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

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September 29, 1982

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

Posted
Amdt. 113
to DPR-47

Mr. Hal B. Tucker, Vice President
Nuclear Production Department
Duke Power Company
P. O. Box 33189
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Tucker:

The Commission has issued the enclosed Amendments Nos. 113 , 113 , and 110 to 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 May 3, 1982, as revised in its entirety by letter dated August 11, 1982, and supplemented by letter dated August 16, 1982.

These amendments revise the TSs to allow full power operation of Oconee Unit 3 during fuel Cycle 7.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Philip C. Wagner

Philip C. Wagner, Project Manager
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 113 to DPR-38
2. Amendment No. 113 to DPR-47
3. Amendment No. 110 to DPR-55
4. Safety Evaluation
5. Notice

cc w/enclosures: See next page

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

Oconee County Library
501 West Southbroad Street
Walhalla, South Carolina 29691

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, Suite 3100
Atlanta, Georgia 30303

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

William T. Orders
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.
DeBevoise & Liberman
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. 113
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 May 3, 1982, as revised August 11, 1982, and supplemented August 16, 1982, 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 Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 113 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: September 29, 1982

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. 113
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 May 3, 1982, as revised August 11, 1982, and supplemented August 16, 1982, 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. 113are 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: September 29, 1982

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. 110
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 May 3, 1982, as revised August 11, 1982, and supplemented August 16, 1982, 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. 110are 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, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 29, 1982

ATTACHMENTS TO LICENSE AMENDMENTS

AMENDMENT NO.113 TO DPR-38

AMENDMENT NO.113 TO DPR-47

AMENDMENT NO.110 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
2.1-2	2.1-2
2.1-3b	2.1-3b
2.1-3d	2.1-3d
2.1-6	2.1-6
2.1-9	2.1-9
2.1-12	2.1-12
2.3-10	2.3-10
3.2-2	3.2-2
3.5-9	3.5-9
3.5-17	3.5-17
3.5-17a	3.5-17a
3.5-17b	3.5-17b
3.5-20	3.5-20
3.5-20a	3.5-20a
3.5-20b	3.5-20b
--	3.5-20c
--	3.5-20d
--	3.5-20e
3.5-23	3.5-23
3.5-23a	3.5-23a
3.5-23b	3.5-23b
3.5-26	3.5-26
3.5-26a	3.5-26a
3.5-26b	3.5-26b

can be related to DNB through the use of the BAW-2 correlation (1). The BAW-2 correlation has 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. A DNBR of 1.30 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 a minimum DNBR of 1.30 is predicted for the maximum possible thermal power (112 percent) when four reactor coolant pumps are operating (minimum reactor coolant flow is 106.5 percent of 131.3 x 10^6 lbs/hr). This curve is based on the combination of nuclear power peaking factors, with potential effects of fuel densification and rod bowing, which result in a more conservative DNBR than any other shape that exists during normal operation.

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 1.30 DNBR limit produced by the combination of the radial peak, axial peak and position of the axial peak that yields no less than a 1.30 DNBR.

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limit is 20.05 kw/ft 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.

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.

The magnitude of the rod bow penalty applied to each fuel cycle is equal to or greater than the necessary burnup dependent DNBR rod bow penalty for the applicable cycle minus a credit of 1% for the flow area reduction factor used in the hot channel analysis. All plant operating limits are based on a minimum DNBR criteria of 1.30 plus the amount necessary to offset the reduction in DNBR due to fuel rod bow.

1. The 1.30 DNBR limit produced by the combination of the radial peak, axial peak and position of the axial peak that yields no less than a 1.30 DNBR.

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limit is 20.15 kw/ft for Unit 2.

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 2.1-3B correspond to the expected minimum flow rates with four pumps, three pumps, and one pump in each loop, respectively.

The curve of Figure 2.1-1B is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in Figure 2.1-3B.

The magnitude of the rod bow penalty applied to each fuel cycle is equal to or greater than the necessary burnup dependent DNBR rod bow penalty for the applicable cycle minus a credit of 1% for the flow area reduction factor used in the hot channel analysis. All plant operating limits are based on a minimum DNBR criteria of 1.30 plus the amount necessary to offset the reduction in DNBR due to fuel rod bow. (3)

The maximum thermal power for three-pump operation is 90.606 percent due to a power level trip produced by the flux-flow ratio 74.7 percent flow x 1.08 = 80.68 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 2.1-3B, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30 or a local quality at the point of minimum DNBR less than 22 percent for that particular reactor coolant pump situation. The curve of Figure 2.1-1B is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in Figure 2.1-3B.

<u>References</u>

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

(2) Oconee 2, Cycle 4 - Reload Report, BAW-1491, August 1978.

(3) Oconee 2, Cycle 6 - Reload Report, BAW-1691, August 1981.

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limits for Unit 3 are 20.5 kw/ft for fuel rod burnup less than or equal to 10,000 MWD/MTU and 21.5 kw/ft - after 10,000 MWD/MTU.

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 2.1-3C correspond to the expected minimum flow rates with four pumps, three pumps, and one pump in each loop, respectively.

The magnitude of the rod bow penalty applied to each fuel cycle is equal to or greater than the necessary burnup dependent DNBR rod bow penalty for the applicable cycle minus a credit of 1% for the flow area reduction factor used in the hot channel analysis. All plant operating limits are based on a minimum DNBR criteria of 1.30 plus the amount necessary to offset the reduction in DNBR due to fuel rod bow. (4)

The maximum thermal power for three-pump operation is 90.65 percent due to a power level trip produced by the flux-flow ration 74.7 percent flow x 1.08 = 80.7 percent power plus the maximum calibration and instrument error (Reference 4). The maximum thermal power for other coolant pump conditions are produced in a similar manner.

For each curve of Figure 2.1-3C a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30 or a local quality at the point of minimum DNBR less than 22 percent for that particular reactor coolant pump situation. The curve of Figure 2.1-1C is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in Figure 2.1-3C.

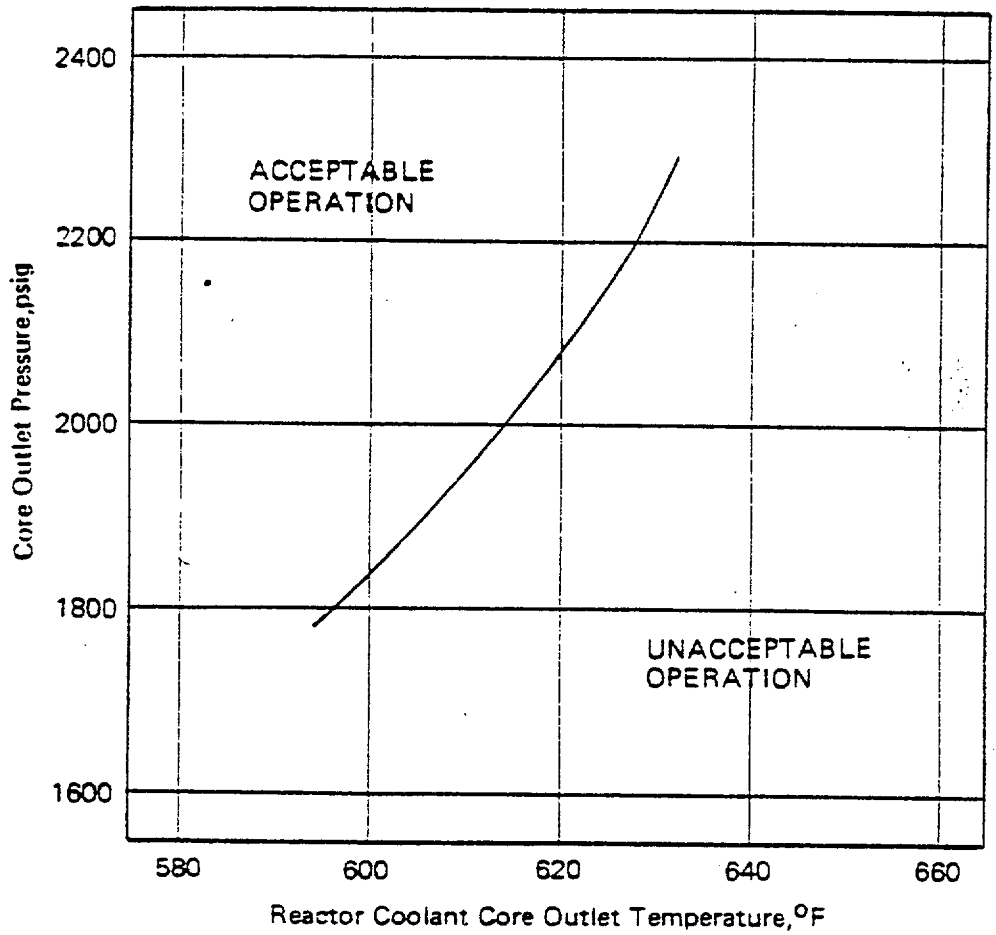
References

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

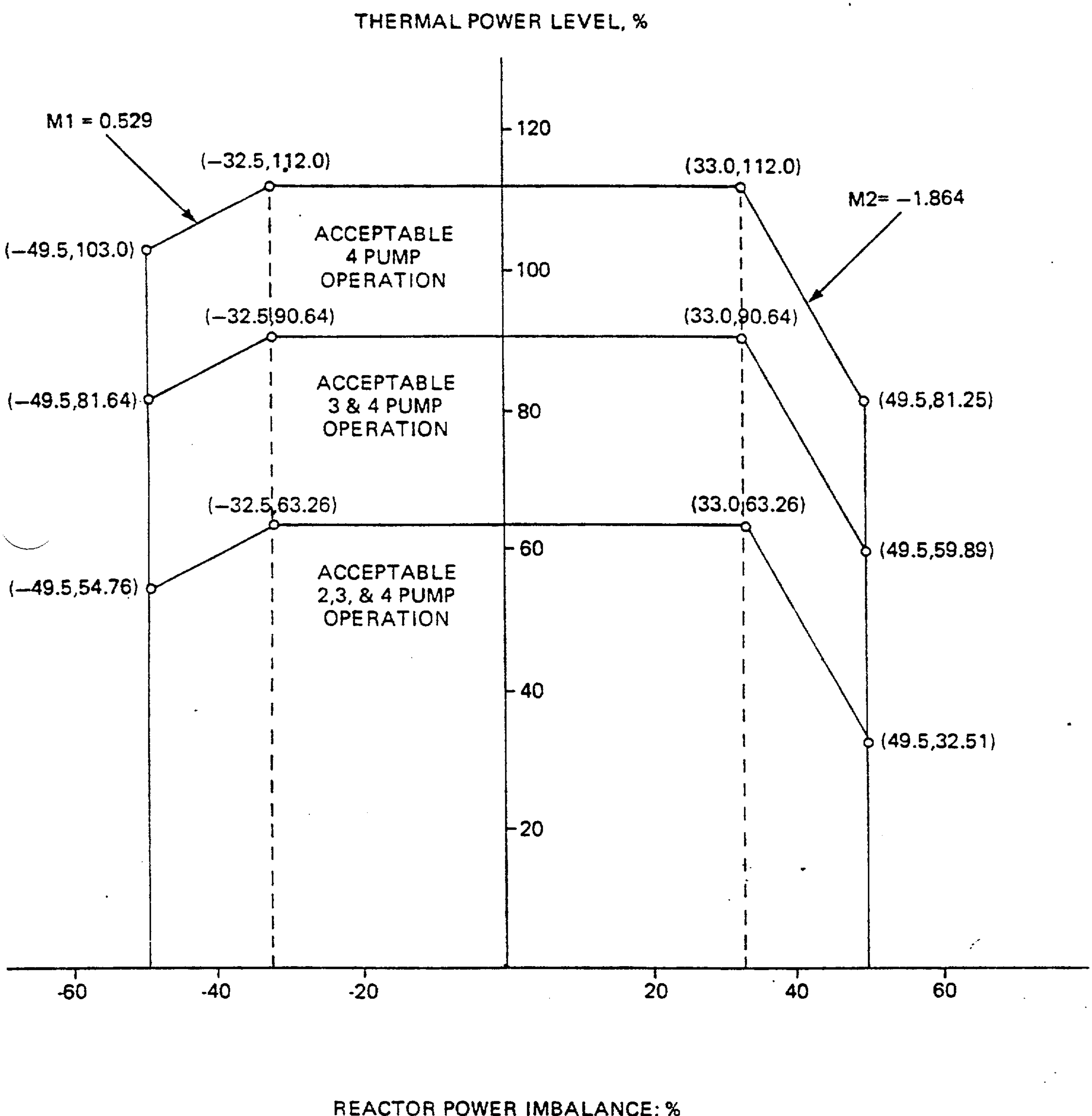
(2) Oconee 3, Cycle 3 - Reload Report - BAW-1453, August 1977.

(3) Amendment 1 - Oconee 3, Cycle 4 - Reload Report - BAW-1486, June 12, 1978.

(4) Oconee 3, Cycle 7 - Reload Report - DPC-RD-2001, Revision 1, July 1982.



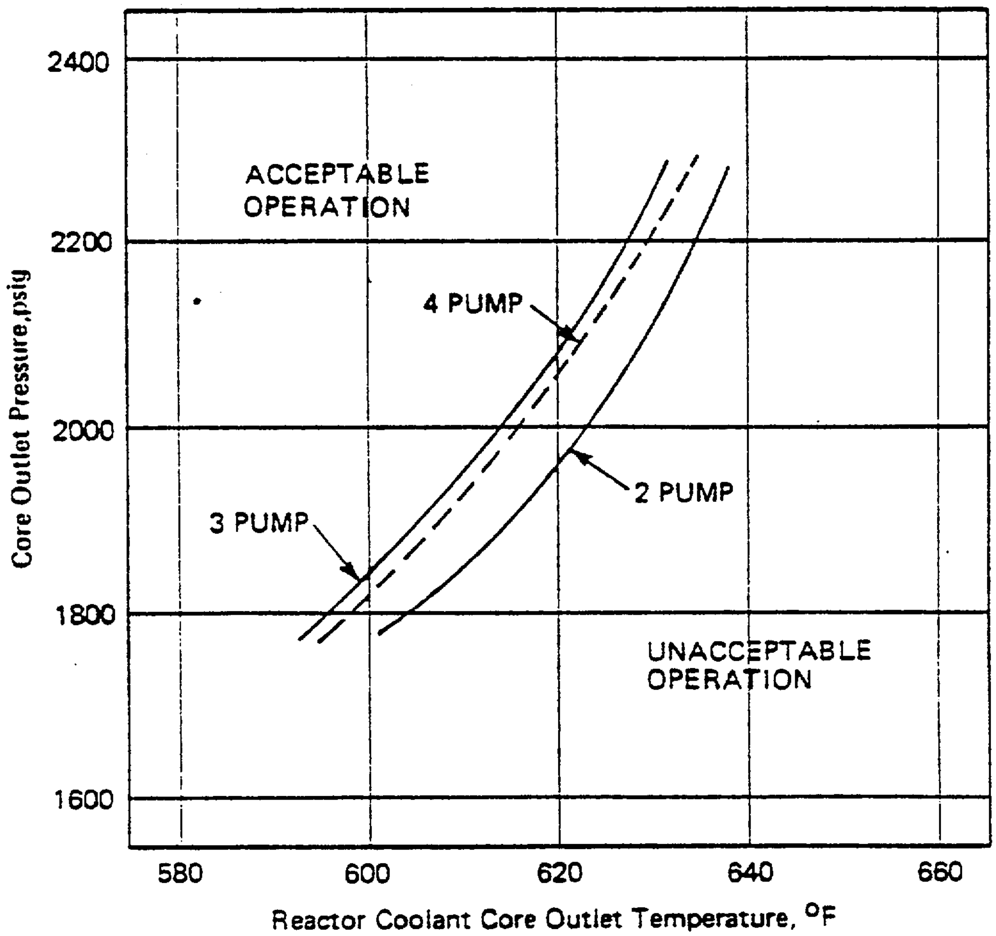
CORE PROTECTION SAFETY LIMITS
UNIT 3
OCONEE NUCLEAR STATION
Figure 2.1-1C



CORE PROTECTION SAFETY LIMIT
UNIT 3

OCONEE NUCLEAR STATION

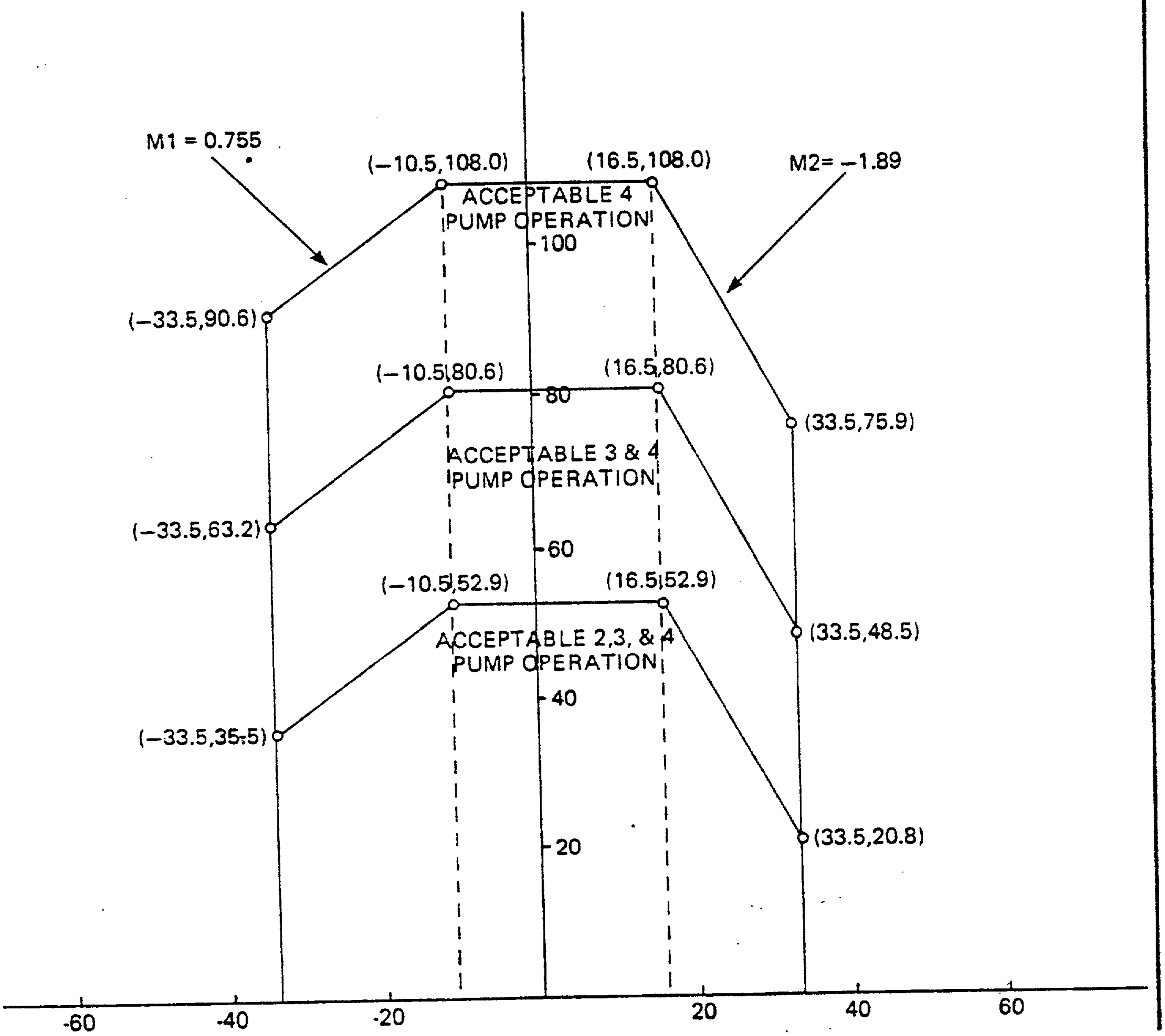
Figure 2.1-2C



PUMPS OPERATING	COOLANT FLOW (GPM)	POWER (% FP)	TYPE OF LIMIT
4	374,880(100%)	112.0	DNBR
3	280,035(74.7%)	90.7	DNBR
2	183,690(49.0%)	63.63	DNBR/QUALITY

CORE PROTECTION SAFETY LIMITS
UNIT 3
OCONEE NUCLEAR STATION
Figure 2.1-3C

THERMAL POWER LEVEL, %



REACTOR POWER IMBALANCE, %

PROTECTIVE SYSTEM
MAXIMUM ALLOWABLE SETPOINTS
UNIT 3
OCONEE NUCLEAR STATION
Figure 2.3-2C

Bases

The high pressure injection system and chemical addition system provide control of the reactor coolant system boron concentration.(1) This is normally accomplished by using any of the three high pressure injection pumps in series with a boric acid pump associated with either the boric acid mix tank or the concentrated boric acid storage tank. An alternate method of boration will be the use of the high pressure injection pumps taking suction directly from the borated water storage tank.(2)

The quantity of boric acid in storage in the concentrated boric acid storage tank or the borated water storage tank is sufficient to borate the reactor coolant system to a 1% $\Delta k/k$ subcritical margin at cold conditions (70°F) with the maximum worth stuck rod and no credit for xenon at the worst time in core life. The current cycles for each unit were analyzed with the most limiting case selected as the basis for all three units. Since only the present cycles were analyzed, the specifications will be re-evaluated with each reload. A minimum of 1020 ft^3 of 8,700 ppm boric acid in the concentrated boric acid storage tank, or a minimum of 350,000 gallons of 1835 ppm boric acid in the borated water storage tank (3) will satisfy the requirements. The volume requirements include a 10% margin and, in addition, allow for a deviation of 10 EFPD in the cycle length. The specification assures that two supplies are available whenever the reactor is critical so that a single failure will not prevent boration to a cold condition. The required amount of boric acid can be added in several ways. Using only one 10 gpm boric acid pump taking suction from the concentrated boric acid storage tank would require approximately 12.7 hours to inject the required boron. An alternate method of addition is to inject boric acid from the borated water storage tank using the makeup pumps. The required boric acid can be injected in less than six hours using only one of the makeup pumps.

The concentration of boron in the concentrated boric acid storage tank may be higher than the concentration which would crystallize at ambient conditions. For this reason, and to assure a flow of boric acid is available when needed, these tanks and their associated piping will be kept at least 10°F above the crystallization temperature for the concentration present. The boric acid concentration of 8,700 ppm in the concentrated boric acid storage tank corresponds to a crystallization temperature of 77°F and therefore a temperature requirement of 87°F. Once in the high pressure injection system, the concentrate is sufficiently well mixed and diluted so that normal system temperatures assure boric acid solubility.

REFERENCES

(1) FSAR, Section 9.1; 9.2
(2) FSAR, Figure 6.2
(3) Technical Specification 3.3

f. If the maximum positive quadrant power tilt exceeds the Maximum Limit of Table 3.5-1, the reactor shall be shut down within 4 hours. Subsequent reactor operation is permitted for the purpose of measurement, testing, and corrective action provided the thermal power and the Nuclear Overpower Trip Setpoints allowable for the reactor coolant pump combination are restricted by a reduction of 2% of thermal power for each 1% tilt for the maximum tilt observed prior to shutdown.

g. Quadrant power tilt shall be monitored on a minimum frequency of once every 2 hours during power operation above 15% full power.

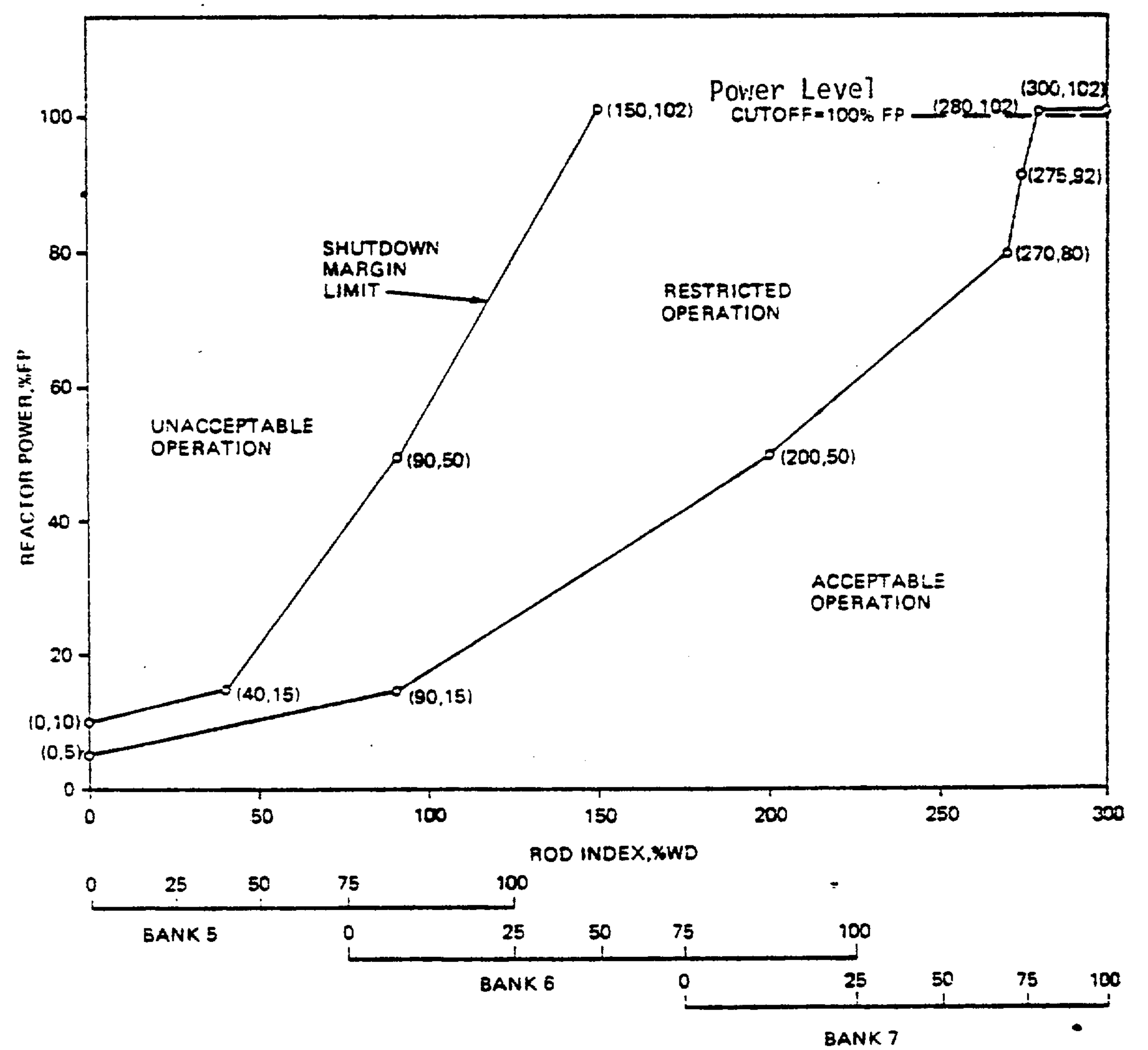
3.5.2.5 Control Rod Positions

a. Technical Specification 3.1.3.5 does not prohibit the exercising of individual safety rods as required by Table 4.1-2 or apply to inoperable safety rod limits in Technical Specification 3.5.2.2.

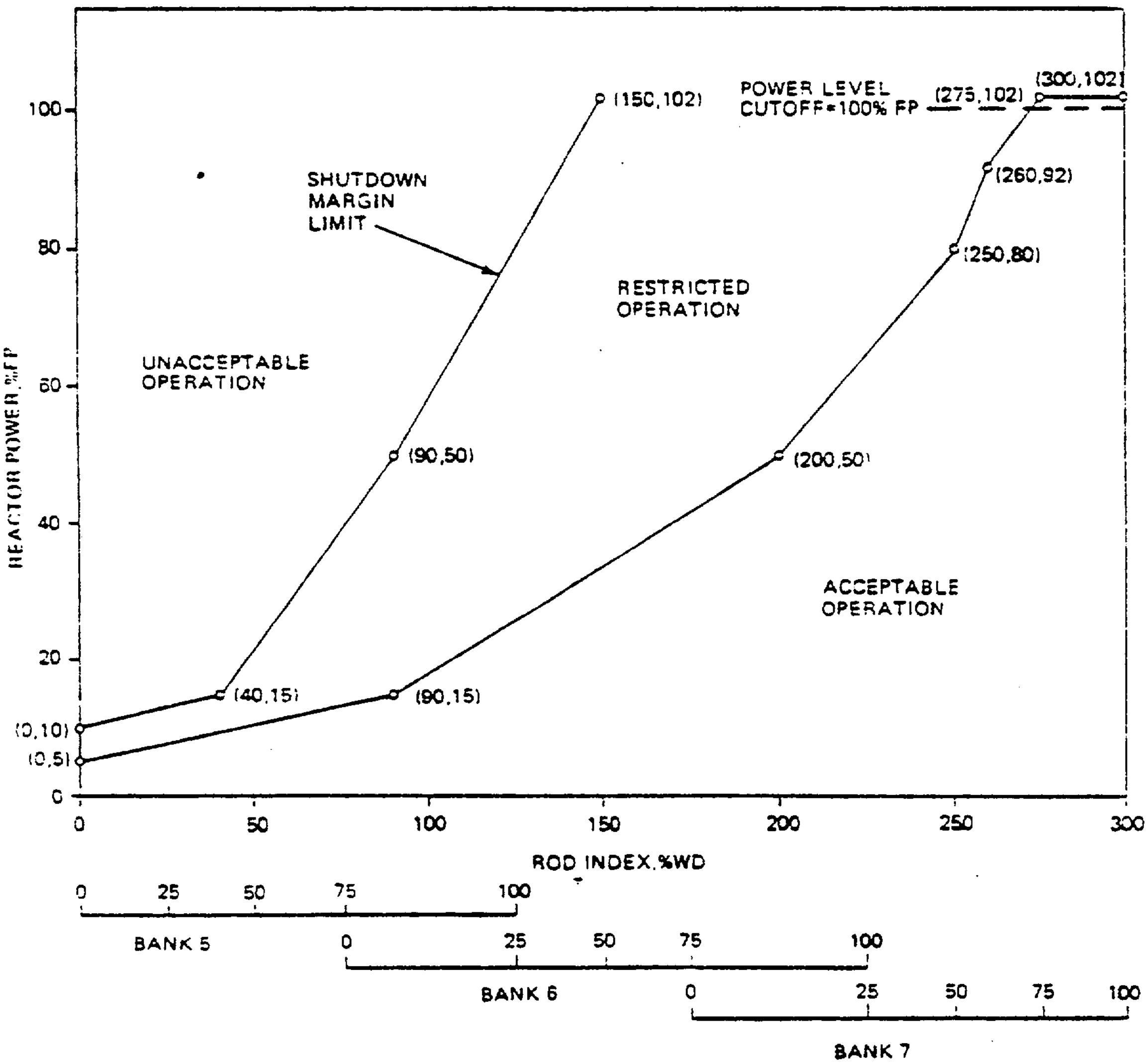
b. Except for physics tests, operating rod group overlap shall be 25% ± 5% between two sequential groups. If this limit is exceeded, corrective measures shall be taken immediately to achieve an acceptable overlap. Acceptable overlap shall be attained within two hours or the reactor shall be placed in a hot shutdown condition within an additional 12 hours.

c. Position limits are specified for regulating and axial power shaping control rods. Except for physics tests or exercising control rods, the regulating control rod insertion/withdrawal limits are specified on figures 3.5.2-1A1, 3.5.2-1A2, and 3.5.2-1A3 (Unit 1); 3.5.2-1B1, 3.5.2-1B2, and 3.5.2-1B3 (Unit 2); 3.5.2-1C1, 3.5.2-1C2, and 3.5.2-1C3 (Unit 3) for four pump operation, on figures 3.5.2-2A1, 3.5.2-2A2, and 3.5.2-2A3 (Unit 1); 3.5.2-2B1, 3.5.2-2B2, and 3.5.2-2B3 (Unit 2); figures 3.5.2-2C1, 3.5.2-2C2, and 3.5.2-2C3 (Unit 3) for three pump operation, and on figures 3.5.2-2A4, 3.5.2-2A5, and 3.5.2-2A6 (Unit 1); 3.5.2-2B4, 3.5.2-2B5, and 3.5.2-2B6 (Unit 2); figures 3.5.2-2C4, 3.5.2-2C5, and 3.5.2-2C6 (Unit 3) for two pump operation. Also, excepting physics tests or exercising control rods, the axial power shaping control rod insertion/withdrawal limits are specified on figures 3.5.2-4A1, 3.5.2-4A2, and 3.5.2-4A3 (Unit 1); 3.5.2-4B1, 3.5.2-4B2, and 3.5.2-4B3 (Unit 2); 3.5.2-4C1, 3.5.2-4C2, and 3.5.2-4C3 (Unit 3).

If the control rod position limits are exceeded, corrective measures shall be taken immediately to achieve an acceptable control rod position. An acceptable control rod position shall then be attained within two hours. The minimum shutdown margin required by Specification 3.5.2.1 shall be maintained at all times.

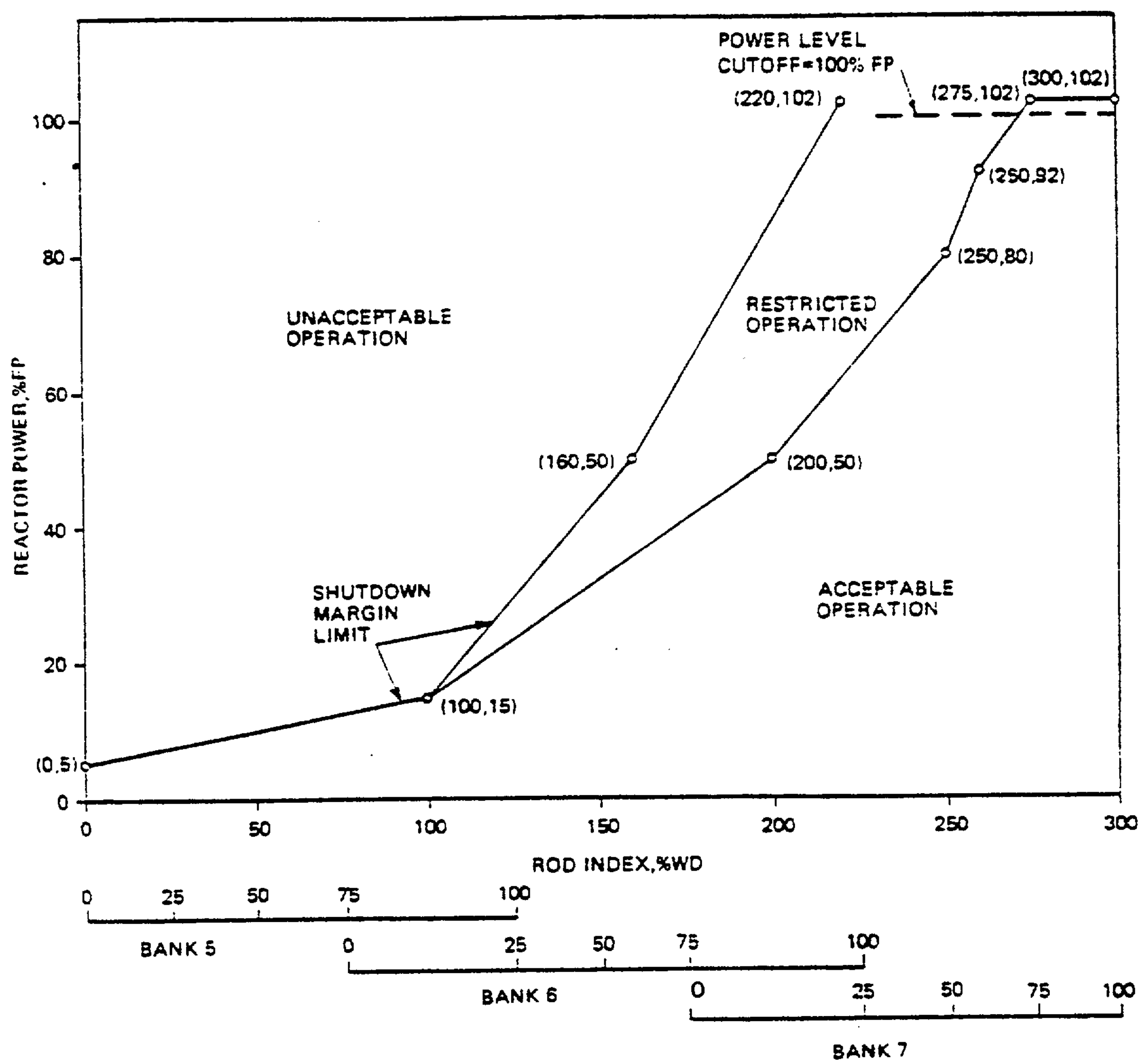


ROD POSITION LIMITS
FOR FOUR-PUMP OPERATION
FROM 0 TO 50 (+10, -0) EFPD
UNIT 3
OCONEE NUCLEAR STATION
Figure 3.5.2-1C1

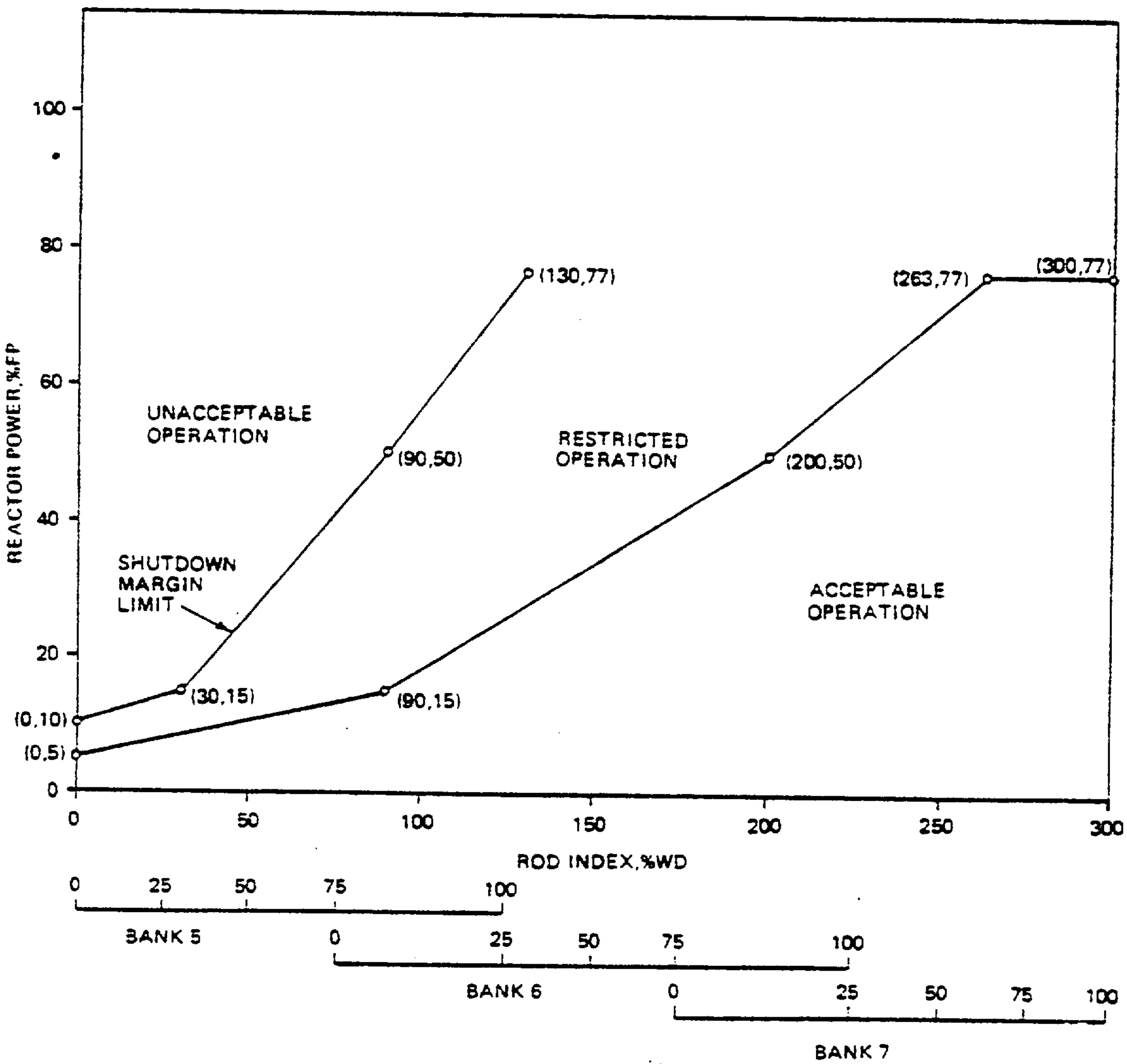


ROD POSITION LIMITS FOR
FOUR-PUMP OPERATION FROM
50 (+10, -0) TO 200 (±10) EFPD
UNIT 3

OCONEE NUCLEAR STATION
Figure 3.5.2-1C2

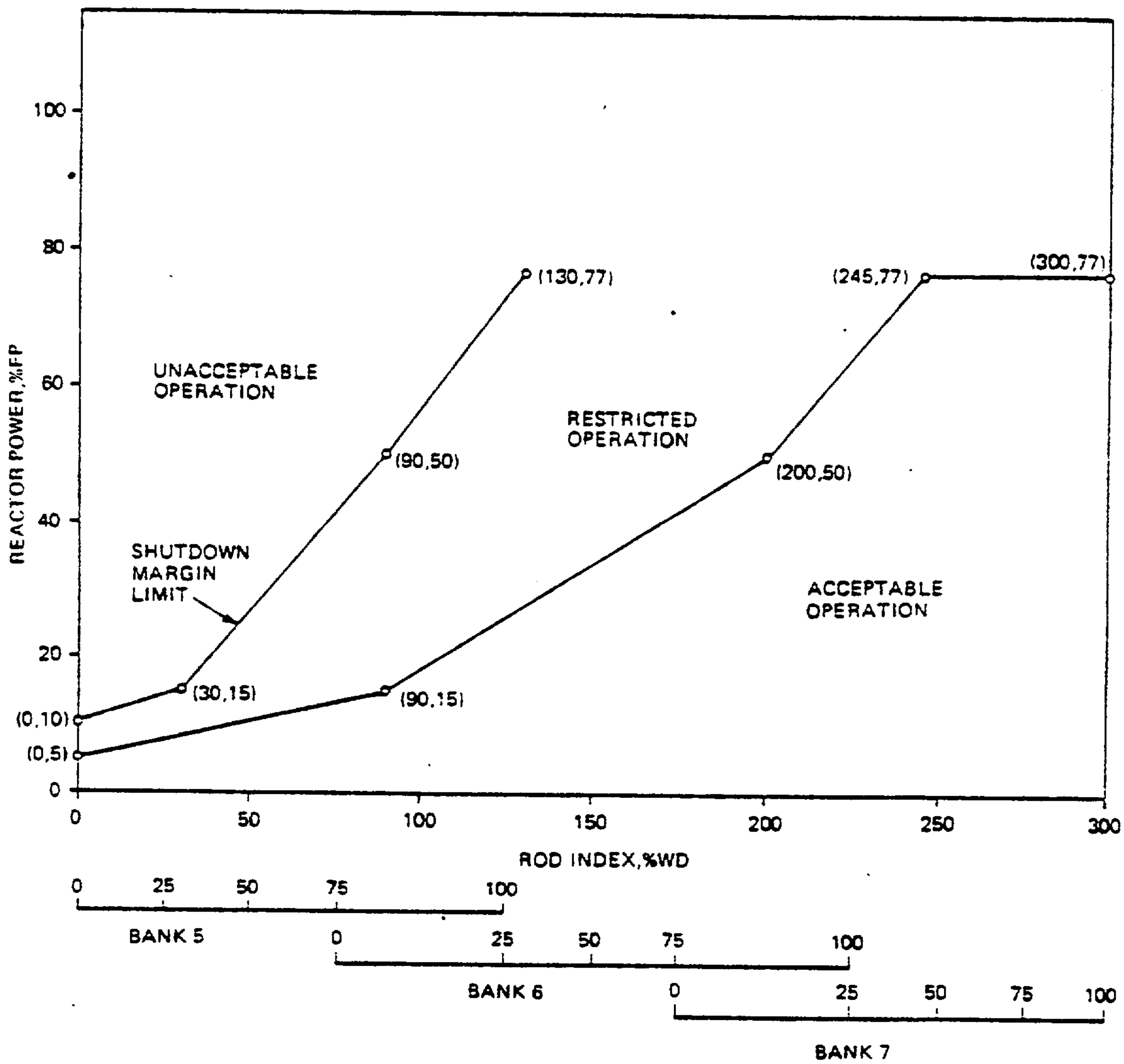


ROD POSITION LIMITS
FOR FOUR-PUMP OPERATION
AFTER 200 (±10) EFPD
UNIT 3
OCONEE NUCLEAR STATION
Figure 3.5.2-1C3



ROD POSITION LIMITS
FOR THREE-PUMP OPERATION
FROM 0 TO 50 (+10, -0) EFPD
UNIT 3
OCONEE NUCLEAR STATION

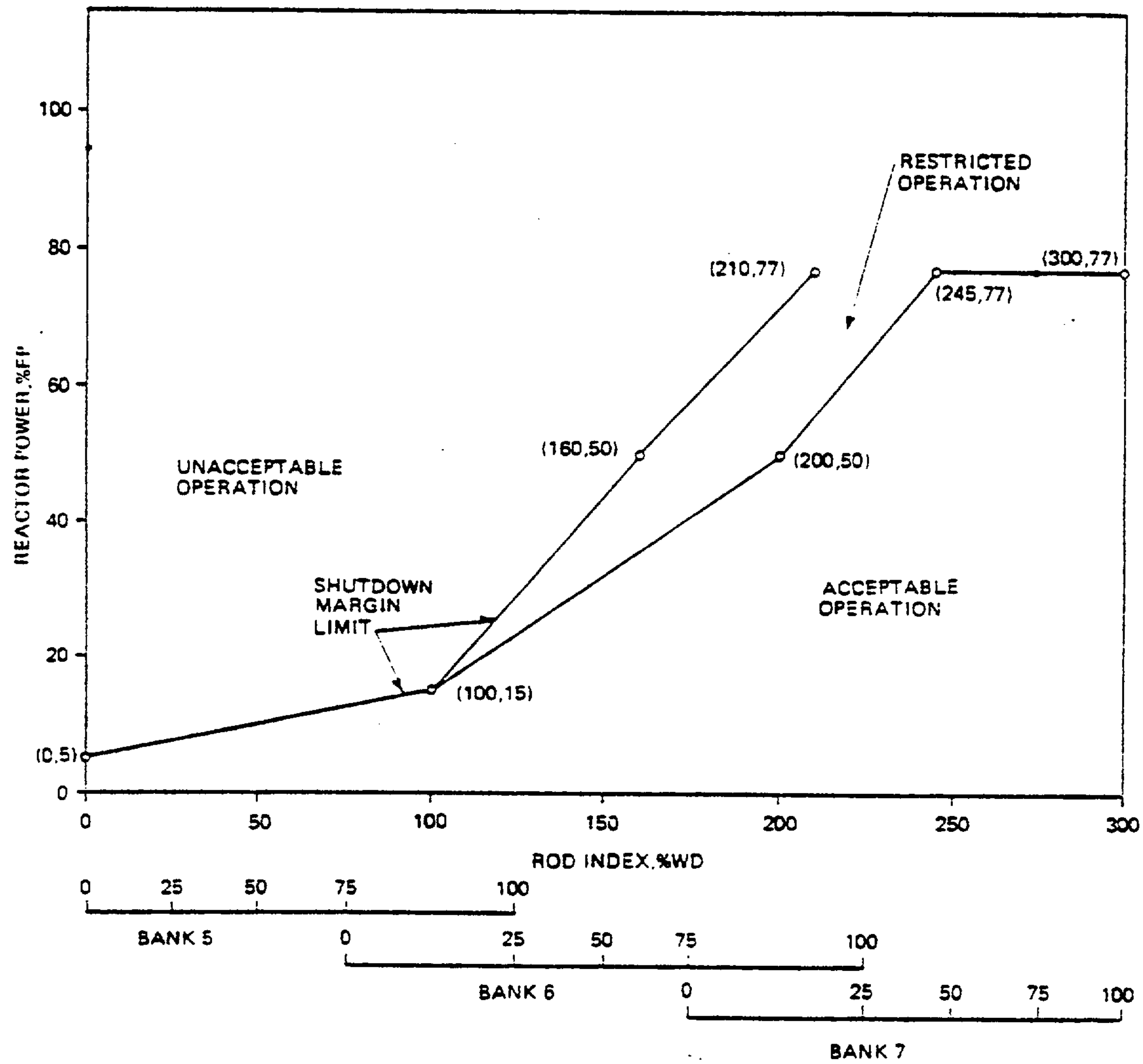
Figure 3.5.2-2C1



ROD POSITION LIMITS FOR THREE-PUMP OPERATION FROM 50 (+10, -0) TO 200 ±10 EFPD
UNIT 3

OCONEE NUCLEAR STATION

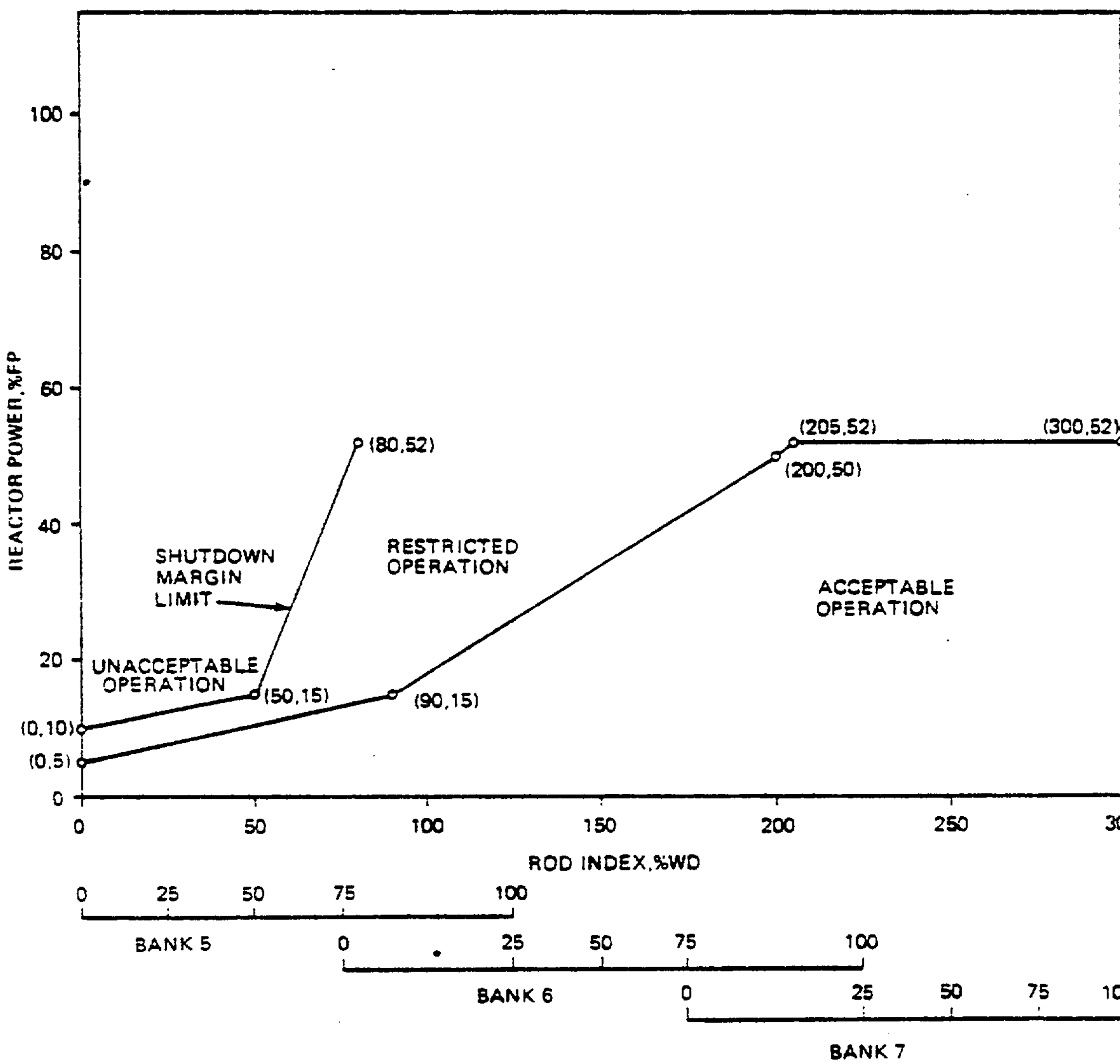
Figure 3.5.2-2C2



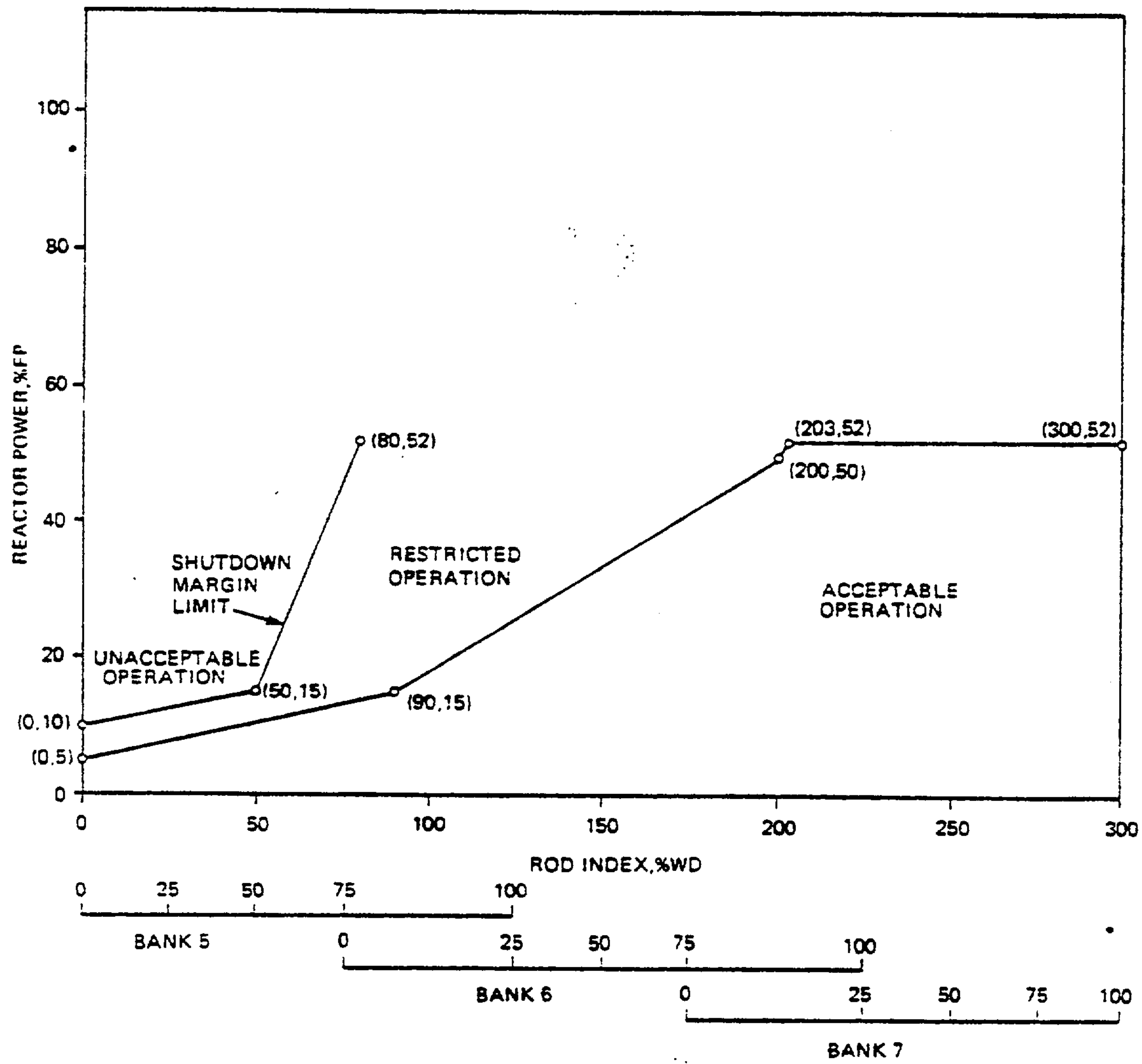
ROD POSITION LIMITS
FOR THREE-PUMP OPERATION
AFTER 200 ±10 EFPD
UNIT 3

OCONEE NUCLEAR STATION

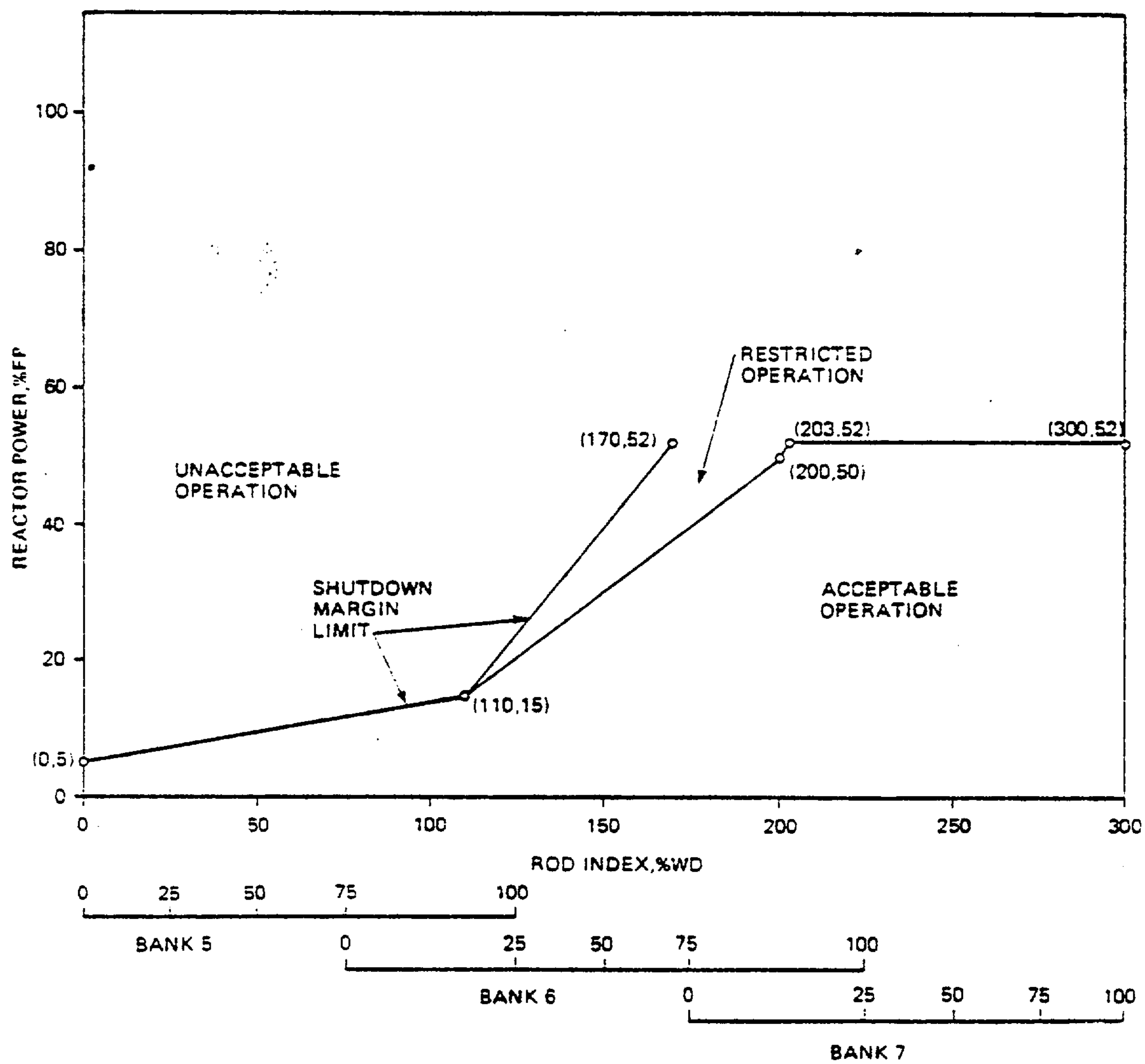
Figure 3.5.2-2C3



ROD POSITION LIMITS
FOR TWO-PUMP OPERATION
FROM 0 TO 50 (+10, -0) EFPD
UNIT 3
OCONEE NUCLEAR STATION
Figure 3.5.2-2C4

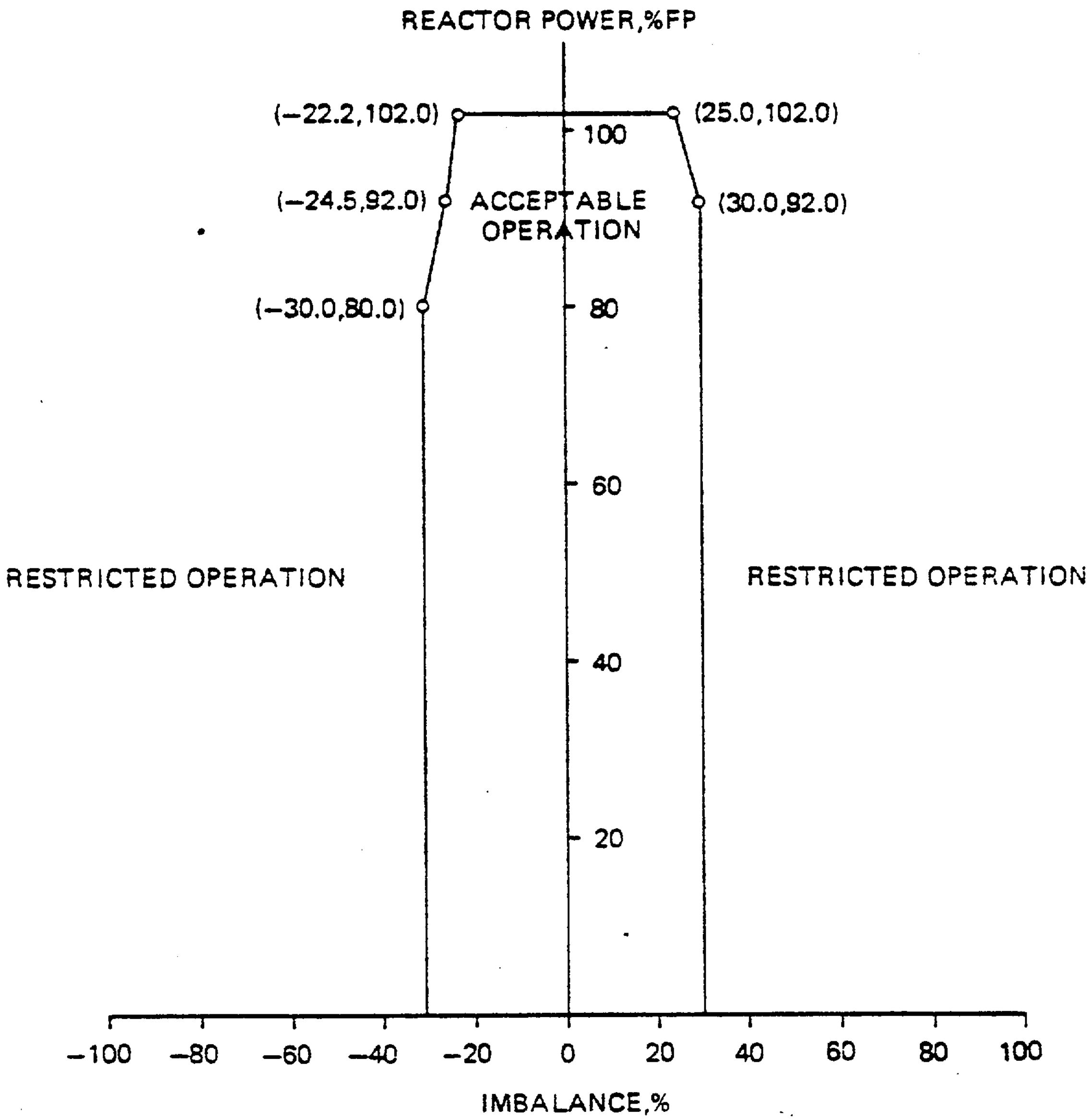


ROD POSITION LIMITS
FOR TWO-PUMP OPERATION FROM
50 (+10, -0) TO 200 ±10 EFPD
UNIT 3
OCONEE NUCLEAR STATION
Figure 3.5.2-2C5



ROD POSITION LIMITS FOR
TWO-PUMP OPERATION
AFTER 200 ±10 EFPD
UNIT 3
OCONEE NUCLEAR STATION

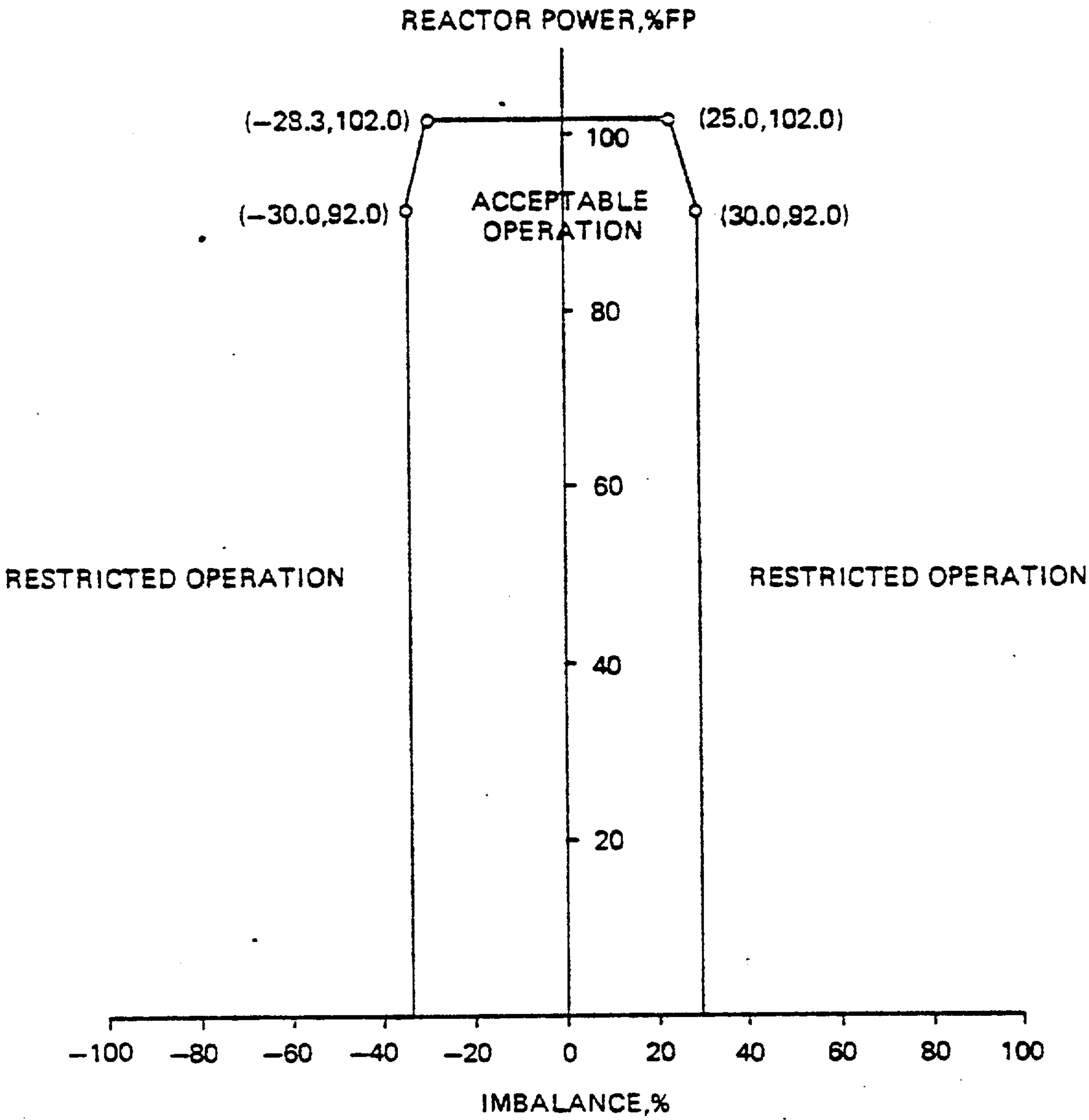
Figure 3.5.2-2C6



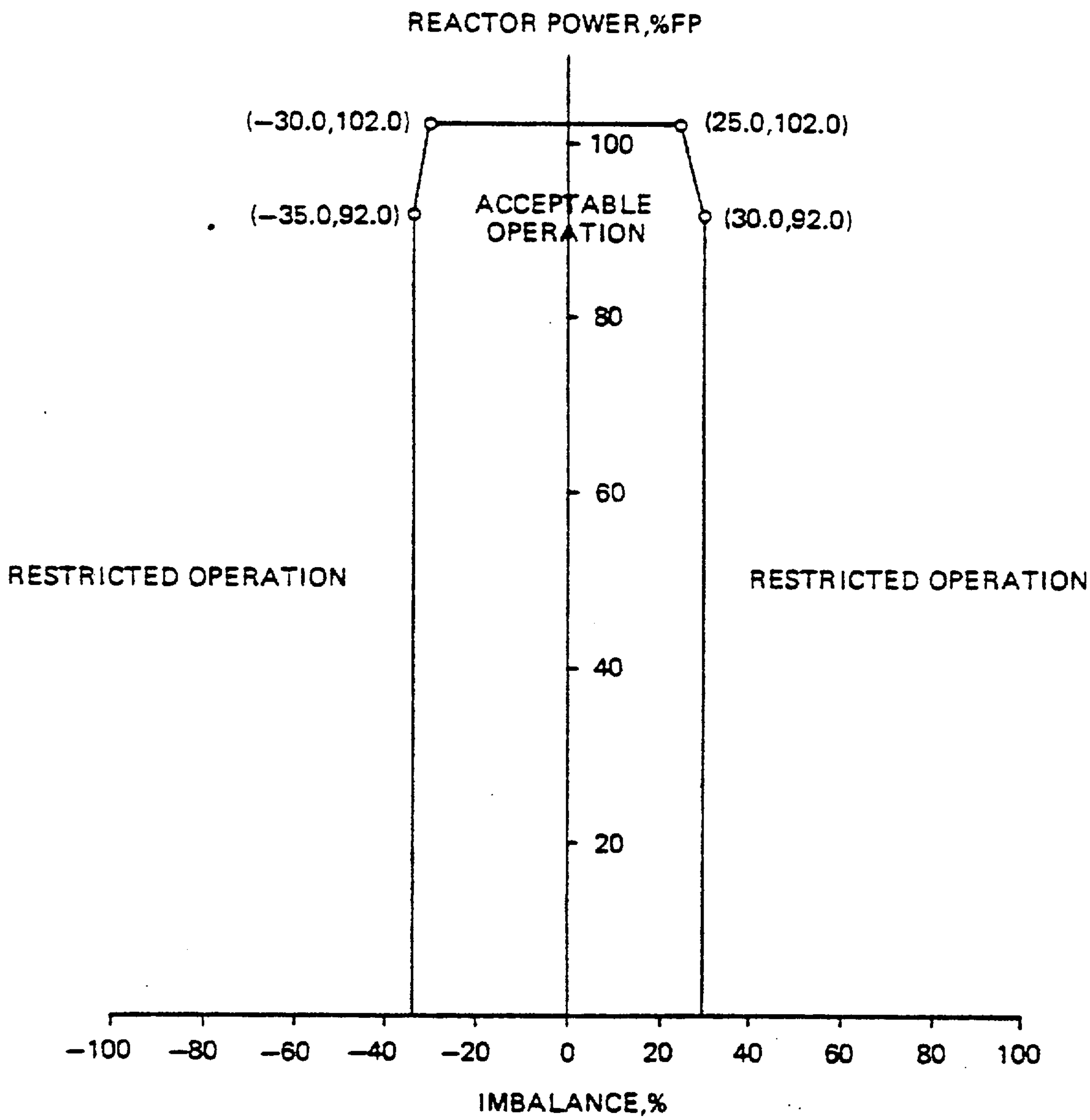
OPERATIONAL POWER
IMBALANCE ENVELOPE FROM
0 TO 50 (+10, -0) EFPD
UNIT 3

OCONEE NUCLEAR STATION

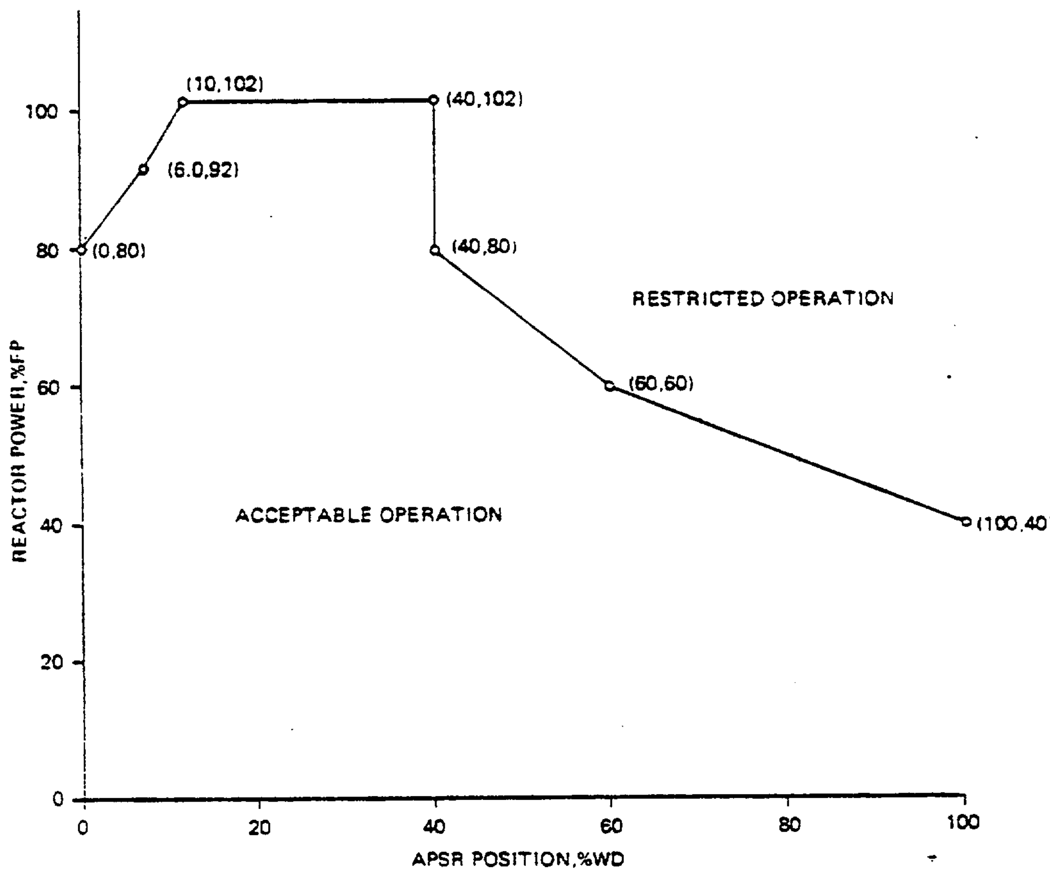
Figure 3.5.2-3C1



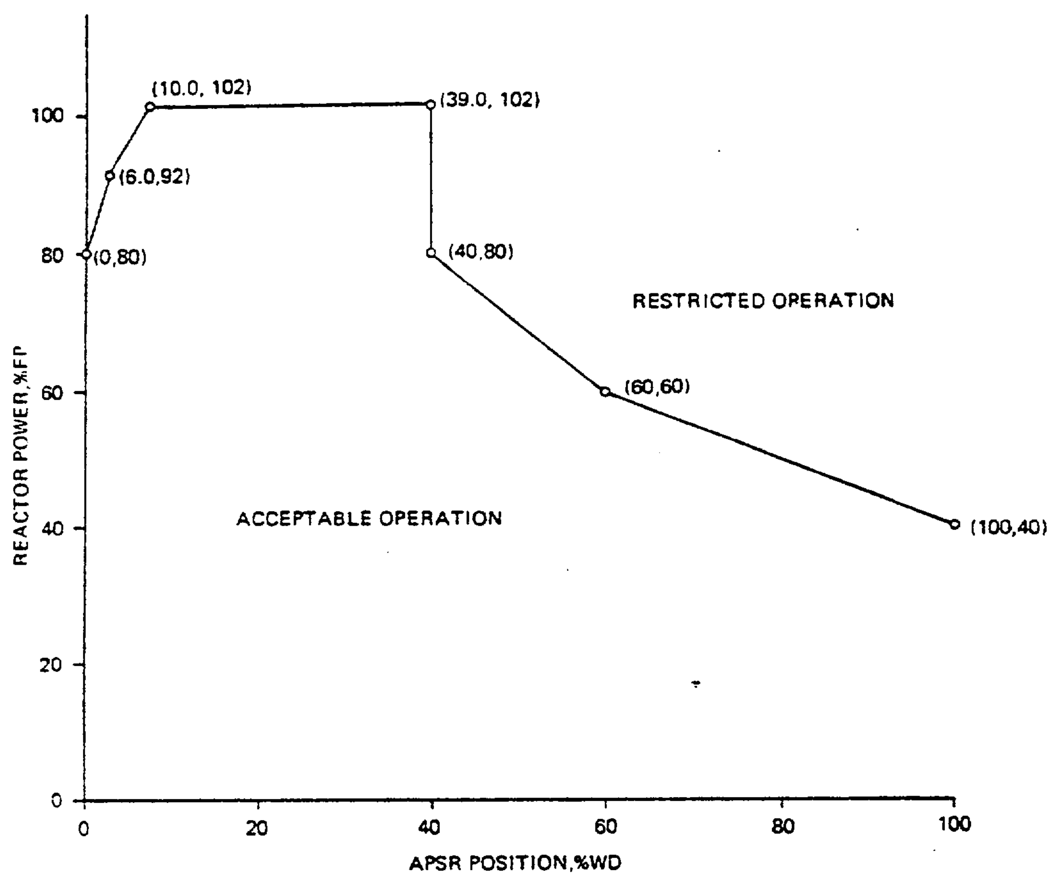
OPERATIONAL POWER
IMBALANCE ENVELOPE FROM
50 (+10, -0) TO 200 ±10 EFPD
UNIT 3
OCONEE NUCLEAR STATION
Figure 3.5.2-3C2



OPERATIONAL POWER
IMBALANCE ENVELOPE
AFTER 200 ±10 EFPD
UNIT 3
OCONEE NUCLEAR STATION
Figure 3.5.2-3C3



APSR POSITION LIMITS
FOR OPERATION
FROM 0 TO 200 ±10 EFPD
UNIT 3
OCONEE NUCLEAR STATION
Figure 3.5.2-4C1



APSR POSITION LIMITS
FOR OPERATION
AFTER 200 ±10 EFPD
UNIT 3

DUKE POWER

OCONEE NUCLEAR STATION
Figure 3.5.2-4C2

Figure 3.5.2-4C3

Deleted during Oconee Unit 3, Cycle 7 Operation

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 113 TO FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 113 TO FACILITY OPERATING LICENSE NO. DPR-47

AMENDMENT NO. 110 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

1.0 Introduction

By letter dated May 3, 1982, Duke Power Company (Duke or the licensee) submitted a license amendment application for the Oconee Nuclear Station (ONS) common Technical Specifications (TSs) to support full power operation of Oconee Unit 3 during fuel Cycle 7. Since changes to the common TSs are involved, a license amendment for all three units is necessary. The application had been compiled on an assumed fuel Cycle 6 length of 365 Effective Full Power Days (EFPD), but Unit 3 was shutdown 17 EFPD early due to unrelated, steam generator problems. This early shutdown required a number of reanalyses to account for the reduced fuel usage. Therefore, by letter dated August 11, 1982, Duke submitted a revised report which replaced the earlier submittal in its entirety. Additional information, requested in a telephone conference held on August 12, 1982, was provided by letter dated August 16, 1982.

The report attached to the August 11, 1982 submittal (Reload Report) was compiled using the "Oconee Nuclear Station Reload Design Methodology", Duke Technical Report NFS-1001, which was approved by NRC letter dated July 29, 1981. Additionally, the startup testing will be performed in accordance with the "Oconee Nuclear Station Generic Startup Physics Test Program" which we approved by letters dated March 23 and May 29, 1981.

The 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 7 is to have an extended length of approximately 440 EFPD. For this reason, burnable poison assemblies are used to limit the required beginning-of-cycle (BOC) soluble boron concentration. Cycle 7 will operate in a rods-out, feed-and-bleed mode as did Cycle 6.

2.0 Evaluation

2.1 Fuel Assembly Design

Although all batches in the Oconee 3 Cycle 7 core will utilize the same Babcock and Wilcox 15x15 fuel design, the Batch 9 assemblies will be of the Mark B5, as opposed to the previously-loaded Mark B4, fuel design. The Mark B5 fuel assembly is identical to the Mark B4 except its upper end fitting has been redesigned to provide a positive holddown of fixed control components such as burnable poison rod assemblies (BPRAs), neutron source rod assemblies, and orifice rod assemblies. Oconee 3 Cycle 7 is the first application of the new design. We have determined that no special treatment of the Mark B5 fuel assembly is necessary because the thermal-hydraulic and fuel rod mechanical analyses are unaffected.

Although the Oconee 3 Cycle 7 core will contain both Mark B4 and Mark B5 fuel assemblies, the fuel rods used in both assemblies are virtually identical. The results of the linear-heat-rate-to-melt analysis show slightly different densification characteristics for the new, Batch 9 fuel as opposed to previous batches. However, the resulting linear heat rate (LHR) values are the same for all batches in the Cycle 7 core. We regard such design changes as within the range of expected fuel rod design variation and, therefore, find them acceptable. Fuel rod cladding collapse, stress and strain, fuel rod internal pressure and fuel rod bowing were all acceptably analyzed.

2.2 Nuclear Design

Comparisons were made between the physics parameters for Cycles 6 and 7. The differences that exist between the parameters are due to the increased cycle length, which tends to increase values of critical boron concentrations. Changes in the radial flux and burnup distributions between cycles also account for the differences in control rod worths, including ejected and stuck rod worths. All safety criteria are still met. Shutdown margin values at beginning and end of cycle are 3.67 and 2.26 percent $\Delta k/k$, respectively, compared to the minimum required value of 1.0 percent. Beginning of cycle radial power distributions show acceptable margins to limits. 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 Cycle 7 is acceptable.

2.3 Thermal-Hydraulic Design

In order to confirm that the thermal-hydraulic design of the reload core has been accomplished using acceptable methods and provide acceptable margin of safety from conditions which could lead to fuel damage during normal and anticipated operational transients, comparisons of the differences between Cycle 7 and Cycle 6 were performed. The main differences are decreased core bypass flow and fuel rod bow compensation and have been shown to be of little consequence for Cycle 7. We, therefore, conclude that the available margin for Cycle 7 has been demonstrated and that the thermal-hydraulic design is acceptable.

2.4 Accident Analyses

The important kinetics parameters for Cycle 7 have been compared to the values used in the Final Safety Analysis Report (FSAR). The initial conditions of the transients in Cycle 7 are bounded by those assumed in the FSAR, and the safety analyses of Cycle 7 are, therefore, bounded by previously accepted analyses.

Two sets of bounding values for allowable Loss of Coolant Accident (LOCA) peak LHRs are given as a function of core height. The first set, which covers the first 50 EFPD, includes reduced LOCA kW/ft limits at low core elevations and are based on the interim LOCA LHR limits. The second set, which covers the balance of the cycle, are the Final Acceptance Criteria LOCA LHR limits. Those limits are identical to those approved for the previous cycle and are satisfactorily incorporated into the TSs for Cycle 7 through the operating limits on control rod position and axial power imbalance.

2.5 Technical Specification Modifications

Oconee Unit 3, Cycle 7 TSs have been modified to account for minor changes in power peaking and control rod worths due to the transition to an 18-month, lumped burnable poison cycle.

We have reviewed the proposed TS revisions for Cycle 7. 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 of Specification 2.5.2. The limiting safety system settings and the limiting condition for operation have been established by approved methods. Changes which reflect the core thermal-hydraulic response still maintain the safety limit Departure from Nucleate Boiling Ratio (DNBR) criterion of 1.30. The control rod withdrawal limits for the various pump combinations and times in core life are presented as well as part length axial power shaping rod position limits. On the basis that previously approved methods were used to obtain the limits, we find these TS modifications acceptable.

Editorial changes were also made for the bases for Units 1 and 2 (pages 2.1-2 and 2.1-3b) to correct referenced Figures. Since these are editorial only, we find them acceptable.

2.6 Summary

We have reviewed the fuels, physics, thermal-hydraulic and accident analyses information presented in the Oconee 3 Cycle 7 reload report. We find the proposed reload and the associated modified TSs acceptable.

3.0 Environmental Consideration

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

4.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated, do not create the possibility of an accident of a type different from any evaluated previously and do not involve a significant reduction in a margin of safety, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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: September 29, 1982

The following NRC staff personnel have contributed to this Safety Evaluation: P. C. Wagner, L. Kopp, A. Gill, J. Voglewede and S. Sun.

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

DUKE POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 113, 113 and 110 to Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55, respectively, issued to Duke Power Company, which revised the Technical Specifications (TSs) for operation of the Oconee Nuclear Station, Units Nos. 1, 2 and 3, located in Oconee County, South Carolina. The amendments are effective as of the date of issuance.

These amendments revise the common TSs to allow full power operation of Oconee Unit 3 during fuel Cycle 7.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR Section 51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

For further details with respect to this action, see (1) the application for amendments dated May 3, 1982 as revised in its entirety on August 11, 1982, and supplemented on August 16, 1982, (2) Amendments Nos. 113 , 113 , and 110 to Licenses Nos. DPR-38, DPR-47 and DPR-55, respectively, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Oconee County Library, 501 West Southbroad Street, Walhalla, South Carolina. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 29th day of September 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

John T. Stolz

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