	79.5	4.5	0
13	75.8	0.8	0	75.8	0.8	0
12.8	75.0	0	0	75.0	0	0
12	69.0	0	0	69.0	0	0
11	61.5	0	0	61.5	0	0
10	54.0	0	0	54.0	0	0
9	46.5	0	0	46.5	0	0
8	39.0	0	0	39.0	0	0
7	31.5	0	0	31.5	0	0
6	24.0	0	0	24.0	0	0
5	16.5	0	0	16.5	0	0
4	9.0	0	0	9.0	0	0
3	1.5	0	0	1.5	0	0
2.8	0	0	0	0	0	0
2	0	0	0	0	0	0
1	0	0	0	0	0	0
0	0	0	0	0	0	0
% FP	CRGP 5	CRGP 6	CRGP 7	CRGP 5	CRGP 6	CRGP 7
	Shutdown Margin Setpoint			Operational Alarm Setpoint		

RI = 300 is withdrawal limit at all power levels.

Oconee 1 Cycle 20
Rod Index Setpoints
4 Pump Operation, 1 Inoperable Rod, BOC to EOC

% FP	Shutdown Margin Setpoint			Operational Alarm Setpoint		
	CRGP 5	CRGP 6	CRGP 7	CRGP 5	CRGP 6	CRGP 7
102	100	100	83.4	100	100	83.4
101	100	100	82.5	100	100	82.5
100	100	100	81.5	100	100	81.5
99	100	100	80.5	100	100	80.5
98	100	100	79.6	100	100	79.6
97	100	100	78.6	100	100	78.6
96	100	100	77.7	100	100	77.7
95	100	100	76.7	100	100	76.7
94	100	100	75.7	100	100	75.7
93	100	100	74.8	100	100	74.8
92	100	100	73.8	100	100	73.8
91	100	100	72.8	100	100	72.9
90	100	100	71.9	100	100	71.9
89	100	100	70.9	100	100	70.9
88	100	100	70.0	100	100	70.0
87	100	100	69.0	100	100	69.0
86	100	100	68.0	100	100	68.1
85	100	100	67.1	100	100	67.1
84	100	100	66.1	100	100	66.1
83	100	100	65.2	100	100	65.2
82	100	100	64.2	100	100	64.2
81	100	100	63.2	100	100	63.3
80	100	100	62.3	100	100	62.3
79	100	100	61.3	100	100	61.3
78	100	100	60.3	100	100	60.4
77	100	100	59.4	100	100	59.4
76	100	100	58.4	100	100	58.4
75	100	100	57.5	100	100	57.5
74	100	100	56.5	100	100	56.5
73	100	100	55.5	100	100	55.6
72	100	100	54.6	100	100	54.6
71	100	100	53.6	100	100	53.6
70	100	100	52.7	100	100	52.7
69	100	100	51.7	100	100	51.7
68	100	100	50.7	100	100	50.7
67	100	100	49.8	100	100	49.8
66	100	100	48.8	100	100	48.8
65	100	100	47.8	100	100	47.8
64	100	100	46.9	100	100	46.9
63	100	100	45.9	100	100	45.9
62	100	100	45.0	100	100	45.0
61	100	100	44.0	100	100	44.0
60	100	100	43.0	100	100	43.0
59	100	100	42.1	100	100	42.1
58	100	100	41.1	100	100	41.1
57	100	100	40.2	100	100	40.2
56	100	100	39.2	100	100	39.2
55	100	100	38.2	100	100	38.2
54	100	100	37.3	100	100	37.3
53	100	100	36.3	100	100	36.3
52	100	100	35.3	100	100	35.3
51	100	100	34.4	100	100	34.4
50	100	100	33.4	100	100	33.4
49	100	100	32.5	100	100	32.5
48	100	100	31.5	100	100	31.5
% FP	CRGP 5	CRGP 6	CRGP 7	CRGP 5	CRGP 6	CRGP 7
	Shutdown Margin Setpoint			Operational Alarm Setpoint		

RI = 300 is withdrawal limit at all power levels.

Continued on next page.

Oconee 1 Cycle 20
Rod Index Setpoints
4 Pump Operation, 1 Inoperable Rod, BOC to EOC

% FP	Shutdown Margin Setpoint			Operational Alarm Setpoint		
	CRGP 5	CRGP 6	CRGP 7	CRGP 5	CRGP 6	CRGP 7
47	100	100	29.5	100	100	29.5
46	100	100	27.5	100	100	27.5
45	100	100	25.5	100	100	25.5
44.8	100	100	25.0	100	100	25.0
44	100	99.2	24.2	100	99.2	24.2
43	100	98.2	23.2	100	98.2	23.2
42	100	97.2	22.2	100	97.2	22.2
41	100	96.2	21.2	100	96.2	21.2
40	100	95.2	20.2	100	95.2	20.2
39	100	94.2	19.2	100	94.2	19.2
38	100	93.2	18.2	100	93.2	18.2
37	100	92.2	17.2	100	92.2	17.2
36	100	91.2	16.2	100	91.2	16.2
35	100	90.2	15.2	100	90.2	15.2
34	100	89.2	14.2	100	89.2	14.2
33	100	88.2	13.2	100	88.2	13.2
32	100	87.2	12.2	100	87.2	12.2
31	100	86.2	11.2	100	86.2	11.2
30	100	85.2	10.2	100	85.2	10.2
29	100	84.2	9.2	100	84.2	9.2
28	100	83.2	8.2	100	83.2	8.2
27	100	82.2	7.2	100	82.2	7.2
26	100	81.2	6.2	100	81.2	6.2
25	100	80.2	5.2	100	80.2	5.2
24	100	79.2	4.2	100	79.2	4.2
23	100	78.2	3.2	100	78.2	3.2
22	100	77.2	2.2	100	77.2	2.2
21	100	76.2	1.2	100	76.2	1.2
20	100	75.2	0.2	100	75.2	0.2
19.8	100	75.0	0	100	75.0	0
19	100	73.5	0	100	73.5	0
18	100	71.5	0	100	71.5	0
17	100	69.5	0	100	69.5	0
16	100	67.5	0	100	67.5	0
15	100	65.5	0	100	65.5	0
14	100	63.5	0	100	63.5	0
13	100	61.5	0	100	61.5	0
12	100	53.0	0	100	53.0	0
11	100	44.5	0	100	44.5	0
10	100	36.0	0	100	36.0	0
9	100	27.5	0	100	27.5	0
8.7	100	25.0	0	100	25.0	0
8	97.0	22.0	0	97.0	22.0	0
7	92.8	17.8	0	92.8	17.8	0
6	88.5	13.5	0	88.5	13.5	0
5	84.2	9.2	0	84.2	9.2	0
4	80.0	5.0	0	80.0	5.0	0
3	75.8	0.8	0	75.8	0.8	0
2.8	75.0	0	0	75.0	0	0
2	68.0	0	0	68.0	0	0
1	59.5	0	0	59.5	0	0
0	51.0	0	0	51.0	0	0
% FP	CRGP 5	CRGP 6	CRGP 7	CRGP 5	CRGP 6	CRGP 7
	Shutdown Margin Setpoint			Operational Alarm Setpoint		

RI = 300 is withdrawal limit at all power levels.

Oconee 1 Cycle 20
Rod Index Setpoints
3 Pump Operation, 1 Inoperable Rod, BOC to EOC

% FP	Shutdown Margin Setpoint			Operational Alarm Setpoint		
	CRGP 5	CRGP 6	CRGP 7	CRGP 5	CRGP 6	CRGP 7
77	100	100	85.2	100	100	85.2
76	100	100	83.4	100	100	83.4
75	100	100	81.5	100	100	81.5
74	100	100	79.6	100	100	79.6
73	100	100	77.8	100	100	77.8
72	100	100	75.9	100	100	75.9
71	100	100	74.1	100	100	74.1
70	100	100	72.2	100	100	72.2
69	100	100	70.4	100	100	70.4
68	100	100	68.5	100	100	68.5
67	100	100	66.7	100	100	66.7
66	100	100	64.8	100	100	64.8
65	100	100	63.0	100	100	63.0
64	100	100	61.1	100	100	61.1
63	100	100	59.3	100	100	59.3
62	100	100	57.4	100	100	57.4
61	100	100	55.6	100	100	55.6
60	100	100	53.7	100	100	53.7
59	100	100	51.9	100	100	51.9
58	100	100	50.0	100	100	50.0
57	100	100	48.2	100	100	48.2
56	100	100	46.3	100	100	46.3
55	100	100	44.5	100	100	44.5
54	100	100	42.6	100	100	42.6
53	100	100	40.8	100	100	40.8
52	100	100	38.9	100	100	38.9
51	100	100	37.1	100	100	37.1
50	100	100	35.2	100	100	35.2
49	100	100	33.4	100	100	33.4
48	100	100	31.5	100	100	31.5
47	100	100	29.5	100	100	29.5
46	100	100	27.5	100	100	27.5
45	100	100	25.5	100	100	25.5
44.8	100	100	25.0	100	100	25.0
44	100	99.2	24.2	100	99.2	24.2
43	100	98.2	23.2	100	98.2	23.2
42	100	97.2	22.2	100	97.2	22.2
41	100	96.2	21.2	100	96.2	21.2
40	100	95.2	20.2	100	95.2	20.2
39	100	94.2	19.2	100	94.2	19.2
38	100	93.2	18.2	100	93.2	18.2
37	100	92.2	17.2	100	92.2	17.2
36	100	91.2	16.2	100	91.2	16.2
35	100	90.2	15.2	100	90.2	15.2
34	100	89.2	14.2	100	89.2	14.2
33	100	88.2	13.2	100	88.2	13.2
32	100	87.2	12.2	100	87.2	12.2
31	100	86.2	11.2	100	86.2	11.2
30	100	85.2	10.2	100	85.2	10.2
29	100	84.2	9.2	100	84.2	9.2
28	100	83.2	8.2	100	83.2	8.2
27	100	82.2	7.2	100	82.2	7.2
26	100	81.2	6.2	100	81.2	6.2
25	100	80.2	5.2	100	80.2	5.2
24	100	79.2	4.2	100	79.2	4.2
% FP	CRGP 5	CRGP 6	CRGP 7	CRGP 5	CRGP 6	CRGP 7
	Shutdown Margin Setpoint			Operational Alarm Setpoint		

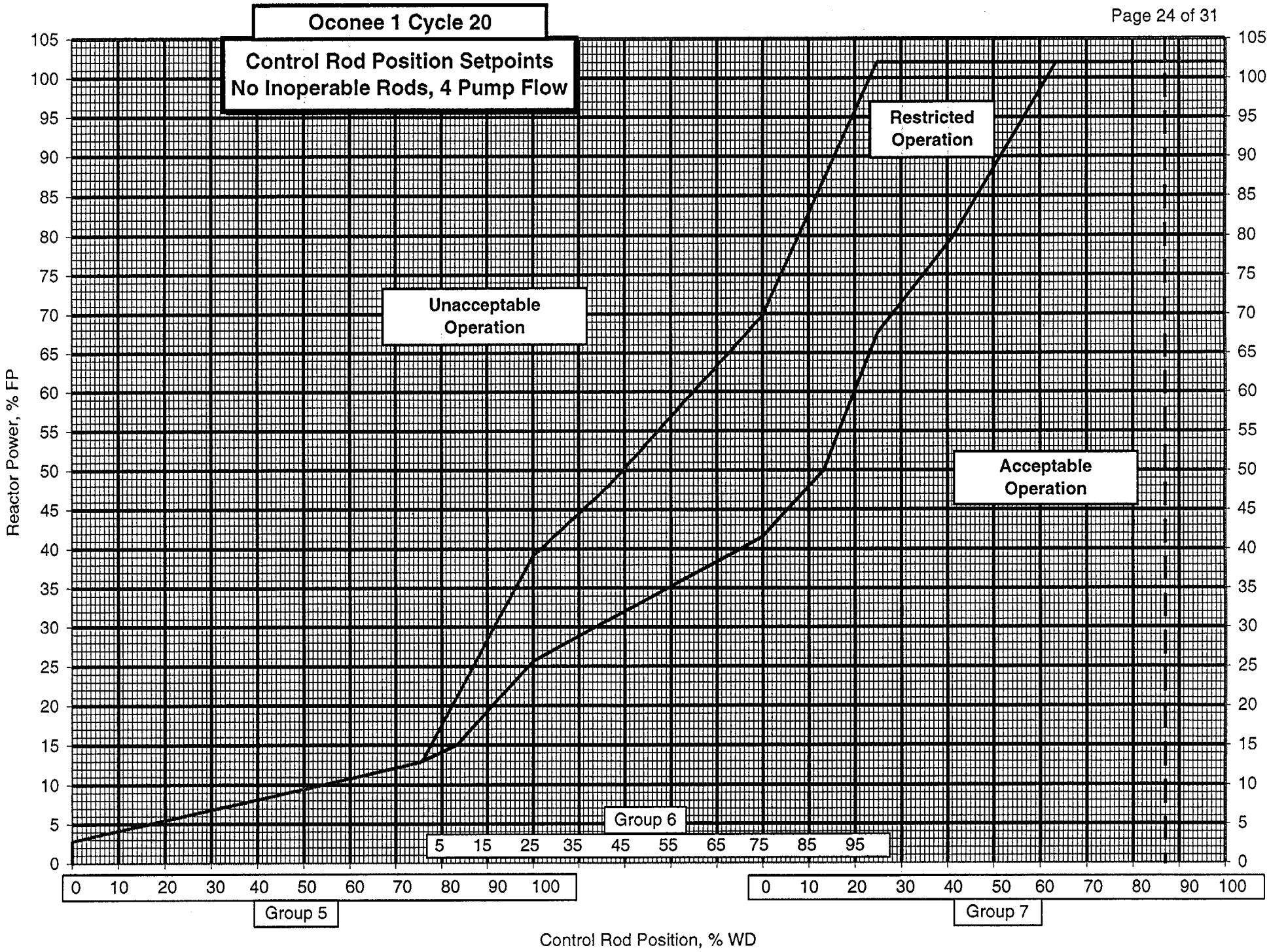
RI = 300 is withdrawal limit at all power levels.

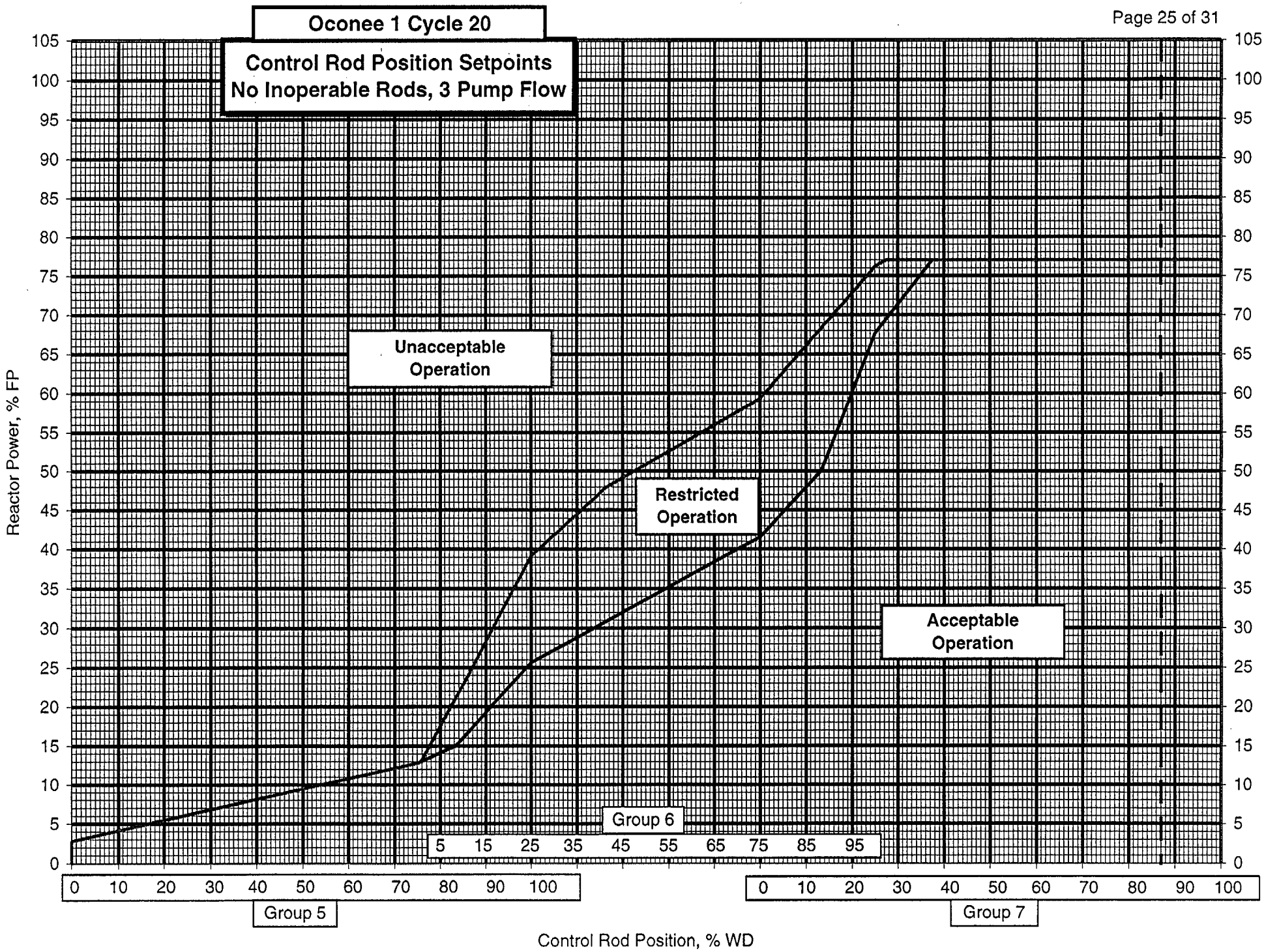
Continued on next page.

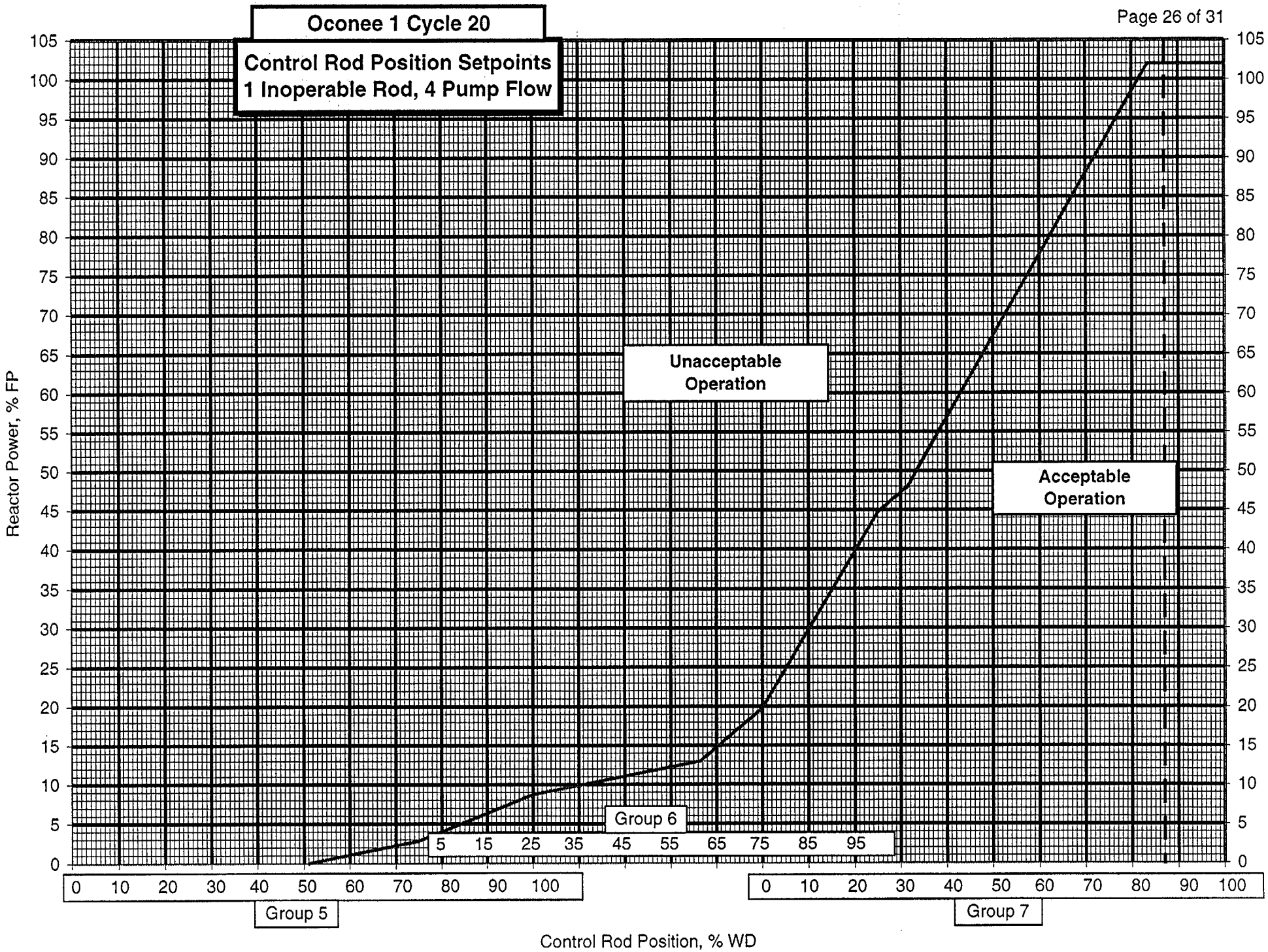
Oconee 1 Cycle 20
Rod Index Setpoints
3 Pump Operation, 1 Inoperable Rod, BOC to EOC

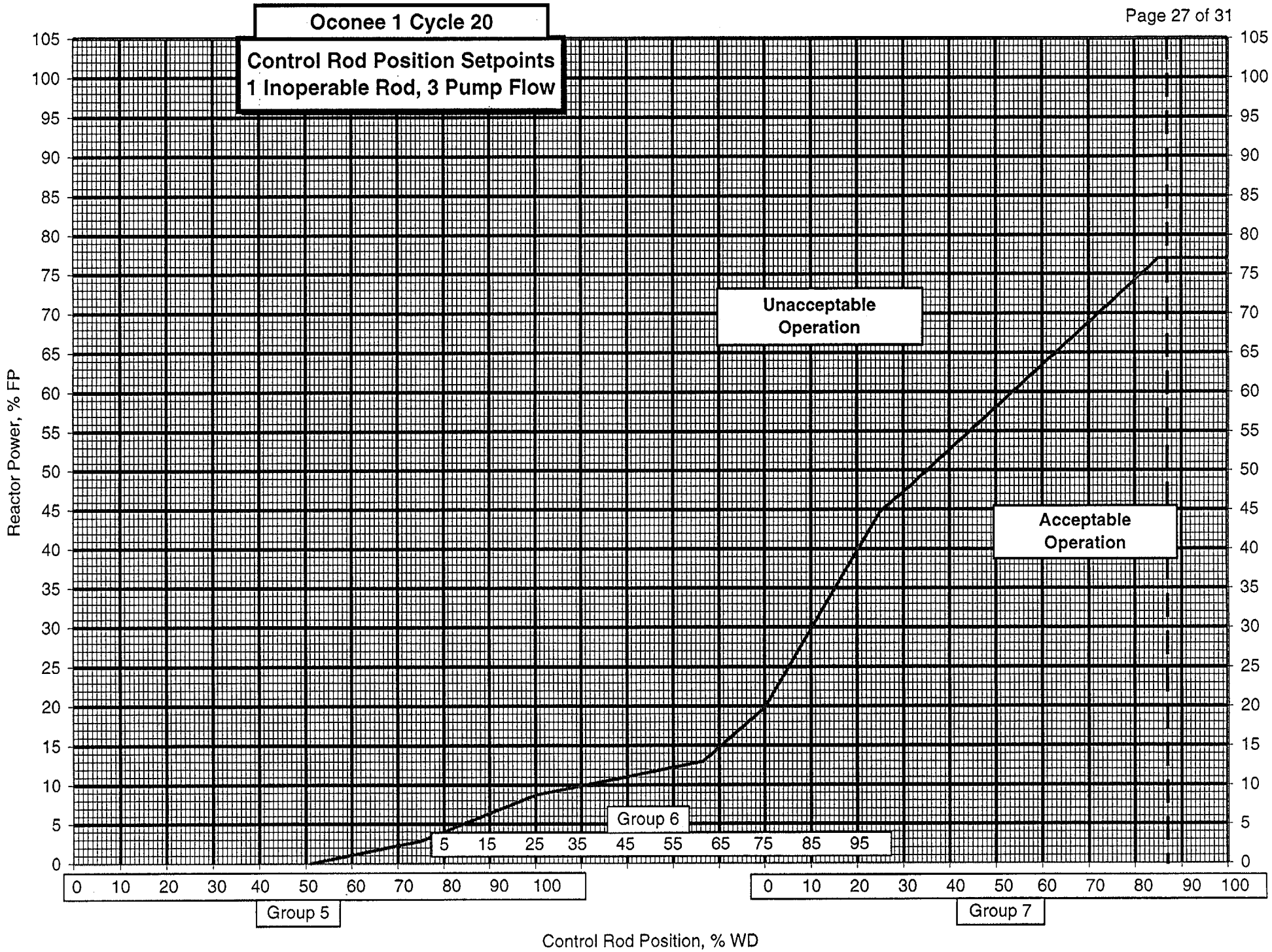
% FP	Shutdown Margin Setpoint			Operational Alarm Setpoint		
	CRGP 5	CRGP 6	CRGP 7	CRGP 5	CRGP 6	CRGP 7
23	100	78.2	3.2	100	78.2	3.2
22	100	77.2	2.2	100	77.2	2.2
21	100	76.2	1.2	100	76.2	1.2
20	100	75.2	0.2	100	75.2	0.2
19.8	100	75.0	0	100	75.0	0
19	100	73.5	0	100	73.5	0
18	100	71.5	0	100	71.5	0
17	100	69.5	0	100	69.5	0
16	100	67.5	0	100	67.5	0
15	100	65.5	0	100	65.5	0
14	100	63.5	0	100	63.5	0
13	100	61.5	0	100	61.5	0
12	100	53.0	0	100	53.0	0
11	100	44.5	0	100	44.5	0
10	100	36.0	0	100	36.0	0
9	100	27.5	0	100	27.5	0
8.7	100	25.0	0	100	25.0	0
8	97.0	22.0	0	97.0	22.0	0
7	92.8	17.8	0	92.8	17.8	0
6	88.5	13.5	0	88.5	13.5	0
5	84.2	9.2	0	84.2	9.2	0
4	80.0	5.0	0	80.0	5.0	0
3	75.8	0.8	0	75.8	0.8	0
2.8	75.0	0	0	75.0	0	0
2	68.0	0	0	68.0	0	0
1	59.5	0	0	59.5	0	0
0	51.0	0	0	51.0	0	0
% FP	CRGP 5	CRGP 6	CRGP 7	CRGP 5	CRGP 6	CRGP 7
	Shutdown Margin Setpoint			Operational Alarm Setpoint		

RI = 300 is withdrawal limit at all power levels.


Oconee 1 Cycle 20
Control Rod Position Setpoints
No Inoperable Rods, 4 Pump Flow
Restricted Operation
Unacceptable Operation
Acceptable Operation
Reactor Power, % FP
0 5 10 15 20 25 30 35 40 45 50 55 60 65 70 75 80 85 90 95 100 105
Group 6
5 15 25 35 45 55 65 75 85 95
0 10 20 30 40 50 60 70 80 90 100
Group 5
0 10 20 30 40 50 60 70 80 90 100
Group 7
Control Rod Position, % WD


Oconee 1 Cycle 20
Control Rod Position Setpoints
No Inoperable Rods, 3 Pump Flow
Unacceptable Operation
Restricted Operation
Acceptable Operation
Reactor Power, % FP
105
100
95
90
85
80
75
70
65
60
55
50
45
40
35
30
25
20
15
10
5
0
Group 6
5 15 25 35 45 55 65 75 85 95
0 10 20 30 40 50 60 70 80 90 100
Group 5
0 10 20 30 40 50 60 70 80 90 100
Group 7
Control Rod Position, % WD


Oconee 1 Cycle 20
Control Rod Position Setpoints
1 Inoperable Rod, 4 Pump Flow
Unacceptable Operation
Acceptable Operation
Reactor Power, % FP
105
100
95
90
85
80
75
70
65
60
55
50
45
40
35
30
25
20
15
10
5
0
Group 6
5 15 25 35 45 55 65 75 85 95
0 10 20 30 40 50 60 70 80 90 100
Group 5
0 10 20 30 40 50 60 70 80 90 100
Group 7
Control Rod Position, % WD


Oconee 1 Cycle 20
Control Rod Position Setpoints
1 Inoperable Rod, 3 Pump Flow
Unacceptable Operation
Acceptable Operation
Reactor Power, % FP
105
100
95
90
85
80
75
70
65
60
55
50
45
40
35
30
25
20
15
10
5
0
Group 6
5 15 25 35 45 55 65 75 85 95
0 10 20 30 40 50 60 70 80 90 100
Group 5
0 10 20 30 40 50 60 70 80 90 100
Group 7
Control Rod Position, % WD

Oconee 1 Cycle 20

2.0 Core Operating Limits -- Not Error Adjusted

The data provided on the following pages satisfies a licensing commitment to identify specific parameters before instrumentation uncertainties are incorporated.

References provided in section 1 of this COLR identify the sources for the data which follows.

Information provided in this section should not be used in plant procedures.

Quadrant Power Tilt Limits

Referred to by ITS 3.2.3.

	Steady State		Transient		Maximum
Core Power Level, %FP	30 - 100	0 - 30	30 - 100	0 - 30	0 - 100
Quadrant Power Tilt, %	5.04	10.00	9.44	12.00	20.00

Variable Low RCS Pressure Protective Limits

Referred to by ITS 2.1.1.

Core Outlet Pressure psia	Reactor Coolant Outlet Temperature, °F 3 RCS Pumps	4 RCS Pumps
1800	581.0	578.3
1900	590.0	587.3
2000	598.9	596.3
2100	607.9	605.2
2200	616.9	614.2
2300	625.9	623.2

Oconee 1 Cycle 20

Axial Power Imbalance Protective Limits

Referred to by ITS 2.1.1

Not for Plant Use

	%FP	RPS	Operational
4 Pumps	0	-48.0	-42.8
	80	-	-42.8
	90	-	-39.7
	100	-48.0	-30.0
	112	-31.1	-
	112	31.1	-
	100	48.0	30.0
	90	-	41.0
	80	-	41.0
	0	48.0	41.0
3 Pumps	0	-48.0	-42.8
	74.6	-48.0	-
	77.0	-	-42.8
	86.6	-31.1	-
	86.6	31.1	-
	77.0	-	41.0
	74.6	48.0	-
	0	48.0	41.0

Oconee 1 Cycle 20

Rod Index Limits

Referred to by ITS 3.2.1

Not for Plant Use

	%FP	Operational RI Insertion Limit	Shutdown Margin RI Insertion Limit No Inop Rod	1 Inop Rod	RI Withdrawal Limit
4 Pumps	102	262	220	280	300
	100	260	-	-	300
	90	250	-	-	300
	80	240	-	-	300
	50	200	140	230	300
	15	90	75	160	300
	5	0	0	75	300
3 Pumps	77	236	220	280	300
	50	200	140	230	300
	15	90	75	160	300
	5	0	0	75	300

Oconee 1 Cycle 20

LOCA Limits

Not for Plant Use

	Core Elevation Feet	LOCA LHR kw/ft Limit Versus Burnup		
Mk-B10F and Mk-B10L Fuel		0 GWd/mtU	30 GWd/mtU	62 GWd/mtU
	0.000	15.6	15.6	11.6
	2.506	16.5	16.5	11.6
	4.264	16.8	16.8	11.6
	6.021	17.0	17.0	11.6
	7.779	17.0	17.0	11.6
	9.536	16.7	16.7	11.6
	12.00	15.8	15.8	11.6
Mk-B11 Fuel		0 GWd/mtU	40 GWd/mtU	62 GWd/mtU
	0.000	15.5	15.5	12.6
	2.506	16.3	16.3	12.6
	4.264	16.5	16.5	12.6
	6.021	16.8	16.8	12.6
	7.779	16.5	16.5	12.6
	9.536	16.2	16.2	12.6
	12.00	15.4	15.4	12.6

ATTACHMENT VII

DESIGN INFORMATION RELATIVE TO
OCONEE UNIT 1 CYCLE 20

TABLE OF CONTENTS

1 INTRODUCTION AND SUMMARY VII-3

2 OPERATING HISTORY VII-4

3 GENERAL DESCRIPTION VII-5

4 FUEL SYSTEM DESIGN VII-10

4.1 Fuel Assembly Mechanical Design VII-10

4.2 Fuel Rod Design VII-10
4.2.1 Cladding Collapse VII-10
4.2.2 Cladding Stress VII-11
4.2.3 Cladding Strain VII-11
4.2.4 Cladding Fatigue VII-11

4.3 Thermal Design VII-11

4.4 Cladding Corrosion VII-11

4.5 Material Compatibility VII-11

5 NUCLEAR DESIGN VII-13

5.1 Physics Characteristics VII-13

5.2 Analytical Input VII-13

5.3 Changes in Nuclear Design VII-13

6 THERMAL-HYDRAULIC DESIGN VII-18

7 ACCIDENT ANALYSIS VII-21

7.1 Safety Analysis VII-21

7.2 ECCS Analysis VII-21

8 PROPOSED CHANGES TO LICENSING BASIS DOCUMENTS VII-26

9 REFERENCES VII-28

List of Tables

Table 4-1 Fuel Design Parameters and Dimensions VII-12
Table 5-1 Oconee 1 Physics Parameters VII-14
Table 5-2 Shutdown Margin Calculation VII-16
Table 6-1 System Uncertainties Included in the Statistical Core Design Analysis VII-19
Table 6-2 Nominal Thermal-Hydraulic Design Conditions VII-20
Table 7-1 Transient Analysis Results VII-23
Table 7-2 LOCA LHR Limits Mark-B10 ..... VII-24
Table 7-3 LOCA LHR Limits Mark-B11 ..... VII-25
Table 8-1 Technical Specification Changes VII-27

List of Figures

Figure 3-1 Core Loading Diagram VII-6
Figure 3-2 Enrichment and Burnup VII-7
Figure 3-3 Control Rod Locations VII-8
Figure 3-4 BPRA Enrichment and Distribution VII-9
Figure 5-1 Two Dimensional Relative Power Distribution
Hot Full Power, BOC, Equilibrium Xenon, Nominal Rod Positions VII-17

1 INTRODUCTION AND SUMMARY

This report justifies the operation of the twentieth cycle of Oconee Nuclear Station, Unit 1, at the rated core power of 2568 MWth. Included are the required analyses as outlined in the USNRC document "Guidance for Proposed License Amendments Relating to Refueling", June 1975.

Cycle 20 for Oconee Unit 1 will be the first Oconee cycle to utilize Mark-B11 fuel including M5 cladding. To support Cycle 20 operation of Oconee Unit 1, this report employs analytical techniques and design bases established in reports that have been previously submitted to the USNRC. The Duke Power non-LOCA transient analysis methods are documented in topical report DPC-NE-3005 (Reference 1).

Section 2 of this report describes the operating history for fuel in Oconee Unit 1. Section 3 is a general description of the reactor core, and the fuel system design is provided in Section 4. Reactor and system parameters and conditions are summarized in Sections 5, 6 and 7. All of the accidents analyzed in the UFSAR (Reference 2) have been reviewed for Cycle 20 operation. In those cases where Cycle 20 characteristics were conservative compared to those analyzed for the generic analysis, a new analysis was not performed. Changes to the Technical Specifications and the Core Operating Limits Report (COLR) are provided in Section 8.

The Technical Specifications have been reviewed, and the modifications for Cycle 20 are justified in this report. Based on the analyses performed, it has been concluded that Oconee Unit 1 Cycle 20 can be safely operated at a core power level of 2568 MWth.

2 OPERATING HISTORY

The reference fuel cycle for the nuclear and thermal-hydraulic analyses of Oconee Unit 1, Cycle 20, is the currently operating Cycle 19. Cycle 19 achieved initial criticality on July 6, 1999. The fuel cycle design length for Cycle 20 - 450 $\pm$ 10 EFPD - is based on an assumed Cycle 19 length of 490 $\pm$ 10 EFPD. No operating anomalies have occurred during previous cycle operations that would adversely affect fuel performance in Cycle 20.

Cycle 20 will operate in a feed-and-bleed mode for its entire design length, as did Cycle 19.

3 GENERAL DESCRIPTION

The Oconee Unit 1 reactor core and fuel design bases are described in detail in Chapter 4 of the UFSAR (Reference 2). The Cycle 20 core consists of 177 fuel assemblies, each of which is a 15 by 15 array containing 208 fuel rods, 16 control rod guide tubes, and one incore instrument guide tube. The Cycle 20 (Batch 22) fuel consists of dished-end, cylindrical pellets of uranium dioxide clad in M5 cladding. All other fuel assemblies consist of dished-end, cylindrical pellets of uranium dioxide clad in cold-worked Zircaloy-4. The Batch 22 fuel assemblies have an average nominal fuel loading of 459.0 kg uranium. All other fuel assemblies have an average nominal fuel loading of 487.2 kg uranium. The undensified nominal active fuel lengths, theoretical densities, fuel and fuel rod dimensions, and other related fuel parameters are given in Table 4-1.

Figure 3-1 is the core loading diagram for Oconee 1, Cycle 20. The 12 assemblies remaining from the original 60 included in Batch 19 (3.68 wt% ^{235}U) will be designated as Batch 19C. The 45 assemblies remaining from the original 60 included in Batch 20 (3.61 wt% ^{235}U) will be designated as Batch 20B. The 60 Batch 21 (44 at 3.68 and 16 at 4.02 wt% ^{235}U, Batch 21A and 21B, respectively) assemblies will be retained along with the 60 Batch 22 feed assemblies (3.40 wt% ^{235}U). The core periphery is composed of Batch 19 and Batch 20 assemblies. The Batch 22 assemblies are distributed evenly throughout the core interior with the rest of the Batch 20 and Batch 21 assemblies. Figure 3-2 is a quarter-core map showing the fuel assembly burnup and enrichment distribution at the beginning of Cycle 20.

Cycle 20 will operate in a rods-out, feed-and-bleed mode. Core reactivity control is supplied mainly by soluble boron and supplemented by 61 full-length Ag-In-Cd control rods and 44 burnable poison rod assemblies (BPRAs). In addition to the full-length control rods, eight Inconel "gray" axial power shaping rods (APSRs) are provided for additional control of the axial power distribution. The Cycle 20 locations of the 69 control rods and the group designations are indicated in Figure 3-3. The Cycle 20 locations and enrichments of the BPRAs are shown in Figure 3-4.

Figure 3-1

Core Loading Diagram

Oconee 1, Cycle 20

X

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						G-15 19C	G-13 20B	H-01 19C	G-03 20B	G-01 19C					
B				N-10 20B	K-06 20B	L-05 21A	22	K-08 21A	22	L-11 21A	K-10 20B	N-06 20B			
C			N-12 20B	22	O-12 21B	22	C-10 21A	22	C-06 21A	22	O-04 21B	22	N-04 20B		
D		L-12 20B	22	E-08 21A	22	D-09 21A	22	F-02 20B	22	D-07 21A	22	H-11 21A	22	L-04 20B	
E		F-09 20B	N-13 21B	22	P-10 20B	22	M-12 21A	22	M-04 21A	22	L-02 20B	22	N-03 21B	F-07 20B	
F	R-07 19C	E-10 21A	22	K-04 21A	22	M-03 20B	P-09 21B	M-13 20B	P-07 21B	C-05 20B	22	K-12 21A	22	E-06 21A	R-09 19C
G	O-07 20B	22	L-03 21A	22	N-11 21A	K-14 21B	C-08 21A	22	H-13 21A	K-02 21B	N-05 21A	22	L-13 21A	22	O-09 20B
W H	R-08 19C	H-09 21A	22	P-06 20B	22	C-11 20B	22	G-09 20B	22	O-05 20B	22	B-10 20B	22	H-07 21A	A-08 19C Y
K	C-07 20B	22	F-03 21A	22	D-11 21A	G-14 21B	H-03 21A	22	O-08 21A	G-02 21B	D-05 21A	22	F-13 21A	22	C-09 20B
L	A-07 19C	M-10 21A	22	G-04 21A	22	O-11 20B	B-09 21B	E-03 20B	B-07 21B	E-13 20B	22	G-12 21A	22	M-06 21A	A-09 19C
M		L-09 20B	D-13 21B	22	F-14 20B	22	E-12 21A	22	E-04 21A	22	B-06 20B	22	D-03 21B	L-07 20B	
N		F-12 20B	22	H-05 21A	22	N-09 21A	22	L-14 20B	22	N-07 21A	22	M-08 21A	22	F-04 20B	
O			D-12 20B	22	C-12 21B	22	O-10 21A	22	O-06 21A	22	C-04 21B	22	D-04 20B		
P				D-10 20B	G-06 20B	F-05 21A	22	G-08 21A	22	F-11 21A	G-10 20B	D-06 20B			
R						K-15 19C	K-13 20B	H-15 19C	K-03 20B	K-01 19C					

Z

XX X	Previous Cycle Location Batch Number

Figure 3-2

Enrichment and Burnup

Oconee 1, Cycle 20

	8	9	10	11	12	13	14	15
H	3.61 36176	3.40 0	3.61 28879	3.40 0	3.61 29303	3.40 0	3.68 19750	3.68 35666
K	3.40 0	3.68 19354	4.02 16431	3.68 19909	3.40 0	3.68 18727	3.40 0	3.61 33937
L	3.61 28879	4.02 16393	3.61 28887	3.40 0	3.68 20621	3.40 0	3.68 20808	3.68 38821
M	3.40 0	3.68 19887	3.40 0	3.61 29291	3.40 0	4.02 14803	3.61 30840	
N	3.61 29303	3.40 0	3.68 20623	3.40 0	3.68 20321	3.40 0	3.61 36634	
O	3.40 0	3.68 18728	3.40 0	4.02 14827	3.40 0	3.61 34213		
P	3.68 19796	3.40 0	3.68 20810	3.61 30868	3.61 36628			
R	3.68 35665	3.61 33913	3.68 38837					

X.XX XXXXX	Initial Enrichment in wt% ^{235}U BOC Burnup in MWd/mtU

Figure 3-3

Control Rod Locations

Oconee 1, Cycle 20

X

		1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	
	A																
	B						4		6		4						
	C					2		5		5		2					
	D				7		8		7		8		7				
	E			2		5		1		1		5		2			
	F		4		8		6		3		6		8		4		
	G			5		1		3		3		1		5			
W	H		6		7		3		4		3		7		6		Y
	K			5		1		3		3		1		5			
	L		4		8		6		3		6		8		4		
	M			2		5		1		1		5		2			
	N				7		8		7		8		7				
	O					2		5		5		2					
	P						4		6		4						
	R																

Group Number

Z

Group	# of Rods	Function	Group	# of Rods	Function
1	8	Safety	5	12	Control
2	8	Safety	6	8	Control
3	8	Safety	7	8	Control
4	9	Safety	8	8	APSRs
			Total:	**69**	

Figure 3-4

BPRA Enrichment and Distribution

Oconee 1, Cycle 20

	8	9	10	11	12	13	14	15
H		1.1		1.1		1.1		
K	1.1				1.1			
L				1.1		0.8		
M	1.1		1.1		0.8			
N		1.1		0.8				
O	1.1		0.8					
P								
R								

X.X	BPRA Concentration in wt% B_4C in Al_2O_3

4 FUEL SYSTEM DESIGN

4.1 Fuel Assembly Mechanical Design

The reinsert fuel is comprised of Mark-B10F and Mark-B10L fuel. The Mark-B10F (previously Mark-B10T) design was presented in the Oconee 3 Cycle 16 reload report (Ref. 3). The Mark-B10L has radial zoned enrichment and a quick disconnect upper end fitting, but is otherwise similar to the Mark-B10F design.

Oconee 1 Cycle 20 will contain the first reload of Mark-B11 fuel. The Mark-B11 design offers improvements in departure from nucleate boiling (DNB) margins and fuel cycle economy while retaining many proven features of the earlier fuel assembly designs. These features include: keyable spacer grids, floating grid restraint system, flow-optimized control rod guide tube assembly, quick disconnect upper end fitting, anti-straddle lower end fitting, Zircaloy intermediate grids, cruciform holddown spring, and debris resistant fuel rods (extended lower end plug on fuel rods).

The primary design changes, which enhance nuclear, thermal-hydraulic and mechanical performance, include the following:

1. Reduced diameter fuel rod,
2. Flow mixing vanes on five of the six intermediate spacer grids,
3. Improved grid restraint system, and
4. M5 fuel rod cladding

The reduced fuel pin diameter increases uranium utilization, which improves fuel cycle economy. Mixing vane grids increase DNB margin by improving the flow mixing. Grid restraint improvements provide additional structural strength to accommodate the increased hydraulic loads from the flow mixing grids. The M5 fuel rod cladding provides additional corrosion margin.

Table 4-1 depicts fuel design parameters for the fuel operating in Oconee 1 Cycle 20.

4.2 Fuel Rod Design

The mechanical evaluation for the Mark-B10 and Mark-B11 fuel rod designs is discussed in this section.

4.2.1 Cladding Collapse

The creep collapse analysis determines the fuel rod burnup at which the cladding collapses. Therefore, the fuel rod burnup is limited to a value that does not exceed the creep collapse criteria. The methods described in BAW-10084P-A, Rev. 3 (Refs. 4 and, 5) are used to analyze cladding creep collapse.

4.2.2 Cladding Stress

Cladding stress is analyzed with the methods described in BAW-10186P-A (Refs. 6, 7, and 8) and BAW-10179-A (Ref. 9). The analyses show the cladding stresses to be within the limit.

4.2.3 Cladding Strain

The uniform, circumferential strain of the cladding is limited to 1.0% (Ref. 8). The methods described in DPC-NE-2008P-A (Ref. 10) are used to analyze cladding strain. This analysis determines conservative limits on linear heat rate, which ensures that the cladding strain will be less than 1.0%.

4.2.4 Cladding Fatigue

The cladding is limited to a cumulative fatigue usage factor of 90 percent (Ref. 8). Cladding fatigue is analyzed with the methods described in BAW-10186P-A (see Ref. 6). The analyses show that the cumulative usage factor is below the limit.

4.3 Thermal Design

Conservative limits on linear heat rate are used to prevent the centerline fuel temperature from exceeding the fuel melting point. The methods described in DPC-NE-2008P-A are used to determine the linear heat rates that result in centerline fuel melt. Representative limits on linear heat rate are depicted in Table 4-1.

The methods described in DPC-NE-2008P-A are also used to analyze internal fuel rod pressure. The fuel rod pressure is limited to a proprietary value over nominal system pressure or must be less than the pressure that causes cladding liftoff (whichever is more conservative). This analysis determines the fuel rod burnup at which these criteria are exceeded. Therefore, the fuel rod burnup is limited to a value that does not exceed the internal fuel rod pressure criteria.

4.4 Cladding Corrosion

Per Reference 6, cladding corrosion is analyzed with the methods and oxide limit defined in Reference 11. The analyses show that fuel cladding oxide is below the 100 micron limit at the end of Cycle 20.

4.5 Material Compatibility

The Mark-B10F/B10L fuel assemblies and the structural components of the Mark-B11 assemblies do not utilize component materials different from previous cycles. Therefore, the chemical compatibility of all possible fuel-cladding-coolant-assembly interactions is identical to that of previous fuel. Reference 12 determined that there were no material compatibility issues for M5 cladding.

Table 4-1

Fuel Design Parameters and Dimensions

Batch number	19	20	21A	21B	22
Fuel assembly type	B10F	B10L	B10L	B10L	B11
Number of assemblies	12	45	44	16	60
Fuel rod OD, inches	0.430	0.430	0.430	0.430	0.416
Fuel rod ID, inches	0.380	0.380	0.380	0.380	0.368
Flex spacers, type	Spring	Spring	Spring	Spring	Spring
Rigid spacers, type	None	None	None	None	None
Undensified active fuel length, inches	142.29	142.29	142.29	142.29	143.05
Fuel pellet OD (mean spec), inches	0.3735	0.3735	0.3735	0.3735	0.3615
Fuel pellet initial density (mean spec), %TD	96.0	96.0	96.0	96.0	96.0
Initial fuel enrichment, w/o ^{235}U	3.68	3.61	3.68	4.02	3.40
Enrichment of radial zoned rods, w/o ^{235}U	-----	3.31	3.38	3.72	3.10
Axial blanket initial enrichment, w/o ^{235}U	2.00	2.00	2.00	2.00	2.00
Max. EOC pin burnup (MWd/mtU)	45,827	52,201	39,334	37,237	21,441
Average linear heat rate @ 100% of 2568 MW, kw/ft	5.72	5.72	5.72	5.72	5.69
Representative linear heat rate to melt values, kw/ft	21.2	21.2	21.2	21.2	22.0

5 NUCLEAR DESIGN

5.1 Physics Characteristics

Table 5-1 compares the core physics parameters of design Cycle 20 with those of the reference Cycle 19. The Cycle 19 and 20 values were generated by Duke Power Company using the CASMO-3/SIMULATE-3 based reload design methods described in Reference 13. Since the core has not yet reached an equilibrium cycle, differences in core physics parameters are to be expected between the cycles. Figure 5-1 illustrates a representative relative power distribution for the beginning of Cycle 20 at full power with equilibrium xenon and nominal rod positions.

The primary reasons for the differences in the physics parameters between Cycles 19 and 20 are the variation in the shuffle pattern, fresh fuel enrichment, and previous end of cycle fuel assembly burnups for Cycle 20. Differences in ejected and stuck rod worths between cycles are due to changes in the radial flux and burnup distributions. All safety criteria associated with these rod worth's are met. The adequacy of the shutdown margin with Cycle 20 stuck rod worths is demonstrated in Table 5-2. The following conservatisms were applied for the shutdown calculations:

1. Poison material depletion allowance.
2. 10% uncertainty on net rod worth.

Flux redistribution was explicitly incorporated since the shutdown analysis was calculated using a three-dimensional model.

5.2 Analytical Input

The Cycle 20 incore measurement calculation constants to be used to compute core power distributions were obtained using CASMO-3/SIMULATE-3 using the same process that was used for the reference cycle.

5.3 Changes in Nuclear Design

The methodology described in Reference 13 has been implemented for both Oconee 1 Cycle 20 and the reference cycle.

Table 5-1

Oconee 1 Physics Parameters [a]

	Cycle 19 [b]	Cycle 20 [c]
Cycle Length, EFPD (Nominal)	490	450
Cycle Burnup, MWd/mtU (Nominal)	14,649	13,743
Average Core Burnup, EOC, MWd/mtU (Nominal)	31,178	30,868
Initial Core Loading, mtU	85.9	84.6
Critical Boron - BOC (no xenon), ppm		
HZP, groups 7 and 8 at nominal positions	2038	1814
HFP, groups 7 and 8 at nominal positions	1855	1645
Critical Boron - EOC (equilibrium xenon), ppm		
HZP, groups 7 and 8 at nominal positions	343	306
HFP, groups 7 and 8 at nominal positions	7	7
Control Rod Worth - HFP, BOC, %Δk/k		
Group 7	0.890	0.943
Group 8[d]	0.151	0.161
Control Rod Worth - HFP, EOC, %Δk/k		
Group 7	1.050	1.061
Group 8	(e)	(e)
Max Ejected Rod Worth - HZP, %Δk/k [f]		
BOC, groups 5-8 inserted	0.373 (L10)	0.272 (L10)
EOC, groups 5-8 inserted	0.340 (L10)	0.286 (L10)
Max Stuck Rod Worth - HZP, %Δk/k		
BOC	0.961 (N12)	0.951 (N12)
EOC	1.274 (N12)	1.206 (N12)
Power Deficit, HFP to HZP, %Δk/k		
BOC	1.082	1.205
EOC	2.555	2.503
Doppler Coeff - HFP, 10^{-5} (Δk/k-°F)		
BOC (no xenon)	-1.41	-1.39
EOC (equilibrium xenon)	-1.63	-1.59

Table 5-1 (cont'd)

	Cycle 19 [b]	Cycle 20 [c]
Moderator Coeff - HFP, 10^{-4} ($\Delta k/k$-°F)		
BOC (no xenon)	-0.30	-0.34
EOC (equilibrium xenon)	-3.32	-3.20
Boron Worth - HFP, ppm/%$\Delta k/k$		
BOC	151	138
EOC	121	112
Xenon Worth - HFP, %$\Delta k/k$		
BOC (4 days)	2.52	2.57
EOC (equilibrium)	2.77	2.81
Effective delayed neutron fraction - HFP		
BOC	0.00618	0.00615
EOC	0.00514	0.00512

(a) EOC Physics Parameters are provided at the end of the burnup window (nominal + 10 EFPD) except where indicated to be nominal.
(b) Based on a 432 ± 10 EFPD Cycle 18 (Actual Cycle 18 length was 435.43 EFPD).
(c) Based on an assumed Cycle 19 length of 490 ± 10 EFPD.
(d) Worth is calculated from 35% to 100% WD for both cycles.
(e) CRGP8 = 100% WD. Therefore, there is no CRGP8 worth at EOC.
(f) Ejected rod worths for both cycles include a 15% uncertainty penalty.

Table 5-2 Shutdown Margin Calculation for Oconee 1, Cycle 20

	BOC, %Δk/k	EOC, %Δk/k
Available Rod Worth		
Total rod worth, HZP	7.702	8.310
Worth reduction due to poison burnup	-0.400	-0.400
Maximum stuck rod, HZP	-0.951	-1.206
Net worth	6.350	6.704
Less 10% uncertainty	-0.635	-0.670
Total available worth	5.715	6.033
Required Rod Worth		
Power deficit, HFP to HZP	1.205	2.503
Max inserted rod worth, HFP	0.310	0.513
SDM Boron Bias, HFP to HZP	0.302	0.501
Total required worth	1.817	3.517
Shutdown Margin		
Total available worth minus total required worth	3.898	2.516

Note: Required shutdown margin is 1.00% Δk/k.

Figure 5-1

Two Dimensional Relative Power Distribution

Oconee 1, Cycle 20

HFP, 004 EFPD, EQXE
NOMINAL ROD POSITIONS

	8	9	10	11	12	13	14	15
H	1.0427	1.3346	1.1202	1.3681	1.1549	1.3039	0.9500	0.3225
K	1.3347	1.2891	1.3259	1.3150	1.3805	1.2643	1.1001	0.3210
L	1.1202	1.3294	1.1124	1.3584	1.2587	1.2522	0.7364	0.2043
M	1.3680	1.3198	1.3583	1.1423	1.3500	1.1397	0.4361	
N	1.1547	1.3895	1.2597	1.3477	1.1402	0.9727	0.2690	
O	1.3034	1.2682	1.2535	1.1383	0.9712	0.3694		
P	0.9488	1.1012	0.7369	0.4356	0.2685			
R	0.3222	0.3214	0.2044					

6 THERMAL-HYDRAULIC DESIGN

The generic and cycle specific analyses supporting Oconee 1 Cycle 20 operation were performed by Duke Power Company using the methodology described in References 2, 14, 15, and 16. Oconee 1 Cycle 20 was analyzed using Duke's Statistical Core Design (SCD) methodology (Reference 16). Uncertainties on parameters that affect DNB performance are statistically combined to determine a statistical DNBR limit (SDL).

Previous Mark-B10 design fuel assemblies consisted of 0.430 inch diameter fuel rods with 2 Inconel and 6 intermediate non-mixing vane Zircaloy grids. The Mark-B11 fuel assembly design is composed of fuel pins with a 0.416 inch outside diameter, 2 Inconel grids, and 6 intermediate Zircaloy grids of which the upper 5 have mixing vanes. The higher pressure drop and higher cladding surface heat flux of the Mark-B11 design is offset by the larger flow area and the presence of mixing vane grids to result in improved thermal performance.

An SDL of 1.43 was calculated using a set of generic uncertainties specifically calculated for Mark-B10 fuel at Oconee with the BWC CHF correlation, Reference 15. Similarly, an SDL of 1.33 was calculated using a set of generic uncertainties specifically calculated for Mark-B11 fuel at Oconee with the BWU-Z CHF correlation with performance factor, Reference 17. The system parameter uncertainties used in DPC-NE-2005P-A, Rev. 2 (Reference 16) and given in Table 6-1 bound the uncertainties specifically calculated for Oconee. The Oconee 1 Cycle 20 nominal thermal-hydraulic design conditions are given in Table 6-2.

To provide design flexibility, margin is added to the SDL to determine a design DNBR limit (DDL). For generic Mark-B10F/L and Oconee 1 Cycle 20 analyses, the DDL is 1.50 (5.0% above the SDL). For the generic Mark-B11 and Oconee 1 Cycle 20 analyses, the DDL is 1.40 (5.0% margin above the SDL). Currently, no generic DNBR penalties must be assessed against the available margin.

Oconee 1 Cycle 20 will contain the first full batch of Mark-B11 fuel assemblies at Oconee. Therefore, there are technical specification changes required for the use of Mark-B11 fuel. Oconee Improved Technical Specifications Sections 2.1 and Bases Sections B 2.1.1 and B 3.4.1 were updated to add the BWU CHF correlation and its associated limits. Section B 2.1.1 was updated to add the topical BAW-10199P to the reference section. Section 5.6.5 was updated to reference revision 2 of DPC-NE-2005P-A. Section 1.1 of the Core Operating Limits Report (COLR) has been updated, as attached, to include revision 2 of the DPC-NE-2005P-A.

The M5 cladding alloy has no significant impact on DNB analyses and is discussed in Section 4. Technical Specifications Section 4.2.1 was updated to add M5 alloy to the discussion of the alloys used in the fuel rod cladding.

Table 6-1

System Uncertainties Included in the
Statistical Core Design Analysis

Reference 16

Parameter		Uncertainty	Distribution
Core power		+/- 2 %	Normal
RCS flow	4 Pump:	+/- 2.0 %	Normal
	3 Pump:	+/- 3.2 %	
	2 Pump:	+/- 4.2 %	
Pressure		+/- 30 psi	Normal
Inlet Temperature		+/- 2 °F	Normal

Table 6-2

Nominal Thermal-Hydraulic Design Conditions
Oconee 1 Cycle 20

		Generic	Cycle 20
Design Power level, MW_{th}		2568	2568
System pressure, psia		2200	2200
Reactor coolant flow, % design flow		107.5	107.5
Design flow, gpm		352,000	352,000
Core bypass flow, %		7.0	6.3
Vessel temperature at 100% FP, °F			
Inlet		555.8	555.8
Outlet		602.2	602.2
Reference design FΔH		1.714	1.714
Reference design axial shape		1.5 Cosine	1.5 Cosine
Active Fuel Length, inches		142.29 (B10F/L) 143.05 (B11)	142.29 (B10F/L) 143.05 (B11)
Average heat flux at 100%FP, 10^3 Btu/hr-ft^2		178 (B10F/L) 183 (B11)	178 (B10F/L) 183 (B11)
CHF correlation		BWC (B10F/L) BWU-Z, w/PF (B11)	BWC (B10F/L) BWU-Z, w/PF (B11)
Statistical DNBR limit		1.43 (B10F/L) 1.33 (B11)	1.43 (B10F/L) 1.33 (B11)
Design DNBR limit		1.50 (B10F/L) 1.40 (B11)	1.50 (B10F/L) 1.40 (B11)
Hot Channel Factors	Power Factor, Fq	Note 1	Note 1
	Flow Area	Note 1	Note 1

Note:
1. A Fq Hot Channel Factor of 1.0132 (Mark-B10 fuel)/1.0133 (Mark-B11 fuel) and a flow area reduction of 3% was used in the derivation of the SDL.

7 ACCIDENT ANALYSIS

7.1 Safety Analysis

On February 17, 2000, all three Oconee units implemented accident analysis utilizing the Duke Power transient analysis methods. The thermal-hydraulic system transients, for these methods, are based upon those provided in Reference 1. For Oconee 1 Cycle 20, each of the UFSAR accident analysis has been evaluated for the change in the feed batch fuel design. Oconee 1 Cycle 20 will be the first core to utilize a full batch of Mark-B11 fuel. The smaller fuel rod diameter of the Mark-B11 design along with the mixing vane grids results in flow diversion during the transition cores. Transition core penalties have been developed and are applied to the mixed cores for both the Mark-B10 and the Mark-B11 DNB limits.

Each UFSAR accident listed has been evaluated with respect to the changes in feed batch fuel design and the Oconee 1 Cycle 20 reload parameters.

- Startup Accident
- Rod Withdrawal At Power Accident
- Moderator Dilution Accident
- Cold Water Accident
- Loss of Coolant Flow Accidents
- Locked Rotor Accident
- Control Rod Misalignment Accidents
- Turbine Trip Accident
- Steam Generator Tube Rupture Accident
- Rod Ejection Accident
- Steam Line Break Accidents
- Small Steam Line Break Accidents

The results of the reanalysis of these events are provided in Table 7-1 and are compared to the reference analysis Oconee 2 Cycle 18.

The radiological consequences (dose analysis) for the Chapter 15 accident analysis have been evaluated for the change in feed batch design. This evaluation concluded that the change in fuel design does not significantly impact the results of the analyses. Therefore, the conclusions from the reference analysis (Oconee 2 Cycle 18) are not impacted, and the resultant doses will remain within the post-accident acceptance criteria.

7.2 ECCS Analysis

LOCA analyses, applicable to the B&W designed Oconee Units 1, 2 and 3 operated by Duke Power Company, have been performed by Framatome Technologies Incorporated (FTI). The LOCA evaluation model, which has been approved by the NRC, is described in topical report BAW-10192P-A (Reference 18). The LOCA analyses comply with the criteria outlined in 10 CFR 50.46:

1. Peak cladding temperature (PCT) shall not exceed 2200 °F.
2. The percentage of local cladding oxidation shall not exceed 17%.
3. The maximum amount of hydrogen generated during the transient shall not exceed that which would be generated by the oxidation of 1% of the fuel cladding.
4. Calculated changes in core geometry shall be such that the core remains amenable to cooling.
5. The mode of long term cooling shall be established.

In 1999, FTI has identified significant PCT increases associated with both reactor coolant pump (RCP) type and two-phase degradation models used in current Oconee RELAP5-based LOCA linear heat rate (LHR) licensing analyses (PSC 1-99). Duke notified the NRC of this LOCA error via Reference 19. FTI reanalyzed the LHR limits expected to have the most significant PCT increase (BOL Mark-B10 fuel) and found that the PCT could increase by 186 °F, with a maximum PCT of 2150 °F. For Oconee 1 Cycle 20, the LHR limits for the Mark-B10 fuel were reduced to restore PCT margin, and as a result, the limiting PCT for the Mark-B10F fuel is 2050 °F with reductions in LHR limits of between 0.3 kw/ft and 0.6 kw/ft. The final Mark-B10 LHR limits are provided in Table 7-2.

For the Mark-B11 fuel, separate analyses were performed to set the Mark-B11 LHR limits as shown in Table 7-3. These analyses included a mixed core evaluation to determine if a mixed core penalty was required during the transition to Mark-B11 fuel. The evaluation concluded that the small hydraulic differences between the Mark-B11 and Mark-B10 fuel assemblies did not create significantly different cross flow diversion. As a result, no PCT or LHR penalty is needed for the mixed core. The maximum PCT for the analysis of theMark-B11 fuel is 2037 °F.

In Reference 20, LHR adjustments were identified to show compliance with the large break LOCA Evaluation Model (EM). The LHR adjustments are used in the maneuvering analysis to ensue the limiting axial and radial peaking factors are bounded by the analyzed cases. The analyzed cases assume an axial peaking factor of 1.7 and the LHR adjustment provides a means to account for axial peaking factors that are significantly different than 1.7. For Oconee 1, Cycle 20 this approach is used in the maneuvering analysis to ensure that the SER requirements are met.

To address M5 cladding, changes to the FTI LOCA evaluation model were needed. Reference 12 provides these revised models which are used to analyze the Mark B-11 fuel (M5 cladding).

All LOCA results were calculated to be in conformance with the five criteria of 10 CFR 50.46, thus demonstrating acceptable results for the operation of Oconee 1 Cycle 20.

Table 7-1
Transient Analysis Results

Transient		Design Limit	O2C18 Mk-B10	Design Limit	O1C20 Mk-B11
Startup Accident	Peak Thermal Power (% FP)	75 (1)	73	75 (1)	77.6 (2)
	Peak RCS Pressure (psig)	2750	2747	2750	2746
	DNBR	1.5	N/A (3)	1.4	N/A (3)
Rod Withdrawl at Power	DNBR	1.5	1.719	1.4	1.870
	Peak RCS Pressure (psig)	2750	2611.5	2750	2608
Cold Water Accident	DNBR	1.5	N/A (3)	1.4	N/A (3)
	Peak Thermal Power (% FP)	100 (1)	96.7	100 (1)	97.8
	Peak RCS Pressure (psig)	2750	2165	2750	2170
Loss of Flow					
4/4 RCP Coastdown	DNBR	1.69	1.93	1.40	2.06
4/2 RCP Coastdown	DNBR	1.69	1.69 (4)	1.49	1.83
3/1 RCP Coastdown	DNBR	1.77	2.02	1.49	2.17
Locked Rotor					
4 RCP Initial Condition	Peak RCS Pressure (psig)	2750	2451.7	2750	2455.0
	DNBR	1.61	1.50 (4)	1.4	1.635
3 RCP Initial Condition	DNBR	1.62	1.33 (4)	1.4	1.579
Dropped Rod	Peak RCS Pressure (psig)	2750	N/A (7)	2750	N/A (7)
	DNBR	1.5	1.672	1.4	1.843
Turbine Trip	Peak RCS Pressure (psig)	2750	2614.1	2750	2611.8
	DNBR	1.5	N/A (3)	1.4	N/A (3)
Rod Ejection	Peak Enthalpy (cal/gm)	280	132.8	280	131.7
	Peak RCS Pressure (psig)	3000	2885	3000	2929
	Failed Fuel Fraction (%)	50 (5)	40.6	50 (5)	40.95
Large Steam Line Break					
w/ offsite power	DNBR	1.5	3.28	1.4	(6)
w/o offsite power	DNBR	1.5	1.45 (4)	1.4	1.805
Small Steam Line Break					
4 RCP Initial Condition	DNBR	1.5	1.443 (4)	1.4	1.605
3 RCP Initial Condition	DNBR	1.53	1.301 (4)	1.4	1.499

(1) permissible power
(2) Since the peak power exceeded the permissible power, DNB was evaluated and determined to be acceptable.
(3) The DNB acceptance criterion is not challenged during this event.
(4) The results of pin census analysis show that DNBR margin exists for all of the fuel rods.
(5) The actual acceptance criterion is the offsite dose needs to be within 10 CFR 100 criteria. The current dose analysis assumes 50% failed fuel.
(6) This case was not analyzed since the system response was similar to the reference analysis which demonstrated large margins.
(7) The accident response indicates that peak pressure will not be a concern for this event.

Table 7-2
LOCA LHR Limits Mark-B10

ELEVATION, ft	LOCA LHR LIMIT, kW/ft vs. BURNUP		
	0 MWd/mtU	30,000 MWd/mtU	62,000 MWd/mtU
0.000	15.6	15.6	11.6
2.506	16.5	16.5	11.6
4.264	16.8	16.8	11.6
6.021	17.0	17.0	11.6
7.779	17.0	17.0	11.6
9.536	16.7	16.7	11.6
12.00	15.8	15.8	11.6

Notes for Table 7-2:

1) The LHR limits presented above represent the power generated by the pin (i.e. all sources of useable energy caused by the fission process).
2) Linear interpolation for LHR limits is allowed between 30,000 MWd/mtU and 62,000 MWd/mtU.
3) The core endpoint (0 or 12 ft) LHR limits are 95% of the adjacent elevation between BOL and 30,000 MWd/mtU. At 62,000 MWd/mtU the endpoints are equivalent to the constant LHR limit values.
4) LHRs are valid for fuel enrichments between 3.0 and 5.0 $^{w}/_{o}$.
5) The BOL and MOL LOCA limits were calculated using a steady-state EDF of 0.973 for initial core energy deposition and a transient EDF of 1.0. The EOL LOCA limits were calculated using a steady-state EDF of 1.0 and a transient EDF of 1.1.

Table 7-3
LOCA LHR Limits Mark-B11

ELEVATION, ft	LOCA LHR LIMIT, kW/ft vs. BURNUP		
	0 MWd/mtU	40,000 MWd/mtU	62,000 MWd/mtU
0.000	15.5	15.5	12.6
2.506	16.3	16.3	12.6
4.264	16.5	16.5	12.6
6.021	16.8	16.8	12.6
7.779	16.5	16.5	12.6
9.536	16.2	16.2	12.6
12.00	15.4	15.4	12.6

Notes for Table 7-3:

1) The LHR limits presented above represent the power generated by the pin (i.e. all sources of useable energy caused by the fission process).
2) Linear interpolation for LHR limits is allowed between 40,000 MWd/mtU and 62,000 MWd/mtU.
3) The core endpoint (0 or 12 ft) LHR limits are 95% of the adjacent elevation between BOL and 40,000 MWd/mtU. At 62,000 MWd/mtU the endpoints are equivalent to the constant LHR limit values.
4) LHRs are valid for fuel enrichments 3.0 and 5.1 $^{w}/_{o}$.
5) The BOL and MOL LOCA limits were calculated using a steady-state EDF of 0.973 for initial core energy deposition and a transient EDF of 1.0. The EOL LOCA limits were calculated using a steady-state EDF of 1.0 and a transient EDF of 1.1.

8 PROPOSED CHANGES TO LICENSING BASIS DOCUMENTS

Revisions to the Technical Specifications, Core Operating Limits Report (COLR) and the Updated Final Safety Analysis Report (UFSAR) have been proposed for Oconee 1 Cycle 20 operation. These revisions reflect changes to accommodate the change in the feed batch fuel design. Table 8-1 lists the Technical Specification and Bases changes.

To implement Mark-B11 fuel, no COLR changes are required other than typical cycle-specific changes. For completeness, the Oconee Unit 1 Cycle 20 COLR is included in this submittal (Attachment VI).

The UFSAR changes resulting from the changes proposed for Oconee 1 Cycle 20 are not included in this submittal since the NRC has indicated that these changes are not reviewed. The update of the Oconee UFSAR will be implemented under 10 CFR 50.59 and submitted per the requirements of 10 CFR 50.71(e).

Table 8-1

Technical Specification Changes

Specification	Description of Change
ITS 2.1.1.2	Add BWU correlation with its DNBR limit of 1.19.
ITS 4.2.1	Add M5 as one of the cladding materials in the description of the fuel assemblies.
ITS 5.6.5b	Add the M5 topical report (BAW-10227P) to the list of references. Revise the thermal hydraulic statistical core design topical (DPC-NE-2005-PA) to be Rev. 2.
B 2.1.1	Add the BWU correlation to analysis Mark-B11 fuel and its DNBR limit of 1.19. Add BAW-10199 PA to the list of references.
B 3.1.4 (A.2.4)	Revise the limits on ejected rod worth to be consistent with the updated analysis which includes Mark-B11 fuel.
B 3.4.1	Add the DNBR acceptance criterion of $\geq$ 1.19 for the BWU correlation.

9 REFERENCES

1. DPC-NE-3005-P, "UFSAR Chapter 15 Transient Analysis Methodology," Duke Power Company, Oconee Nuclear Station, Rev. 1 (SER dated May 25, 1999).

2. Updated Final Safety Analysis Report, Oconee Nuclear Station, Duke Power Company.

3. Letter from J. W. Hampton (DEC) to Document Control Desk (NRC), Subject: Oconee Nuclear Station, Docket Nos. 50-269, -270, -287, Unit 3 Cycle 16 Reload Technical Specifications, Supplement 4, dated May 2, 1995.

4. Letter from H. N. Berkow (NRC) to M. S. Tuckman (DEC), Subject: Duke Power use of CROV Computer Code, dated June 19, 1995

5. BAW-10084P-A, Rev. 3, Program to determine In-reactor Performance of B&W Fuels – Cladding Creep Collapse, July 1991.

6. Letter from D. LaBarge (NRC) to W. R. McCollum, Jr. (DEC), Subject: Use of Framatome Cogema Fuels Topical Report on High Burnup – Oconee Nuclear Station, Units 1, 2, and 3, (TAC Nos. MA0405, MA0406, and MA0407), dated March 1, 1999.

7. Letter from M. S. Tuckman (DEC) to Document Control Desk (NRC), Subject: Duke Energy Corporation's use of FCF's Extended Burnup Evaluation Topical Report BAW-10186P-A, dated August 25, 1999.

8. BAW-10186P-A, Extended Burnup Evaluation, June 1997.

9. BAW-10179-A, Safety Criteria and Methodology for Acceptable Cycle Reload Analyses, August 1993.

10. DPC-NE-2008P-A, Duke Power Company Fuel Mechanical Reload Analysis Methodology using TACO3, April 1995.

11. Letter from J. H. Taylor (FCF) to Document Control Desk (NRC), Subject: Application of BAW-10186P-A, Extended Burnup Evaluation, dated October 28, 1997.

12. BAW-10227P, Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel, September 1997.

13. DPC-NE-1004A, Nuclear Design Methodology Using CASMO-3 / SIMULATE-3P, Duke Power Company, November 1992.

14. DPC-NE-2003P-A, Oconee Nuclear Stations Core Thermal- Hydraulic Methodology Using VIPRE-01, Duke Power Company, October 1989.

15. BAW-10143P-A, BWC Correlation of Critical Heat Flux, Babcock & Wilcox, April 1985.

16. DPC-NE-2005P-A, Rev. 2, Duke Power Company, Thermal-Hydraulic Statistical Core Design Methodology, June 1999.

17. BAW-10199P, The BWU Critical Heat Flux Correlations, Babcock and Wilcox, Addendum 1, (SER pending).

18. BAW-10192-PA, "BWNT LOCA – BWNT Loss-of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants," Rev. 0, (SER Dated February 18, 1997).

19. Letter from M. S. Tuckman (Duke) to USNRC, "Duke Energy Corporation, Oconee Nuclear Station, Units 1, 2 and 3, Docket Numbers 50-269, 50-270, and 50-287, Report Pursuant to 10CFR50.46, Error in LOCA Analysis," February 4, 1999.

20. Letter from M. S. Tuckman (Duke) to USNRC, "Duke Energy Corporation, Oconee Nuclear Station, Units 1, 2 and 3, Docket Numbers 50-269, 50-270, and 50-287, Report Pursuant to 10CFR50.46, Error Related to Application of the LBLOCA Evaluation Mocel," July 8, 1999.