DMB 016

NOV 23 1984

Dockets Nos. 50-269, 50-270
and 50-287

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Mr. Hal B. Tucker
Vice President - Steam Production
Duke Power Company
Post Office Box 33189
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Tucker:

The Commission has issued the enclosed Amendments Nos.132 , 132, and 129 to Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units Nos. 1, 2 and 3. These amendments consist of changes to the Station's common Technical Specifications (TSs) in response to your request dated September 11, 1984, as supplemented on October 22, 26, and November 1, 1984.

These amendments revise the TSs to support the operation of Oconee Unit 1 at full rated power during the upcoming Cycle 9. The amendments change the following areas:

1. Core Protection Safety Limits (TS 2.1);
2. Protective System Maximum Allowable Setpoints (TS 2.3);
3. Rod Position Limits (TS 3.5.2); and
4. Power Imbalance Limits (TS 3.5.2).

These TS changes are being issued prior to the expiration of the notice period to preclude an unnecessary delay in plant startup from the current outage. In the original submittal, you projected a startup date of November 26, 1984, but in later letters of November 6 and November 16, 1984, you proposed a startup date for November 24, 1984, because Oconee Unit 1 shutdown earlier than scheduled.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance and Final Determination of No Significant Hazards Consideration and Opportunity for Hearing will be included in the Commission's next monthly FEDERAL REGISTER notice.

Sincerely,

Helen Nicolaras, Project Manager
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 132to DPR-38
2. Amendment No. 132to DPR-47
3. Amendment No. 129to DPR-55
4. Safety Evaluation

cc w/enclosures:
See next page

*See previous white for concurrences.

ORB#4:DL	ORB#4:DL	ORB#4:DL	OELD	AD/OR:DL
RIngram	HNicolaras;cf*	JStolz*	Johnson	GLainas
11/21/84	11/14/84	11/14/84	11/21/84	11/23/84

with changes to notice and SER as noted - also note that a new comment period - not required - is being provided

A copy of the Safety Evaluation is also enclosed. Notice of Issuance and Final Determination of No Significant Hazards Consideration and Opportunity for Hearing will be included in the Commission's next monthly FEDERAL REGISTER notice.

Sincerely,

Helen Nicolaras, Project Manager
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. to DPR-38
2. Amendment No. to DPR-47
3. Amendment No. to DPR-55
4. Safety Evaluation

cc w/enclosures:
See next page

ORB#4:DL	ORB#4:DL	ORB#4:DL	OELD	AD/OR:DL
RIngram	HNicolaras;cf	JStolz		GLainas
11/14/84	11/14/84	11/14/84	11/ /84	11/ /84

Duke Power Company

cc w/enclosure(s):

Mr. William L. Porter
Duke Power Company
P. O. Box 33189
422 South Church Street
Charlotte, North Carolina 28242

Office of Intergovernmental Relations
116 West Jones Street
Raleigh, North Carolina 27603

Honorable James M. Phinney
County Supervisor of Oconee County
Walhalla, South Carolina 29621

Mr. James P. O'Reilly, Regional Administrator
U. S. Nuclear Regulatory Commission, Region II
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

Heyward G. Shealy, Chief
Bureau of Radiological Health
South Carolina Department of Health
and Environmental Control
2600 Bull Street
Columbia, South Carolina 29201

Regional Radiation Representative
EPA Region IV
345 Courtland Street, N.E.
Atlanta, Georgia 30308

Mr. J. C. Bryant
Senior Resident Inspector
U.S. Nuclear Regulatory Commission
Route 2, Box 610
Seneca, South Carolina 29678

Mr. Robert B. Borsum
Babcock & Wilcox
Nuclear Power Generation Division
Suite 220, 7910 Woodmont Avenue
Bethesda, Maryland 20814

Manager, LIS
NUS Corporation
2536 Countryside Boulevard
Clearwater, Florida 33515

J. Michael McGarry, III, Esq.
Bishop, Liberman, Cook, Purcell & Reynolds
1200 17th Street, N.W.
Washington, D. C. 20036

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

DUKE POWER COMPANY

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 132
License No. DPR-38

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Duke Power Company (the licensee) dated September 11, 1984, as supplemented on October 22, 26 and November 1, 1984, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-38 is hereby amended to read as follows:

   3.B <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No.132 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 23, 1984

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

DUKE POWER COMPANY

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 132
License No. DPR-47

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Duke Power Company (the licensee) dated September 11, 1984, as supplemented on October 22, 26 and November 1, 1984, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-47 is hereby amended to read as follows:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 132are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 23, 1984

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

DUKE POWER COMPANY

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 129
License No. DPR-55

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Duke Power Company (the licensee) dated September 11, 1984, as supplemented on October 22, 26 and November 1, 1984, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-55 is hereby amended to read as follows:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No.129 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 23, 1984

ATTACHMENTS TO LICENSE AMENDMENTS

AMENDMENT NO.132 TO DPR-38

AMENDMENT NO.132 TO DPR-47

AMENDMENT NO.129 TO DPR-55

DOCKETS NOS. 50-269, 50-270 AND 50-287

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment numbers and contain vertical lines indicating the area of change.

Remove Pages	Insert Pages
2.1-1	2.1-1
2.1-2	2.1-2
2.1-3	2.1-3
2.1-7	2.1-7
2.3-2	2.3-2
2.3-3	2.3-3
2.3-8	2.3-8
2.3-11	2.3-11
3.5-15 (3 pages)	3.5-15 (3 pages)
3.5-18 (3 pages)	3.5-18 (3 pages)
3.5-21 (3 pages)	3.5-21 (3 pages)
3.5-24 (3 pages)	3.5-24 (3 pages)
3.5-27 (3 pages)	3.5-27 (1 page)

2 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS, REACTOR CORE

Applicability

Applies to reactor thermal power, reactor power imbalance, reactor coolant system pressure, coolant temperature, and coolant flow during power operation of the plant.

Objective

To maintain the integrity of the fuel cladding.

Specification

The combination of the reactor system pressure and coolant temperature shall not exceed the safety limit as defined by the locus of points established in Figure 2.1-1A-Unit 1. If the actual pressure/temperature point is below
2.1-1B-Unit 2
2.1-1C-Unit 3
and to the right of the line, the safety limit is exceeded.

The combination of reactor thermal power and reactor power imbalance (power in the top half of the core minus the power in the bottom half of the core expressed as a percentage of the rated power) shall not exceed the safety limit as defined by the locus of points (solid line) for the specified flow set forth in Figure 2.1-2A-Unit 1. If the actual reactor-thermal-power/power
2.1-2B-Unit 2
2.1-2C-Unit 3
imbalance point is above the line for the specified flow, the safety limit is exceeded.

Bases - Unit 1

The safety limits presented for Oconee Unit 1 have been generated using the BAW-2 & BWC critical heat flux (CHF) correlations[1,3]. The BAW-2 correlation applies to fuel batches 9B and 10C while the BWC correlation applies to batches 10B and 11. The reactor coolant system flow rate utilized is 106.5 percent of the design flow (131.32×10^6 lbs/hr) based on four-pump operation.[2]

To maintain the integrity of the fuel cladding and to prevent fission product release, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB). At this point, there is a sharp reduction of the heat transfer coefficient, which would result in high cladding temperatures and the possibility of cladding failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of neutron power, reactor coolant flow, temperature, and pressure can be related to DNB through the use of the CHF correlations (1, 3). The BAW-2 and BWC correlations have been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual 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 1.30 (BAW-2) or 1.18 (BWC). A DNBR of 1.30 (BAW-2) or 1.18 (BWC) corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur; this is considered a conservative margin to DNB for all operating conditions. The difference between the actual core outlet pressure and the indicated reactor coolant system pressure has been considered in determining the core protection safety limits. The difference in these two pressures is nominally 45 psi; however, only a 30 psi drop was assumed in reducing the pressure trip setpoints to correspond to the elevated location where the pressure is actually measured.

The curve presented in Figure 2.1-1A represents the conditions at which the minimum allowable DNBR is predicted to occur for the limiting combination of thermal power and reactor coolant pump configuration. The curve is based upon the design nuclear power peaking factors including potential effects of fuel densification.

The curves of Figure 2.1-2A are based on the more restrictive of two thermal limits and include the effects of potential fuel densification and rod bowing:

1. The combination of the radial peak, axial peak and position of the axial peak that yields no less than the CHF correlation limit.

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limit is 20.5 kw/ft for 9B, and 10C, and 11 Batches of fuel and 17.6 kw/ft for the 10B gadolinia fuel Batch for Unit 1.

Power peaking is not a directly observable quantity and therefore limits have been established on the bases of the reactor power imbalance produced by the power peaking.

The specified flow rates of Figure 2.1-3A correspond to the expected minimum flow rates with four pumps, three pumps, and one pump in each loop, respectively.

A B&W topical report discussing the mechanisms and resulting effects of fuel rod bow has been approved by the NRC (4). The report concludes that the DNBR penalty due to rod bow is insignificant and unnecessary because the power production capability of the fuel decreases with irradiation. Therefore, no rod bow DNBR penalty needs to be considered for thermal-hydraulic analyses.

The maximum thermal power for three-pump operation is 88.8 percent due to a power level trip produced by the flux-flow ratio (74.7 percent flow x 1.08 = 80.67 percent power plus the maximum calibration and instrument error). The maximum thermal power for other coolant pump conditions is produced in a similar manner.

For each curve of Figure 2.1-3A a pressure-temperature point above and to the left of the curve would result in a DNBR greater than the CHF correlation limit or a local quality at the point of minimum DNBR less than the CHF correlation quality limit for that particular reactor coolant pump situation. The curve of Figure 2.1-1A is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in Figure 2.1-3A.

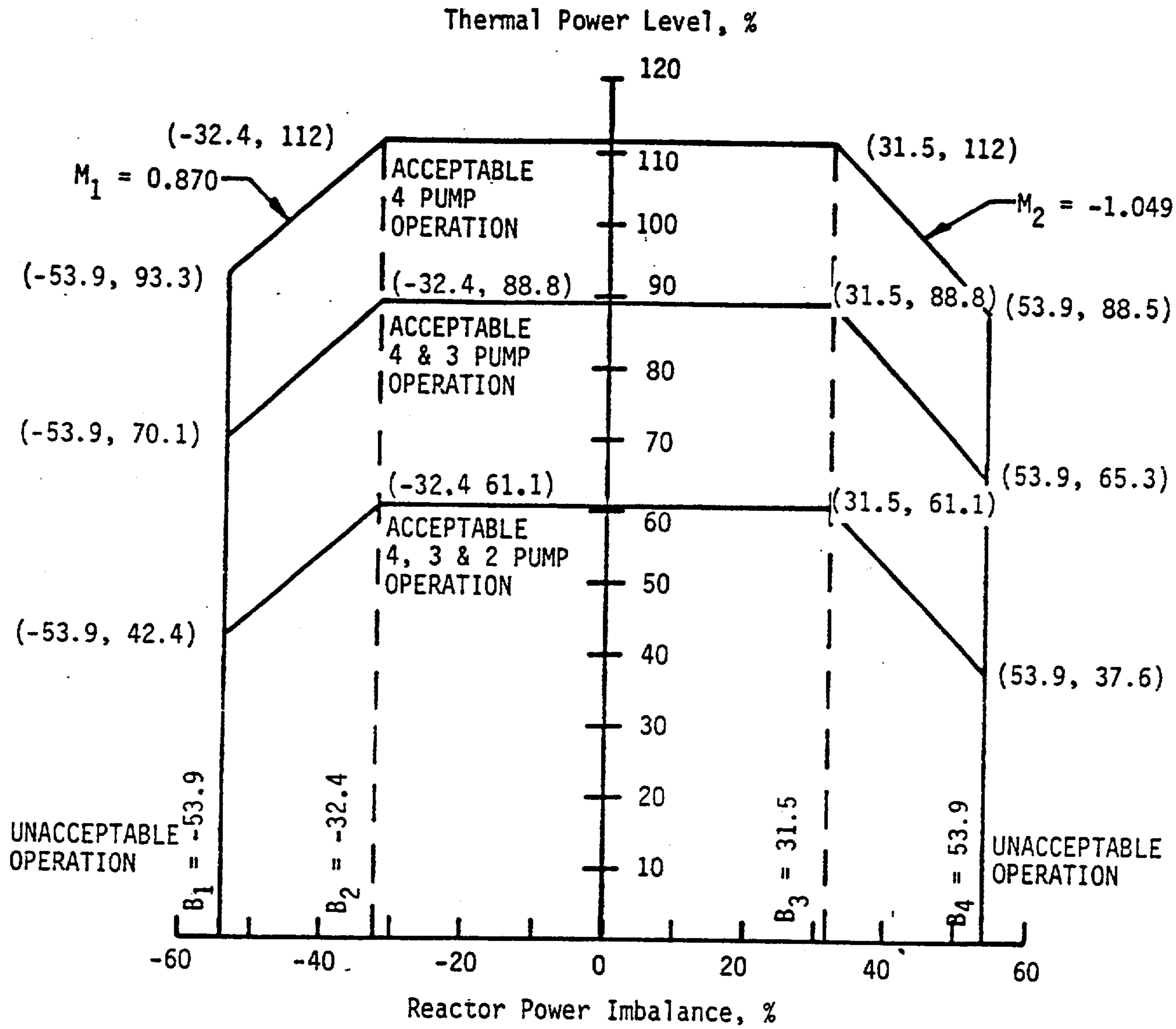
References

(1) Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water, BAW-10000, March, 1970.

(2) Oconee 1, Cycle 4 - Reload Report - BAW-1447, March, 1977.

(3) Correlation of 15x15 Geometry Zircaloy Grid Rod Bundle CHF Data with the BWC Correlation, BAW-10143P, Part 2, Babcock & Wilcox, Lynchburg, Virginia, August 1981.

(4) Fuel Rod Bowing in Babcock & Wilcox Fuel Designs, BAW-10147P-A, Rev. 1, Babcock & Wilcox, May 1983.



CURVE	RC FLOW (GPM)
1	374,880
2	280,035
3	183,690

CORE PROTECTION SAFETY LIMITS
UNIT 1
OCONEE NUCLEAR STATION
Figure 2.1-2A

During normal plant operation with all reactor coolant pumps operating, reactor trip is initiated when the reactor power level reaches 105.5% of rated power. Adding to this the possible variation in trip setpoints due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which is more conservative than the value used in the safety analysis. (4)

Overpower Trip Based on Flow and Imbalance

The power level trip set point produced by the reactor coolant system flow is based on a power-to-flow ratio which has been established to accommodate the most severe thermal transient considered in the design, the loss-of-coolant flow accident from high power. Analysis has demonstrated that the specified power-to-flow ratio is adequate to prevent a DNBR of less than the minimum allowable value should a low flow condition exist due to any electrical malfunction.

The power level trip setpoint produced by the power-to-flow ratio provides both high power level and low flow protection in the event the reactor power level increases or the reactor coolant flow rate decreases. The power level trip setpoint produced by the power-to-flow ratio provides overpower DNB protection for all modes of pump operation. For every flow rate there is a maximum permissible power level, and for every power level there is a minimum permissible low flow rate. Typical power level and low flow rate combinations for the pump situations of Table 2.3-1A are as follows:

1. Trip would occur when four reactor coolant pumps are operating if power is 108% and reactor flow rate is 100%, or flow rate is 92.59% and power level is 100%.

2. Trip would occur when three reactor coolant pumps are operating if power is 80.67% and reactor flow rate is 74.7% or flow rate is 69.44% and power level is 75%.

3. Trip would occur when one reactor coolant pump is operating in each loop (total of two pumps operating) if the power is 52.92% and reactor flow rate is 49.0% or flow rate is 45.37% and the power level is 49%.

   The analyses to determine the flux-to-flow ratios account for calibration and instrument errors and the maximum variation in RC flow in such a manner as to ensure a conservative setpoint. A Monte-Carlo simulation technique is used to determine the combined effects of calibration and instrument uncertainties with the final string uncertainties used in the analyses corresponding to the 95/95 tolerance limits.

The power-imbalance boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking kw/ft limits or DNBR limits. The reactor power imbalance (power in the top half of core minus power in the bottom half of core) reduces the power level trip produced by the power-to-flow ratio such that the boundaries of Figure 2.3-2A - Unit 1 are produced. The power-to-flow ratio reduces the power
2.3-2B - Unit 2
2.3-2C - Unit 3

level trip and associated reactor power/reactor power-imbalance boundaries by 1.08% - Unit 1 for 1% flow reduction.
1.07% - Unit 2
1.08% - Unit 3

Pump Monitors

The pump monitors prevent the minimum core DNBR from decreasing below the minimum allowable value by tripping the reactor due to the loss of reactor coolant pump(s). The circuitry monitoring pump operational status provides redundant trip protection for DNB by tripping the reactor on a signal diverse from that of the power-to-flow ratio. The pump monitors also restrict the power level for the number of pumps in operation.

Reactor Coolant System Pressure

During a startup accident from low power or a slow rod withdrawal from high power, the system high pressure setpoint is reached before the nuclear overpower trip setpoint. The trip setting limit shown in Figure 2.3-1A - Unit 1
2.3-1B - Unit 2
2.3-1C - Unit 3
for high reactor coolant system pressure (2300 psig) has been established to maintain the system pressure below the safety limit (2750 psig) for any design transient. (1)

The low pressure (1800) psig and variable low pressure ($11.14\ T_{out}$-4706) trip
(1800) psig ($11.14\ T_{out}$-4706)
(1800) psig ($11.14\ T_{out}$-4706)
setpoints shown in Figure 2.3-1A have been established to maintain to DNB
2.3-1B
2.3-1C
ratio greater than or equal to the minimum allowable value for those design accidents that result in a pressure reduction. (2,3)

Due to the calibration and instrumentation errors the safety analysis used a variable low reactor coolant system pressure trip value of ($11.14\ T_{out}$ - 4746)
($11.14\ T_{out}$ - 4746)
($11.14\ T_{out}$ - 4746)

Coolant Outlet Temperature

The high reactor coolant outlet temperature trip setting limit (618°F) shown in Figure 2.3-1A has been established to prevent excessive core coolant
2.3-1B
2.3-1C
temperatures in the operating range. Due to calibration and instrumentation errors, the safety analysis used a trip setpoint of 620°F.

Reactor Building Pressure

The high reactor building pressure trip setting limit (4 psig) provides positive assurance that a reactor trip will occur in the unlikely event of a loss-of-coolant accident, even in the absence of a low reactor coolant system pressure trip.

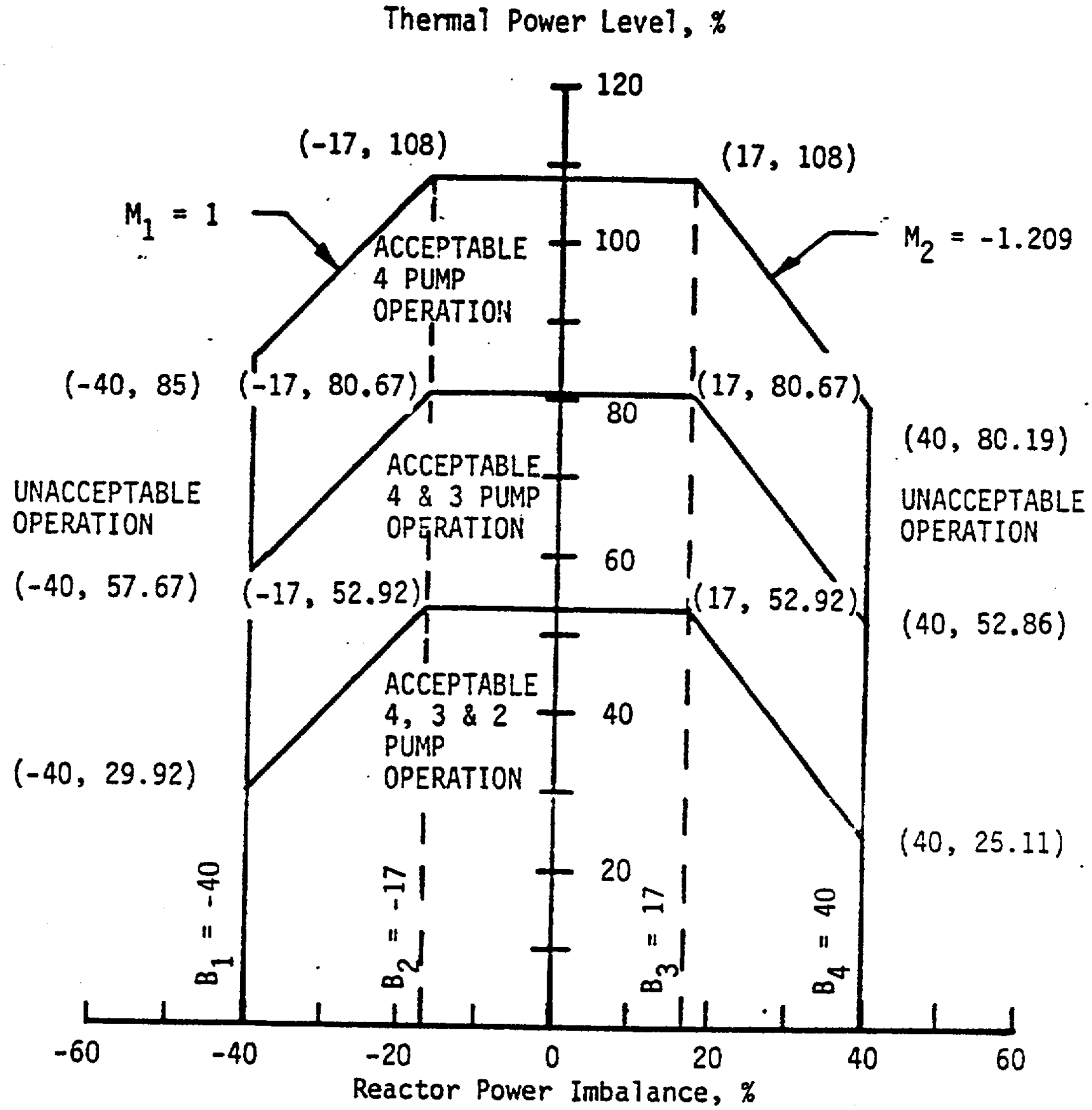
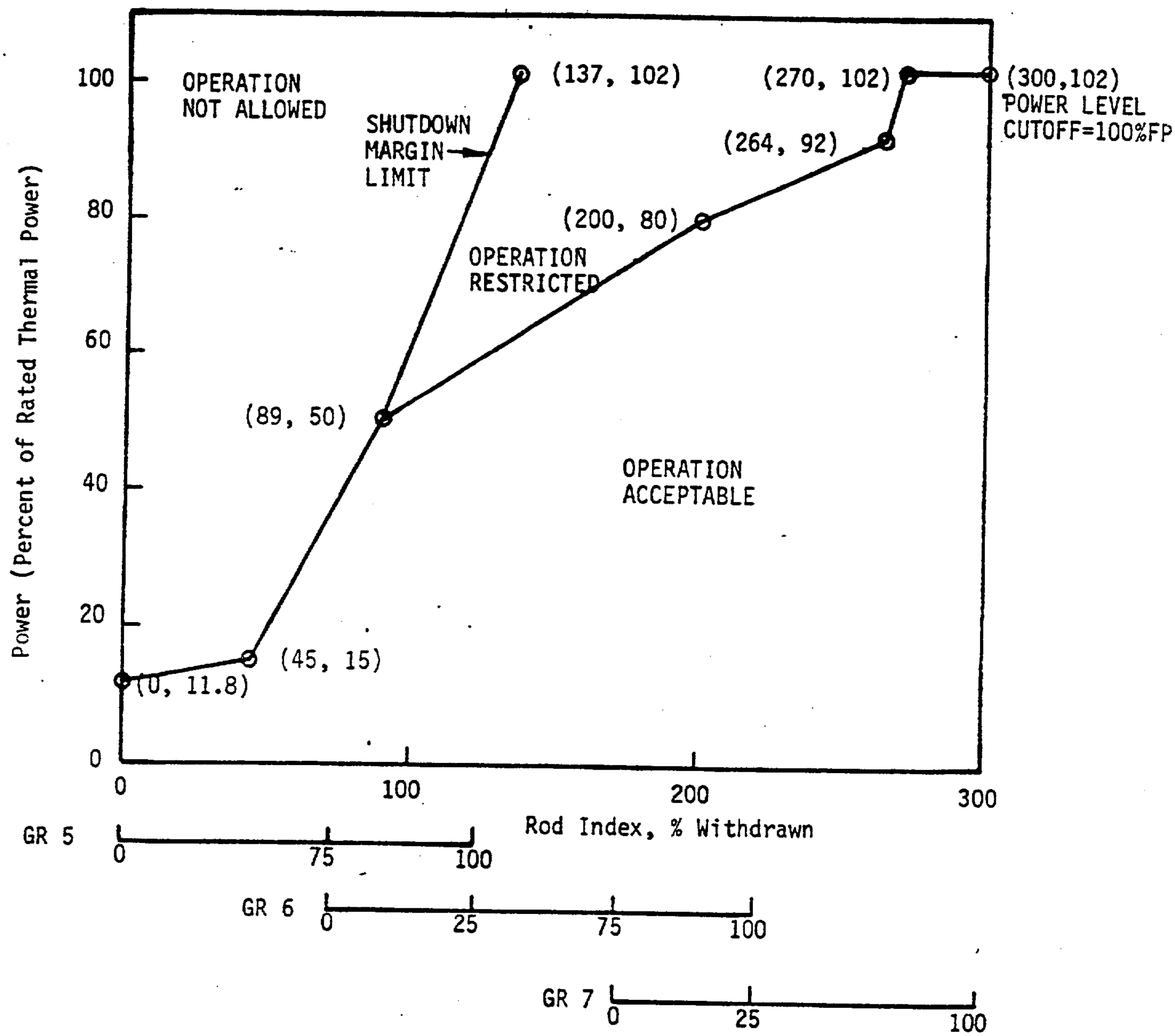

Thermal Power Level, %
120
100
80
60
40
20
(-17, 108)
(17, 108)
M1 = 1
M2 = -1.209
ACCEPTABLE
4 PUMP
OPERATION
(-40, 85)
(-17, 80.67)
(17, 80.67)
(40, 80.19)
ACCEPTABLE
4 & 3 PUMP
OPERATION
UNACCEPTABLE
OPERATION
UNACCEPTABLE
OPERATION
(-40, 57.67)
(-17, 52.92)
(17, 52.92)
(40, 52.86)
ACCEPTABLE
4, 3 & 2
PUMP
OPERATION
(-40, 29.92)
(40, 25.11)
B1 = -40
B2 = -17
B3 = 17
B4 = 40
-60
-40
-20
0
20
40
60
Reactor Power Imbalance, %

Table 2.3-1A
Unit 1

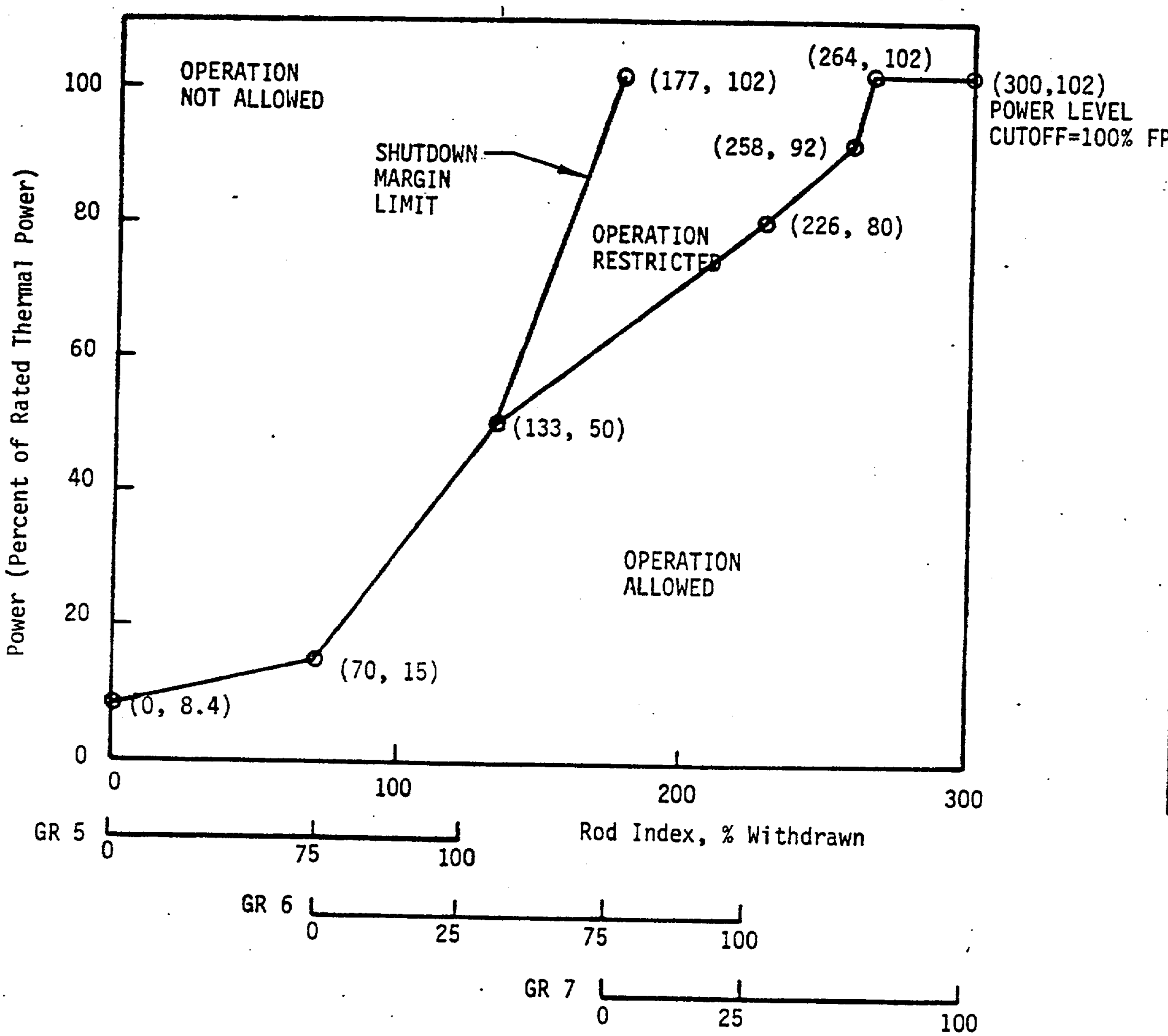
Reactor Protective System Trip Setting Limits

	RPS Segment	Four Reactor Coolant Pumps Operating (Operating Power –100% Rated)	Three Reactor Coolant Pumps Operating (Operating Power -75% Rated)	One Reactor Coolant Pump Operating In Each Loop (Operating Power -49% Rated)	Shutdown Bypass
1.	Nuclear Power Max. (% Rated)	105.5	105.5	105.5	5.0$^{(3)}$
2.	Nuclear Power Max. Based on Flow (2) and Imbalance, (% Rated)	1.08 times flow minus reduction due to imbalance	1.08 times flow minus reduction due to imbalance	1.08 times flow minus reduction due to imbalance	Bypassed
3.	Nuclear Power Max. Based on Pump Monitors, (% Rated)	NA	NA	55%	Bypassed
4.	High Reactor Coolant System Pressure, psig, Max.	2300	2300	2300	1720$^{(4)}$
5.	Low Reactor Coolant System Pressure, psig, Min.	1800	1800	1800	Bypassed
6.	Variable Low Reactor Coolant System Pressure psig, Min.	$(11.14\ T_{out} - 4706)^{(1)}$	$(11.14\ T_{out} - 4706)^{(1)}$	$(11.14\ T_{out} - 4706)^{(1)}$	Bypassed
7.	Reactor Coolant Temp. F., Max.	618	618	618	618
8.	High Reactor Building Pressure, psig, Max.	4	4	4	4

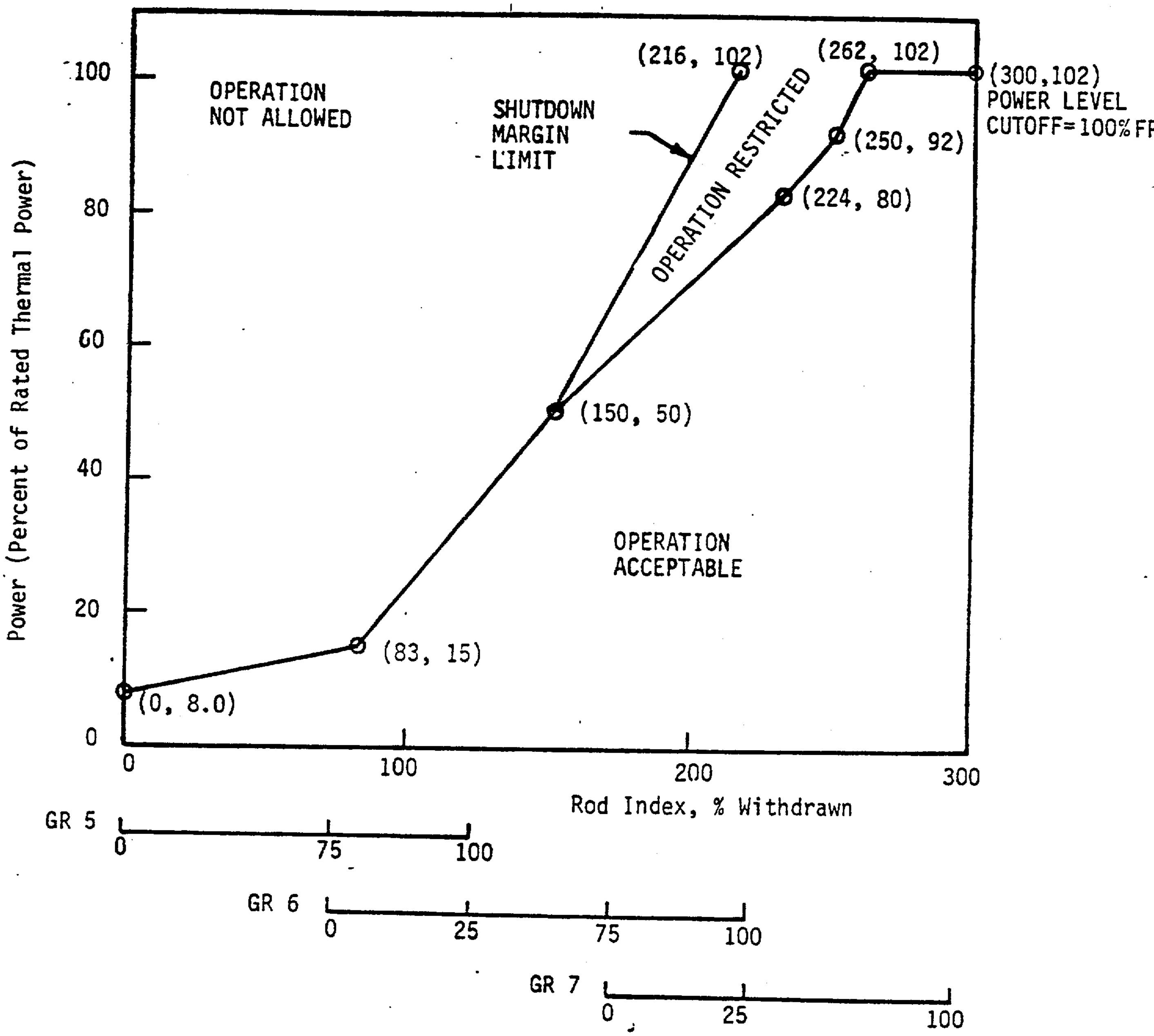
(1) T_{out} is in degrees Fahrenheit (°F).
(2) Reactor Coolant System Flow, %.
(3) Administratively controlled reduction set only during reactor shutdown.
(4) Automatically set when other segments of the RPS are bypassed.



ROD POSITION LIMITS
FOR FOUR PUMP OPERATION
FROM 0 TO 30 + 10/-0 EFPD
UNIT 1
OCONEE NUCLEAR STATION
Figure 3.5.2-1
(1 of 3)



ROD POSITION LIMITS
FOR FOUR PUMP OPERATION
FROM 30 + 10/-0 TO 250 ± 10 EFPD
UNIT 1
OCONEE NUCLEAR STATION
Figure 3.5.2-1
(2 of 3)

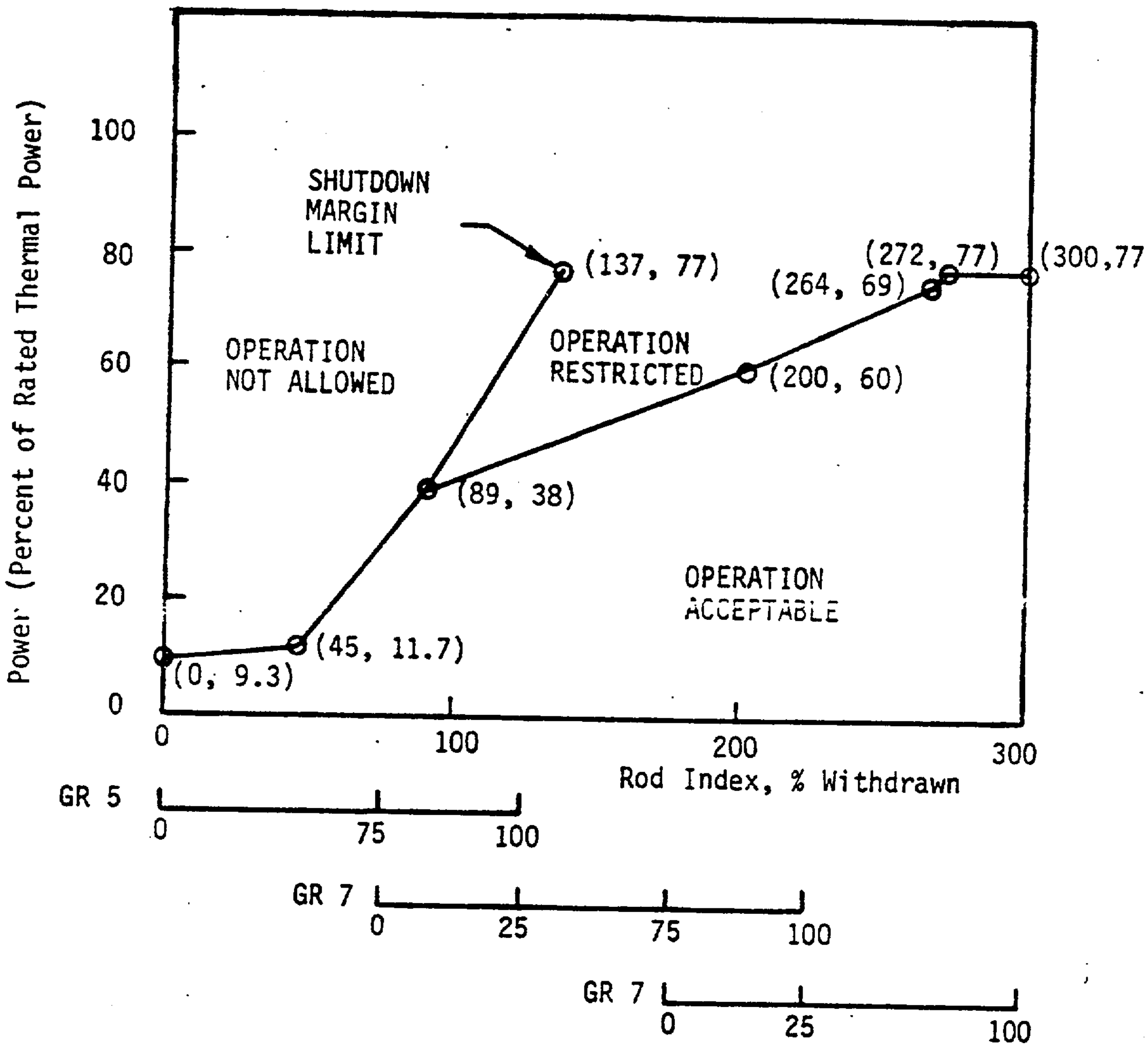


ROD POSITION LIMITS
FOR FOUR PUMP OPERATION
AFTER 250 ± 10 EFPD
UNIT 1

OCONEE NUCLEAR STATION

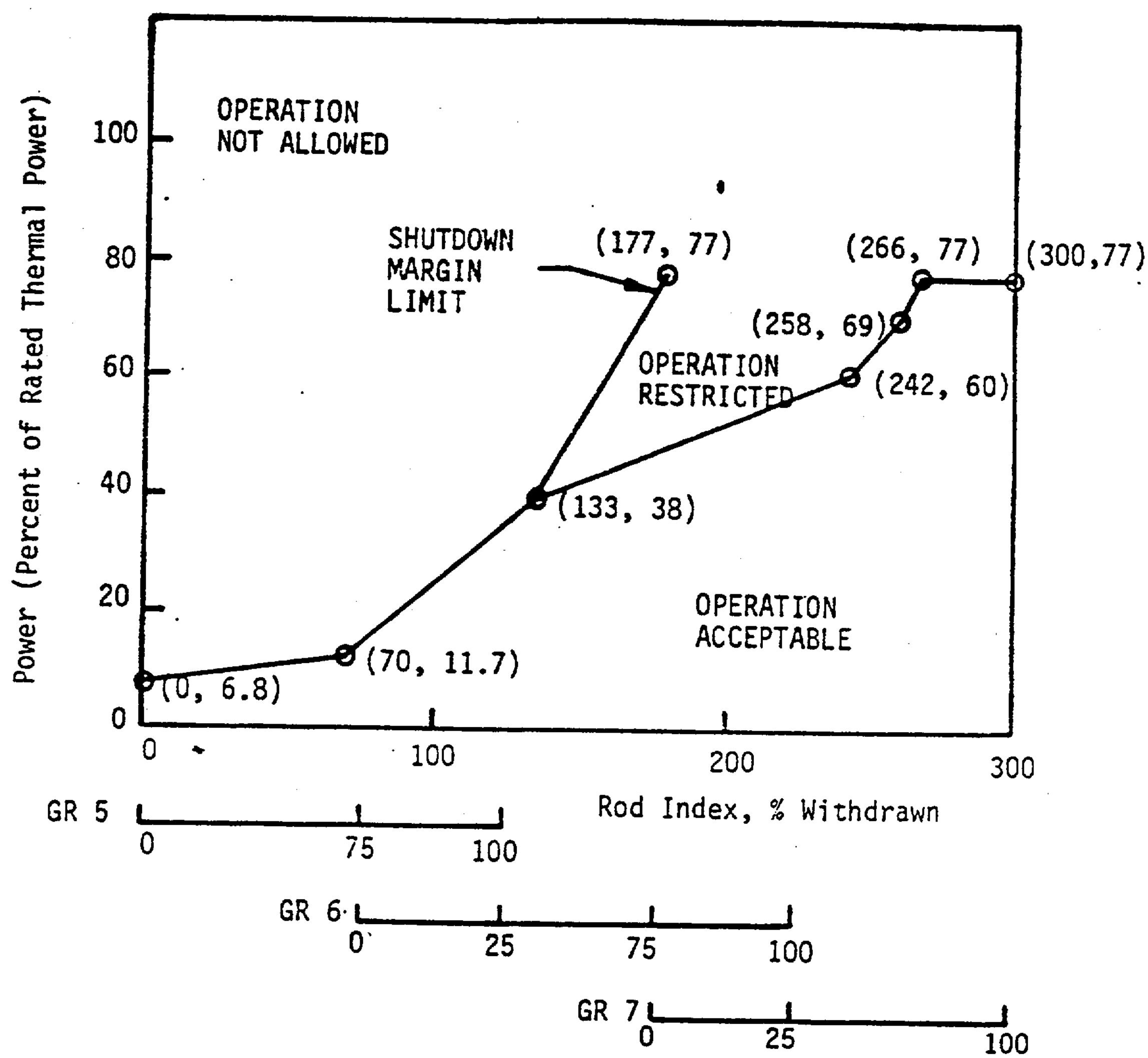
Figure 3.5.2-1

(3 of 3)

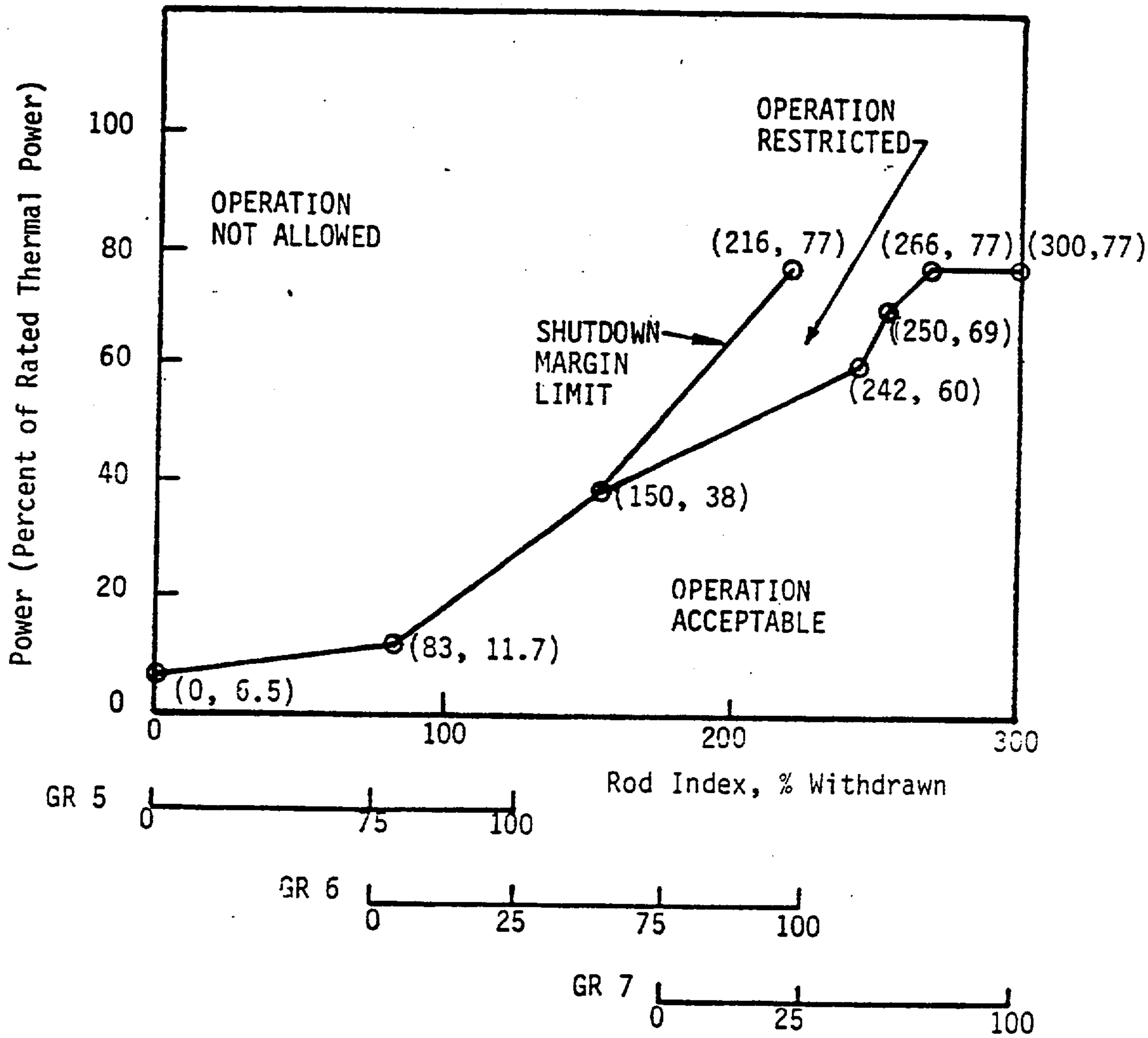


ROD POSITION LIMITS
FOR THREE PUMP OPERATION
FROM 0 TO 30 + 10/-0 EFPD
UNIT 1

OCONEE NUCLEAR STATION
Figure 3.5.2-4
(1 of 3)

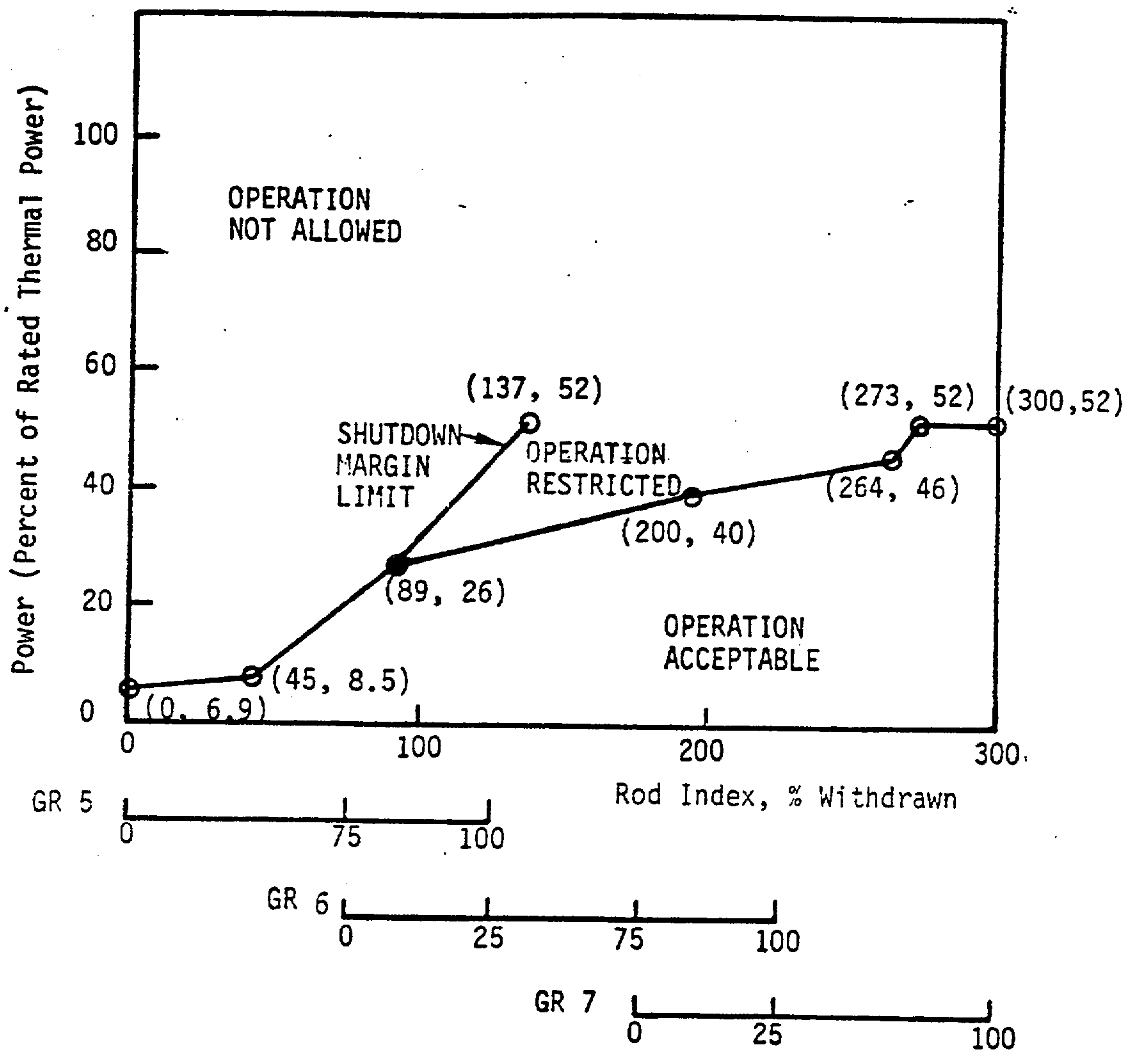


ROD POSITION LIMITS
FOR THREE PUMP OPERATION
FROM 30 + 10/-0 TO 250 ± 10 EFPD
UNIT 1
OCONEE NUCLEAR STATION
Figure 3.5.2-4
(2 of 3)



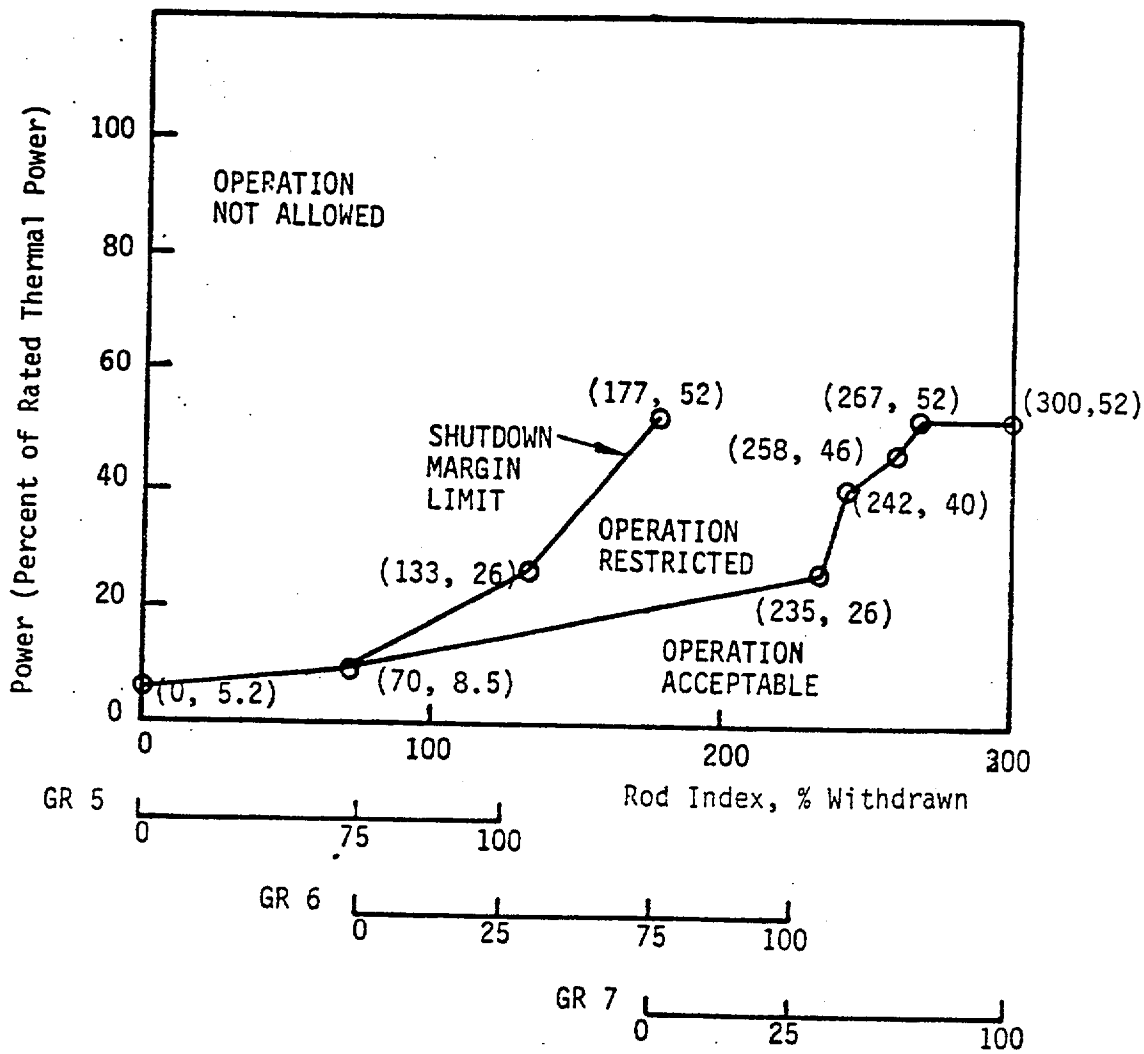
ROD POSITION LIMITS
FOR THREE PUMP OPERATION
AFTER 250 ± 10 EFPD
UNIT 1

OCONEE NUCLEAR STATION
Figure 3.5.2-4
(3 of 3)



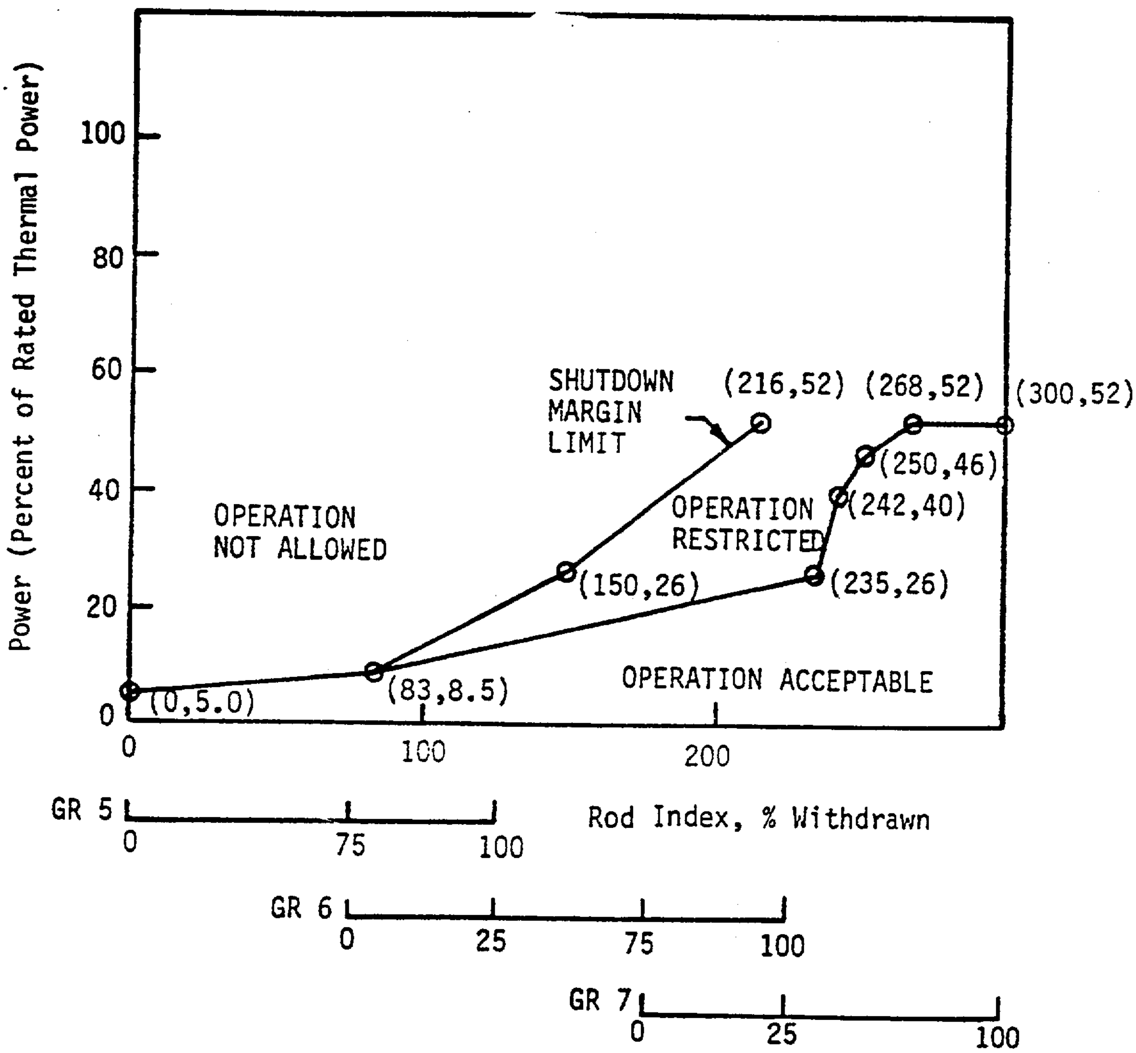
ROD POSITION LIMITS
FOR TWO PUMP OPERATION
FROM 0 TO 30 + 10/-0 EFPD
UNIT 1

OCONEE NUCLEAR STATION
Figure 3.5.2-7
(1 of 3)



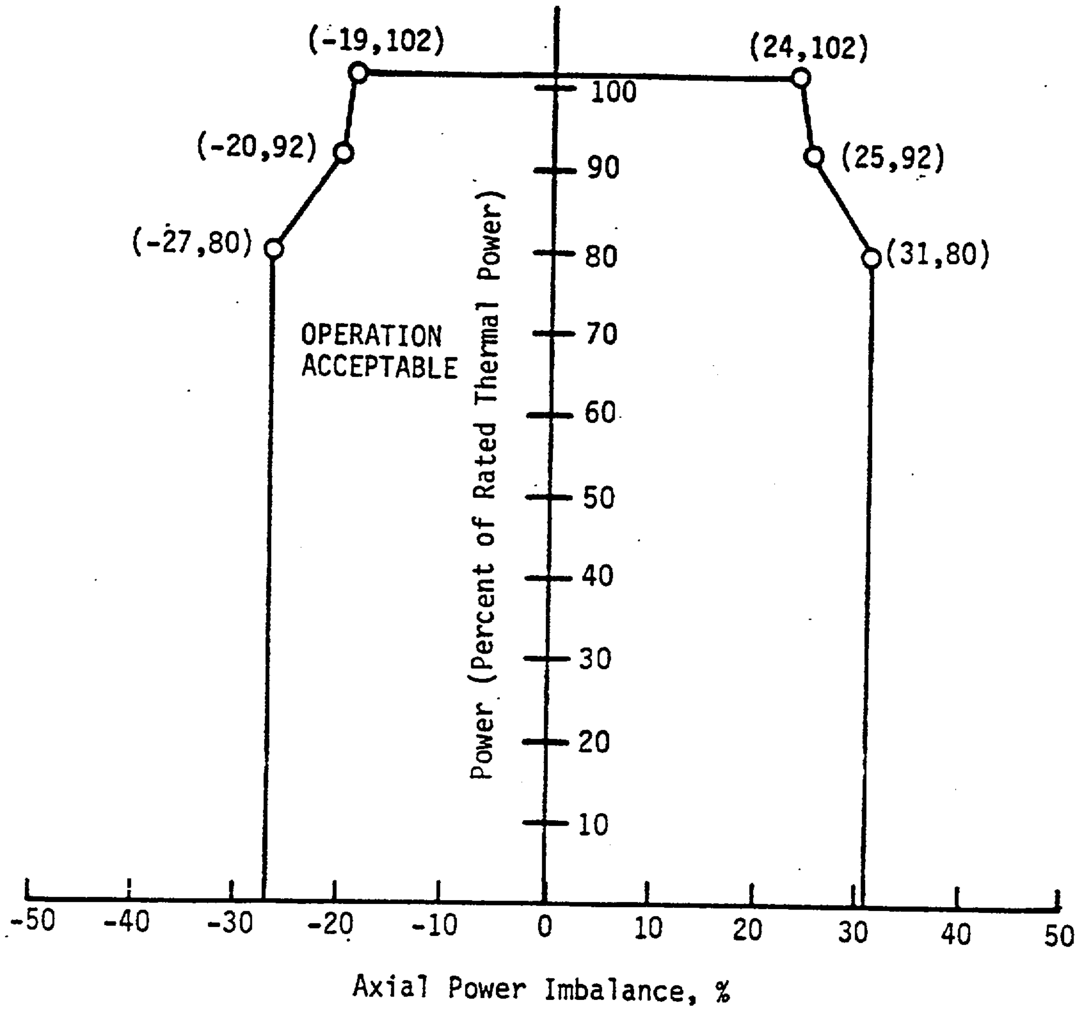
ROD POSITION LIMITS
FOR TWO PUMP OPERATION
FROM 30 + 10/-0 TO 250 ± 10 EFPD
UNIT 1

OCONEE NUCLEAR STATION
Figure 3.5.2-7
(2 of 3)

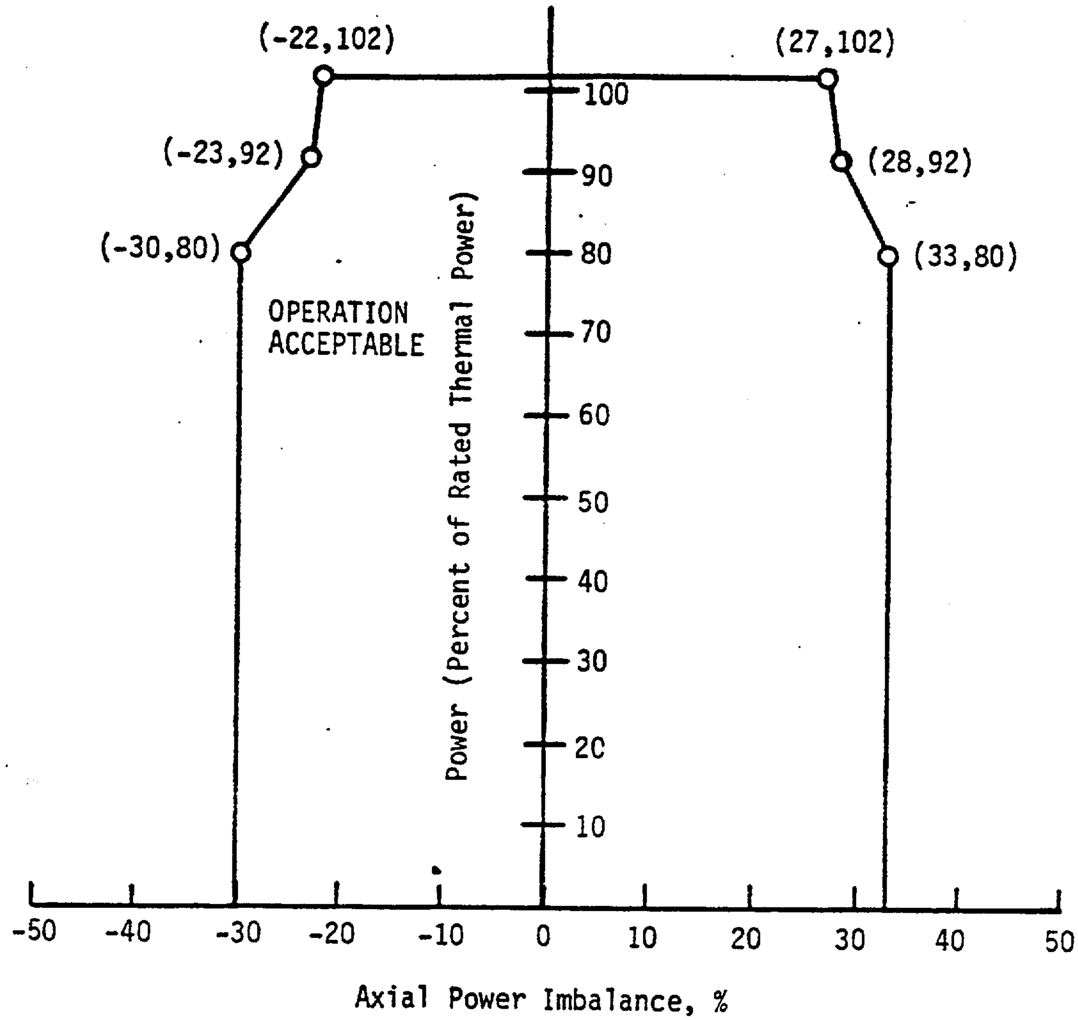


ROD POSITION LIMITS
FOR TWO PUMP OPERATION
AFTER 250 ± 10 EFPD
UNIT 1

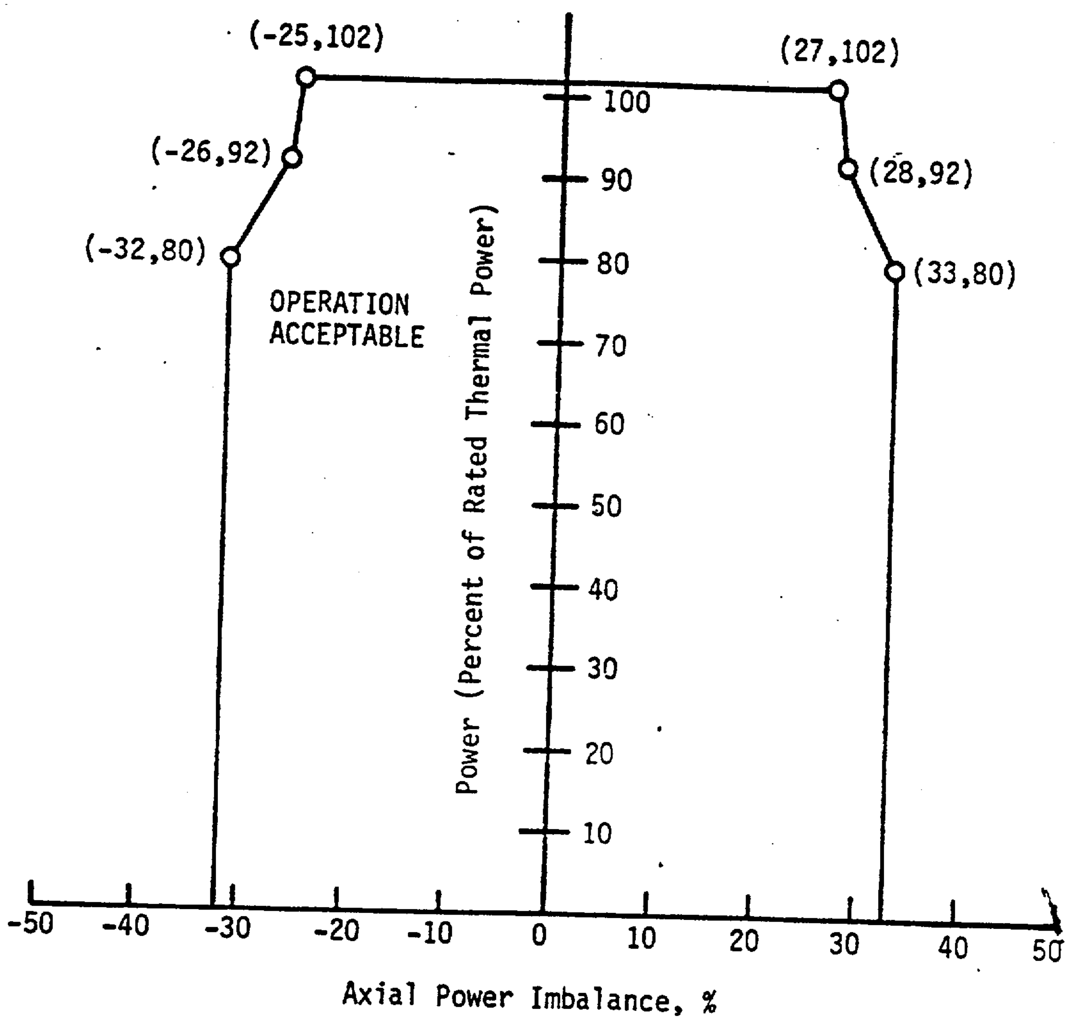
OCONEE NUCLEAR STATION
Figure 3.5.2-7
(3 of 3)



OPERATIONAL POWER
IMBALANCE ENVELOPE
FROM 0 TO 30 + 10/-0 EFPD
UNIT 1
OCONEE NUCLEAR STATION
Figure 3.5.2-10
(1 of 3)



OPERATIONAL POWER
IMBALANCE ENVELOPE
FROM 30 + 10/-0 TO 250 ± 10 EFPD
UNIT 1
OCONEE NUCLEAR STATION
Figure 3.5.2-10
(2 of 3)



OPERATIONAL POWER
IMBALANCE ENVELOPE
AFTER 250 ± 10 EFPD
UNIT 1
OCONEE NUCLEAR STATION
Figure 3.5.2-10
(3 of 3)

Figure 3.5.2-13
(Deleted)

[Note that no rod position limits exist for Unit 1 axial power shaping rods.]

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 132 TO FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 132 TO FACILITY OPERATING LICENSE NO. DPR-47

AMENDMENT NO. 129 TO FACILITY OPERATING LICENSE NO. DPR-55

DUKE POWER COMPANY

OCONEE NUCLEAR STATION, UNITS NOS. 1, 2, AND 3

DOCKETS NOS. 50-269, 50-270 AND 50-287

INTRODUCTION

By letter dated September 11, 1984, as supplemented on October 22, 26, and November 1, 1984 (Ref. 1, 6, 17, and 18), Duke Power Company (the licensee) proposed changes to the Technical Specifications (TSs) of Facility Operating Licenses Nos. DPR-38, DPR-47, and DPR-55 for the Oconee Nuclear Station, Units Nos. 1, 2 and 3. These amendments would consist of changes to the Station's common TSs. Oconee Unit 1 is currently completing a refueling outage and was originally scheduled for plant restart on November 26, 1984 (Ref. 1). References 25 and 26 state that Oconee Unit 1 shutdown earlier than scheduled and startup is scheduled for November 24, 1984.

These amendments would authorize proposed changes to the Oconee Nuclear Station TSs which are required to support the operation of Oconee Unit 1 at full rated power during the upcoming Cycle 9. The proposed amendments would change the following areas:

1. Core Protection Safety Limits (TS 2.1);
2. Protective System Maximum Allowable Setpoints (TS 2.3);
3. Rod Position Limits (TS 3.5.2); and
4. Power Imbalance Limits (TS 3.5.2).

To support the license amendment application, the licensee submitted a Babcock and Wilcox (B&W) report, BAW-1841 (Ref. 2), "Oconee Unit 1, Cycle 9 Reload Report," as an attachment to Reference 1. A summary of the Cycle 9 operating parameters is included in the report, along with safety analyses.

The Cycle 9 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. Cycle 9 is to have a length of approximately 410 effective full power days (EFPD) of operation. As has been the case for Cycle 8, Cycle 9 will be operated in a rods-out, feed-and-bleed mode with core reactivity control supplied mainly by soluble boron in the reactor coolant and supplemented by 61 full length control rod assemblies (CRAs) and 60 burnable poison rod assemblies (BPRAs). In addition, eight axial power shaping rods (APSRs) are provided for additional control of the axial power distribution. The licensed core full power level remains at 2568 MWt.

During the refueling outage, 113 fuel assemblies will be reinserted similar to those previously used and 64 fuel assemblies will be discharged and replaced by new but substantially similar assemblies of the Mark BZ type. The Mark BZ fuel assemblies are the same as previously approved and used assemblies in terms of fuel rods, end grid, end fittings, and guide tubes and differ only slightly from previously approved assemblies in the use of Zircaloy spacer grids rather than Inconel Intermediate spacer grids. The Mark BZ fuel assemblies are discussed further in Section 7.0.

EVALUATION

1.0 Evaluation of the Fuel System Design

1.1 Fuel Assembly Mechanical Design

Cycle 9 will contain 45 Mark B and four Mark BZ fuel assemblies in Batch 9B, 60 Mark B and four Mark GdB assemblies in Batch 10C and 10B, respectively, and 64 Mark BZ assemblies in Batch 11. All of these fuel assemblies are mechanically interchangeable. The Mark BZ design is similar to the Mark B fuel assembly except that the six intermediate Inconel spacer grids have been replaced with Zircaloy grids. Four Mark BZ assemblies have been previously approved as demonstration assemblies in the last two cycles of Unit 1.

For Cycle 9, a significant portion of the core will contain Mark BZ fuel. The design (Ref. 3) of these assemblies has been reviewed and approved by the NRC staff (Ref. 4). However, in our Safety Evaluation Report on Asymmetric LOCA Loads for Oconee Units 1, 2 and 3 (Ref. 5), we identified a measured critical load (P_{crit}) versus maximum spacer grid load and a resulting temperature rise of 12°F in peak cladding temperature (PCT) due to a fully collapsed grid (41% of flow area reduction) in the core peripheral assemblies for standard fuel. At our request, the licensee has submitted a corresponding analysis for an Oconee mixed core and a pure Mark BZ fuel core considering the reduction in strength due to the use of Zircaloy grids (Ref. 6). The PCT increase for Mark BZ fuel was estimated on the basis of calculations performed for Mark B fuel. The Mark B PCT analysis assumed the maximum flow area reduction of 41% along the entire assembly. Since dynamic response analyses showed that the maximum horizontal impact forces and maximum flow area reduction occur on the two mid-height spacer grids for both the Mark B and Mark BZ assemblies and the calculated maximum flow area reduction for Mark BZ fuel was 37%, we conclude that the assumptions used in the Mark B analysis are conservative. Therefore, we believe that these conservatisms as well as the similarities in the grid geometry justify the estimation of the PCT increase for Mark BZ fuel on the basis of calculations performed for Mark B fuel. The PCT increase was evaluated for the racking failure mode only since this case resulted in a larger flow area reduction than the crushing failure mode. Since the maximum horizontal impact forces occur in a peripheral fuel assembly, any racking or crushing failures would also be observed there. For these reasons, the PCT analysis for the Mark B fuel is expected to bound the analysis for the Mark BZ fuel. The Mark BZ assemblies are, therefore, acceptable for use in Cycle 9 and future cycles.

The 62 retainer assemblies used on the two fuel assemblies that contain regenerative neutron source assemblies and on the 60 assemblies that contain BPRAs will undergo a fourth cycle of irradiation during Cycle 9. Based on the results of an examination of retainers which have undergone irradiation during the last three cycles, we conclude that a fourth cycle of irradiation is acceptable.

1.2 Fuel Rod Design

In addition to the Mark B and the Mark BZ fuel assemblies, four assemblies (Batch 10B) will contain fuel pellets containing both urania (UO_2) and gadolinia (Gd_2O_3) as described in Reference 7. These four Mark GdB lead test assemblies (LTAs) are part of a joint Duke Power/Babcock and Wilcox/ Department of Energy program to develop and demonstrate an advanced fuel design incorporating urania-gadolinia for extended burnup in pressurized water reactors. Since the addition of four assemblies containing gadolinia to the Oconee 1 Cycle 9 core does not affect the operating limits in the Technical Specifications nor adversely affect the existing safety analyses, we approve the continued use and irradiation of the LTAs in Cycle 9. However, this should not be construed as an approval of the urania-gadolinia design for full scale applications.

1.2.1 Cladding Collapse

The licensee has stated that the cladding collapse time for the most limiting Cycle 9 assembly (including the gadolinia-bearing LTAs) was conservatively determined to be greater than the maximum projected residence time for any Cycle 9 assembly. The creep collapse analysis used the CROV computer code (Ref. 8). These methods and procedures have been reviewed and approved by the NRC staff. We conclude that cladding collapse has been appropriately considered and will not occur for Cycle 9 operation.

1.2.2 Cladding Stress and Strain

The cladding stress and strain analyses for the Cycle 9 fuel designs, including the gadolinia LTAs and the gray APSRs, are either bounded by conditions previously analyzed for Oconee Unit 1 or were analyzed specifically for Cycle 9 using methods and limits previously reviewed and approved by the NRC staff. We conclude that the analysis of cladding stress and strain has been appropriately considered for Cycle 9 operation.

1.3 Fuel Thermal Design

The thermal behavior of all fuel in the Cycle 9 core, with the exception of the gadolinia-bearing LTAs, is virtually identical. The thermal analysis for reinserted Batches 9B, 10B, 10C and feed Batch 11 fuel was performed with the approved TACO2 (Ref. 9) code using the approved methodology described in Reference 10. The centerline fuel melt (CFM) limits of 20.5 kw/ft for UO_2 fuel and 17.6 kw/ft for $UO_2Gd_2O_3$ fuel were predicted using TACO2. These latter Mark GdB LTAs are loaded in the core in such a manner so as to ensure that there is sufficient margin to offset any negative impact on the loss of coolant accident (LOCA) kw/ft limits discussed in Section 4.0 of this Safety Evaluation (SE). In addition, these LTAs will be limited to a design peak of 1.67 to ensure that they are not thermally limiting. These CFM limiting values are incorporated into the TSs, and we find them acceptable.

Standard Review Plan (SRP) 4.2, Section II.S.1(f), contains the requirement that the fuel rod internal gas pressure should remain below normal system pressure during normal operation unless otherwise justified. Based on TACO2 analyses, the licensee has stated that the internal pressure in the highest burnup rod will not reach the nominal Reactor Coolant System (RCS) pressure of 2200 psia. We find this acceptable and conclude that the fuel rod internal pressure limits have been adequately considered for Cycle 9 operation.

2.0 Evaluation of Nuclear Design

The nuclear design parameters characterizing the Oconee Unit 1 Cycle 9 core have been computed by methods previously used and approved for B&W reactors. Comparisons are made between the physics parameters for Cycle 8 and 9. Changes in the radial flux and burnup distributions as well as changes in rod groupings and the gray APSRs (described below) between cycles account for the differences in control rod worths, including ejected and stuck rod worths. All safety criteria are still met.

The highly-absorbing (black) APSRs utilized during the previous cycles have been replaced by less absorbing (gray) APSRs in Cycle 9. These gray APSRs have a greater absorber length than the previously employed ones and utilize an Inconel absorber instead of the previous silver-indium-cadmium (Ag-In-Cd) alloy. Since gray APSRs are being utilized, there are now eight control rods in group 7 and twelve in group 5 to reduce the negative offset response to the group 7 rod movement. Previous cycles utilizing black APSRs contained twelve rods in group 7 and eight in group 5. Calculations with approved B&W models were used to verify that these gray APSRs provide adequate axial power distribution control and will not adversely affect Cycle 9 operation. Revisions to the TSs to account for these changes for Cycle 9 operation were made in accordance with methods and procedures found acceptable in connection with previous reloads. The replacement of the black APSRs by gray APSRs and the changes in control rod groupings for Cycle 9 are, therefore, acceptable.

Shutdown margin calculations for Cycle 9 include the effects of poison material depletion, a 10% calculational uncertainty, and flux redistribution as well as a maximum worth stuck rod. Beginning and end-of-cycle shutdown margins show adequate reactivity worth exists above the total required worth during the cycle. The required shutdown margin is 1.00% Δ k/k, the shutdown margins at the beginning and end-of-cycle are 3.15%Δk/k and 2.35% Δ k/k, respectively.

The four Mark GdB LTAs which were inserted at the beginning of the last cycle and had an initial enrichment of 4.0 weight percent U-235 will undergo a second cycle of irradiation during Cycle 9. Based on a reduction in U-235 enrichment due to the previous cycle burnup of over 15,000 MWD/MTU for these four LTAs, B&W has calculated the Mark GdB assemblies average enrichment at beginning of Cycle 9 to be 2.48 weight percent U-235 with the highest enriched pin being approximately 2.57 weight percent U-235. The effect of these fuel assemblies on the nuclear design continues to meet all criteria including those applicable to radial power peaking, ejected rod worths, moderator temperature coefficient (MTC), and shutdown margin.

Based on our review, we conclude that approved methods have been used, that the nuclear design parameters meet applicable criteria and that the nuclear design of Oconee 1 Cycle 9 is acceptable.

3.0 Evaluation of Thermal-Hydraulic Design

Cycle 9 fuel includes 64 fresh Mark BZ fuel assemblies, four irradiated Mark GdB LTAs, and four irradiated Mark BZ demonstration assemblies, all of which incorporate Mark BZ spacer grids. The effect of the higher pressure drop caused by these grids, and by the BPRA retainers, is a slightly lower flow in the Mark BZ assemblies. Therefore, the departure from nucleate boiling (DNB) margin for these assemblies is reduced. In order to preserve the DNB margin, the radial - local design power peaking has been reduced to 1.67 for the Mark BZ and Mark GdB assemblies for Cycle 9 only. All other Mark B assemblies continue to have a 1.71 design radial times local peak. The maximum expected peaking during Cycle 9 is 1.416. We concur that the reduction in allowable power peaking limits compensates for the reduced thermal margin so that required safety margins are maintained.

The safety limits presented for Cycle 9 have been generated with the BAW-2 (Ref. 11) used in the previous cycle, and BWC (Ref. 12) critical heat flux (CHF) correlations. The BWC correlation was used to predict the DNB behavior of the Mark BZ fuel assemblies with a departure from nucleate boiling ratio (DNBR) of 1.18 corresponding to a 95 percent probability at a 95 percent confidence level that DNB will not occur. The subchannel analysis for Cycle 9 used the CHATA and TEMP core thermal-hydraulic codes. B&W has previously provided data comparisons to the NRC which the staff concluded justified the use of the BWC correlation with these codes.

A B&W topical report (Ref. 13) discussing the mechanisms and resulting effects of bowing in B&W fuel has been reviewed and approved (Ref. 14). The report concludes that the DNBR penalty due to rod bow need not be imposed for those assemblies with significant bow because the power production capability of the fuel decreases sufficiently with irradiation to offset the effects of bowing. Post irradiation measurements on Mark BZ lead demonstration assemblies verified that the methodology of Reference 13 conservatively predicts the rod bow in Mark BZ assemblies also. Therefore, we conclude that no rod bow penalty need be considered for Cycle 9 operation.

4.0 Safety Analyses

The important kinetics parameters for Cycle 9 have been compared to the values used in the Final Safety Analysis Report (FSAR) and/or the densification report. The licensee has shown that the Cycle 9 values are bounded by those previously used. The licensee has also determined that the initial conditions of the transients in Cycle 9 are bounded by either the FSAR, the fuel densification report, or previous reload analyses. These analyses have been previously accepted by the NRC.

B&W has performed a generic LOCA analysis for the B&W 177-FA, lowered-loop nuclear steam supply system (NSSS) using the final acceptance criteria (FAC) Emergency Core Cooling System (ECCS)evaluation model (Ref. 15). The combination of average fuel temperature as a function of linear heat rate (LHR) and the lifetime pin pressure data used is conservative relative to those calculated for this cycle. Three sets of bounding values for allowable LOCA peak LHRs for Cycle 9 are given as a function of core height. These limits apply during the periods 0 to 30 EFPD, 30 to 250 EFPD, and for the balance of the cycle. These results are based upon a bounding analytical assessment of NUREG-0630 on LOCA and operating LHR limits performed by B&W (Ref. 16). The B&W analyses have been approved by the NRC staff and the LHR limits are satisfactorily incorporated into the TSs for Cycle 9 through the operating limits on rod index and axial power imbalance.

5.0 Technical Specification Modifications

Oconee Unit 1 Cycle 9 TSs have been modified to account for changes in power peaking and control rod worths, the replacement of black APSRs by gray APSRs, use of the BWC CHF correlation, and elimination of the DNBR rod bow penalty factor. We have reviewed the proposed specification revisions for Cycle 9. These changes concern the (1) Core Protection Safety Limits of Specification 2.1, (2) Protective System Maximum Allowable Setpoints of Specification 2.3, and (3) Rod Position Limits and Operational Power Imbalance Envelope of Specification 3.5.2. On the basis that approved methods were used to obtain these limits, we find these TS modifications acceptable.

In Reference 18, the licensee provided clarification to Reference 1, which transmitted the TS amendment request to support the operation of Oconee Unit 1 during Cycle 9. Specifically, Reference 18 included a minor revision to the text comprising "Bases-Unit 1" of TS 2.1 and changed the words "22 percent" to "the CHF correlation quality limit". This revision makes the TSs consistent with transition cores containing both Mark B and Mark BZ type fuel assemblies. As discussed in Reference 1, the BAW-2 and the BWC CHF correlations are applicable to the Mark B and the Mark BZ type fuel assemblies, respectively. The present wording is applicable only to a full Mark B core using the BAW-2 correlation. Therefore, the change in wording reflects the use of both the BAW-2 (for Mark B) and the BWC (for Mark BZ) correlations in the transition Cycle 9 core, and thus merely provides the required generality for the description. The licensee states that the omission of the change in the Reference 1 submittal constitutes an administrative oversight and this revision possesses no significance with respect to the safety analyses included in the original submittal (Ref. 1).

6.0 Evaluation Findings

We have reviewed the fuels, physics, thermal-hydraulic, and accident information presented in the Oconee Unit 1 Cycle 9 reload report. We find the proposed reload and the associated modified TSs acceptable.

7.0 Mark BZ Fuel Assembly Design Review - Introduction

By letter dated October 7, 1983 (Ref. 19) as supplemented on October 22, 1984 (Ref. 6), the licensee submitted topical report BAW-1781P, "Rancho Seco Cycle 7 Reload Report, Volume 1, Mark BZ Assembly Design Report", for NRC staff review. The same report was submitted by Sacramento Municipal Utility District (SMUD, Ref. 20). The Babcock & Wilcox Mark BZ fuel design is based on the approved Mark B fuel assembly design. The Mark BZ fuel assembly has an array of 15x15 fuel rods with six Zircaloy intermediate spacer grids, replacing the six Inconel Intermediate spacer grids of the Mark B fuel assembly. The other components such as fuel rods, end grids, end fittings, and guide tubes are the same for both designs. Since the Mark B fuel design has been approved, this Safety Evaluation will address only the adequacy of the Zircaloy grids of the Mark BZ fuel design. Therefore, the NRC staff has reviewed the Rancho Seco Topical and has found acceptable the Mark BZ fuel design for all lowered-loop B&W designed 177 fuel assembly plants (Ref. 4).

7.1 Fuel Mechanical Design

In order to maintain similar mechanical strength, the Zircaloy grid is made wider and thicker than the Inconel grid. However, the total assembly weight is about the same for both designs because fuel and cladding contribute much more in weight. Although most aspects of the Mark BZ fuel performance will not differ from those of the Mark B fuel, we raised several questions about possible deviations from previous Mark B fuel analysis. In a letter dated July 13, 1984 (Ref. 21), SMUD provided responses to address our questions. We will discuss these differences in the following sections.

7.2 Design Bases

In response to our question concerning design bases, the licensee states that the design bases for Mark BZ fuel are identical to those approved for Mark B fuel. In addition, the design evaluations to verify the adequacy of the Mark BZ fuel design were performed according to SRP Section 4.2.II.C. As far as mechanical design is concerned, our evaluation has determined that there is little difference between the Mark B and Mark BZ fuel designs with respect to the criteria and limits of this section of the Standard Review Plan. We therefore conclude that design bases of the Mark BZ fuel assembly are acceptable.

7.3 Holddown Spring

In the past, the B&W Mark B fuel has experienced significant holddown spring failures. We questioned the adequacy of the Mark BZ fuel design regarding the holddown spring failure. SMUD stated that Mark BZ fuel has incorporated several major changes to improve holddown spring performance. They include increasing wire diameter, changing to a more fatigue-resisting material, and tighter fabrication control. SMUD concluded that these procedures will reduce the possibility of fatigue-induced failures. Inasmuch as SMUD has provided assurance that the holddown spring is redesigned to alleviate the past problem, we conclude that holddown spring failure is adequately addressed in the Mark BZ fuel design, subject to acceptable results from the holddown spring surveillance program for Mark B fuel described in Reference 24.

7.4 Assembly Liftoff

The function of the holddown spring is to maintain the fuel assemblies seated in the lower core plate during the worst-case hydraulic load. The new holddown spring design raised a concern as to whether holddown capability remains intact. SMUD's analysis showed that there is enough positive holddown margins for Mark BZ fuel to prevent assembly liftoff in the most limiting condition of maximum hydraulic lift force. We thus conclude that the Mark BZ fuel assembly will maintain its holddown capability during the worst-case hydraulic load.

7.5 Seismic and LOCA Loads

Appendix A to SRP 4.2 describes a fuel assembly structural response analysis under combined seismic and LOCA loads including asymmetric blowdown loads. Although the Mark BZ fuel assembly has mechanical strength similar to the Mark B fuel assembly, we will require a plant-specific analysis of combined seismic-and-LOCA loads for mixed core and a pure Mark BZ fueled core. It is permissible to perform a bounding analysis or to make a comparison with the previous approved Mark B fuel results using approved methods to demonstrate conformance to Appendix A of Standard Review Plan Section 4.2.

7.6 Post-Irradiation Surveillance

A lead test assembly of Mark BZ fuel was irradiated earlier in Oconee 2 and was examined non-destructively after discharge. The result showed that the fuel performed adequately.

In response to our questions, SMUD indicated that B&W has a surveillance plan of visual examinations on selected assemblies for a total of nine demonstration assemblies with Zircaloy spacer grids, which are currently being irradiated in Oconee 1. Visual examination includes water channel, holddown spring, and length measurement at the end of each cycle. Considering that Mark BZ fuel has only one significant change (spacer grids), we conclude that the post-irradiation surveillance, though minimal, is acceptable per SRP 4.2 guidelines.

7.7 Thermal Hydraulic Design

The acceptance criterion specified in Section 4.4 of the Standard Review Plan for the thermal hydraulic design requires that there is at least a 95 percent probability at a 95 percent confidence level that the hot fuel rod in the core does not experience a DNB during normal operation or anticipated operational occurrences. The safety analysis of the Mark BZ fuel design must demonstrate that this criterion is met.

7.7.1 Hydraulic Characteristics

Since the Zircaloy spacer grid in the Mark BZ fuel design has a grid height and grid strip thickness larger than the Inconel grid used in the Mark B fuel, and since the outer grid strips of the Zircaloy grid lead-in taps, the Mark BZ fuel has higher hydraulic resistance than than the Mark B fuel. B&W has performed flow tests of a full length Mark BZ prototype fuel assembly using the Control Rod Drive Line facility. A fuel bundle flow distribution test was also performed using laser Doppler Velocimeter. These tests provided data for development of grid form loss coefficients on both an assembly and a subchannel basis. The pressure drop across the Mark BZ assembly is found to increase by less than 3 percent over the Mark B assembly and therefore, its impact on the reactor system flow rate is insignificant.

7.7.2 Thermal Margin Analysis

In the evaluation of the effect of Mark BZ fuel design on thermal margin, B&W performed analyses with regard to the variable pressure-temperature (P-T) limit envelope and the maximum allowable peaking (MAP) limits which are used as bases for the reactor protection system setpoints for the low DNBR trip. The steady state analysis for the Mark BZ fuel assembly was performed with a two-pass method where the closed channel thermal hydraulic code, CHATA, was used for the core-wide analysis, and the subchannel TEMP code was used for the hot assembly/hot channel analysis. The two-pass method and the two thermal hydraulic codes have been approved for licensing analysis and have been used extensively in many B&W reactors. In contrast to the B&W-2 correlation used for the Mark B fuel with a DNBR limit of 1.3, critical heat flux was calculated using the BWC correlation which has been approved (Ref. 22) for the Mark BZ fuel with a DNBR limit of 1.18. The results of analyses show that the Mark BZ fuel has less restrictive P-T limits than Mark B fuel and that the MAP limits for Mark BZ and Mark B have only a small difference.

An analysis was also performed to determine the effect of Mark BZ fuel on thermal margin for reactor transients. Since the partial loss of coolant flow transient is the limiting anticipated operational occurrence event, this event was analyzed for Mark BZ fuel. The analysis was performed by evaluating the flux/flow setpoint which is designed for the DNBR protection for partial pump flow operation. The flux/flow limit is determined from a thermal hydraulic analysis for the pump coastdown transient using the approved RADAR code to ensure that the hot channel DNBR will not exceed the minimum DNBR limit. A comparison was made of the flux/flow limits for a full core Mark BZ and a full core Mark B. The results shown in Figure 5-2 of topical report BAW-1781P show almost identical flux/flow limits for both Mark B and Mark BZ fuel. Since the flux/flow trip setpoint is to ensure that the minimum DNBR limit will not be violated during partial loss of flow transient if a reactor trip is initiated as soon as the ratio of reactor power to RC flow reaches the flux/flow limit, and since the flux/flow limits for Mark BZ fuel are very close to those for the Mark B fuel, we conclude that a full core of Mark BZ fuel has no significant impact on thermal margin.

7.7.3 Transitional Mixed Core

Incompatibility in the hydraulic characteristics has an additional effect on thermal margin during transitional mixed core cycles when both Mark BZ and Mark B fuel assemblies co-exist in the core. Since Mark BZ fuel has higher hydraulic resistance, the presence of Mark BZ fuel tends to force more flow into the Mark B fuel. Therefore, if a Mark B fuel assembly is the limiting assembly, the hot channel will receive more coolant and yield better DNB performance compared to a whole core of Mark B assemblies. As a result, the existing safety analysis for Mark B fuel is bounding and applicable to a transitional mixed core. For the cases where a Mark BZ assembly is limiting, a transitional mixed core will result in worse DNB performance than a whole core of Mark BZ assemblies. In response (Ref. 21) to an NRC staff question, SMUD provided a description on how the hydraulic incompatibility between the Mark BZ and Mark B fuel is accounted for. The limiting assembly is assumed to be a Mark BZ fuel assembly. In the core-wide analysis, the core is modeled with 177 parallel channels and the number of the Mark BZ assemblies is conservatively chosen to be less than or equal to the actual number of Mark BZ assemblies in the cycle being analyzed. This approach is conservative because more flow would be diverted to the Mark B assemblies having less hydraulic resistance and, therefore, ensures that the lowest flow rate is used in the highest powered fuel assembly. This conservative assembly flow rate is then input into the subchannel TEMP code for the hot assembly/hot channel DNBR calculation. For transient analysis using the RADAR code, the hot channel flow rate and DNBR are benchmarked against the steady state TEMP results. Therefore, the approach is also conservative. The fact that the Mark BZ assembly has less flow in a mixed core will result in lower maximum allowable peaking and a lower enthalpy rise factor $F_{\Delta H}$ in order to maintain the same DNBR limit compared to a whole core of Mark BZ fuel. As indicated in the report, the amount of peaking reduction necessary during the transitional mixed core will be identified in the cycle specific section of a licensee's reload report, and the $F_{\Delta H}$ reduction will be determined by an analysis of the flux/flow setpoint. We, therefore, conclude that the transitional mixed core has been adequately addressed.

7.7.4 ECCS Analysis

The ECCS analysis has shown that the increase in the core pressure drop due to the higher hydraulic resistance of the Mark BZ fuel design has no adverse effect on the core thermal hydraulic conditions and thus the LOCA limits. However, since the BWC correlation is used for the Mark BZ fuel CHF calculation, the analysis shows that use of BWC results in the prediction of earlier inception of DNB at higher (above 6 feet) core elevation relative to the generic 177 FA lowered loop ECCS evaluation using the B&W-2 correlation. This earlier inception of DNB results in a reduction of less than 1 kw/ft in the LOCA limit at the 6-foot core elevation, but does not affect the plant operational limits since the LOCA limits at lower elevations are not affected, and previous analysis has shown that the LOCA limits at the 2- and 4-foot elevation are the controlling parameters.

The use of Zircaloy spacer grid for the Mark BZ also increases the amount of Zircaloy material for metal-water reaction. However, by conservatively assuming the Zircaloy grid temperature equal to that of the hottest point of the cladding, the maximum local oxidation of the Zircaloy grid is 6.2%, the same as the Zircaloy cladding at the 10-foot elevation. This local oxidation is below the limit of 17% specified in 10 CFR 50.46. In response (Ref. 4) to an NRC staff question, B&W also performed an ECCS analysis for the transitional mixed core at the 2- and 6-foot elevations to determine the effects on peak cladding temperature and LOCA limit. In the analysis, Mark BZ fuel assemblies were assumed in the hot channel and surrounding bundle locations and the Mark B fuel in all remaining locations. The results of analyses are compared to the results of a full core of Mark BZ fuel and show negligible impact on the peak cladding temperature and LOCA limit.

7.7.5 Rod Bowing

Although the licensee used a Mark B fuel rod bowing correlation from the approved report BAW-10147, there is only one datum point of Mark BZ fuel presented in the data base of Mark B fuel. We questioned whether one datum point was sufficient to represent Mark BZ fuel bowing magnitude. In response to our question, SMUD stated that the change from the Inconel to Zircaloy grids may actually improve the rod bowing performance because Zircaloy grids reduce axial compression on Mark BZ fuel rods. In addition, two more data points for the Mark BZ fuel bundle were added to the overall analysis. These two points are also below the predicted value of rod bow; thus the rod bow correlation appears to conservatively predict Mark BZ fuel behavior using the current rod bowing correlation of Mark B fuel. This is consistent with observations of other PWR fuel assembly designs with Zircaloy grids and fuel rods. We therefore conclude that the Mark BZ fuel rod bowing analysis is adequate and the effect of rod bow on DNBR as discussed in BAW-10147 is applicable to the Mark BZ fuel.

7.8 Summary and Conclusion

We have reviewed the topical report B&W-1781P and conclude that the Mark BZ fuel is acceptable for reload application and the report is referencable for all lowered-loop B&W designed 177 fuel assembly plants. A licensee referencing the Mark BZ fuel design is required to submit a plant-specific analysis of combined seismic and LOCA loads according to Appendix A to SRP 4.2. The thermal margin reduction, i.e. the reduction of the maximum allowable peaking and F_H, during transitional mixed core having both the Mark B and Mark BZ fuel assemblies must be addressed in the reload licensing reports for the reload cycles having a mixed core. The NRC staff has reviewed Duke Power Company's submittals (Refs 19 and 6) and the topical report, "Rancho Seco Cycle 7 Reload Report, Volume 1, Mark BZ Assembly Design Report" (Ref. 20) and has concluded that the Mark BZ fuel design is acceptable for the Oconee Unit 1 Cycle 9 and for all lowered-loop B&W designed 177 fuel assembly plants.

EXIGENT CIRCUMSTANCES

The following reasons describe the exigent circumstances:

1. In Ref. 26, the licensee states that originally, the refueling outage was scheduled to begin on October 12, 1984. However, the October 8th date was referred to within the September 11th letter since this was the date that the nominal cycle length would fall based on a 100% capacity factor. The decision to begin the Oconee 1 refueling outage on October 5th was based on the following considerations:

   a. In order to assure the maximum electrical output from the Oconee Nuclear Station during the colder winter months, an earlier shutdown date was desirable.

   b. The planned outage duration for Oconee 1 was 49 days. The McGuire Nuclear Station Unit 1 was scheduled to begin a maintenance outage on November 24, 1984. In order to avoid having these two outages overlap, an October 5th shutdown date for Oconee 1 was chosen.

The Commission has determined that exigent circumstances exist in that swift action is necessary to avoid a delay in startup not related to safety and finds that for the reasons stated above, these circumstances caused the outage to start earlier than scheduled.

2. NRC regulation 10 CFR 50.91 describes the procedures that will be followed on applications received after May 6, 1983, requesting license amendment. These procedures require that, in addition to other requirements, a 30-day comment period will be provided to allow for public comment on the Commission's proposed no significant hazards consideration determination. The notice of such determination related to these amendments was published in the Federal Register on October 24, 1984, and, therefore, the 30-day comment period should have expired on November 23, 1984. The expiration date, however, given in the Federal Register was November 26, 1984 at 49 FR 42814.

In connection with requests indicating an exigency, the Commission expects its licensees to apply for license amendments in a timely fashion. However, with this consideration in mind, it has been determined that an extraordinary circumstance has arisen where the licensee and the Commission must act quickly, and the licensee has made a good effort to make a timely application.

FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards considerations if operation of the facility in accordance with the amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

The information in this SE provides the basis for evaluating these license amendments against these criteria. The request for amendment changes the TSs to reflect new operating limits based on the fuel to be inserted into the core. These parameters are based on the new physics of the core and fall within the acceptance criteria. Since the requested change does not affect the original design basis, plant operating conditions, the physical status of the plants, and dose consequences of potential accidents, we conclude that:

(1) Operation of the facilities in accordance with the amendments would not significantly increase the probability or consequences of an accident previously evaluated.

(2) Operation of the facilities in accordance with the amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Operation of the facilities in accordance with the amendments would not involve a significant reduction in a margin of safety.

Accordingly, we conclude that the amendments to Facility Operating Licenses DPR-38, DPR-47, and DPR-55 to support operation of Oconee Unit 1 at full rated power during the upcoming Cycle 9, involve no significant hazards considerations.

STATE CONSULTATION

In accordance with the Commission's regulations, consultation was held with the State of South Carolina by telephone. The State expressed no concern either from the standpoint of safety or of our no significant hazards consideration determination.

ENVIRONMENTAL CONSIDERATION

These amendments involve a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has made a final no significant hazards consideration finding with respect to these amendments. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: November 23, 1984

Principal Contributors: L. Kopp, Y. Hsii, S. Wu

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