DUKE POWER COMPANY

OCONEE NUCLEAR STATION

ATTACHMENT 1

PROPOSED TECHNICAL SPECIFICATION REVISION

OCONEE 2, CYCLE 9

REMOVE PAGES	INSERT PAGES
3.1-8	3.1-8
3.1-9	3.1-9
3.1-23	3.1-23
3.5-7	3.5-7
3.5-8	3.5-8
3.5-9	3.5-9
3.5-10	3.5-10
3.5-11	3.5-11
3.5-12	3.5-12
	6.6-3a

3.1.3 Minimum Conditions for Criticality

Specification

3.1.3.1 The reactor coolant temperature shall be above 525°F except for portions of low power physics testing when the requirements of Specification 3.1.9 shall apply.

3.1.3.2 Reactor coolant temperature shall be above the criticality limit of
3.1.2-1A (Unit 1)
3.1.2-1B (Unit 2)
3.1.2-1C (Unit 3)

3.1.3.3 When the reactor coolant temperature is below the minimum temperature specified in 3.1.3.1 above, except for portions of low power physics testing when the requirements of Specification 3.1.9 shall apply, the reactor shall be subcritical by an amount equal to or greater than the calculated reactivity insertion due to depressurization.

3.1.3.4 The reactor shall be maintained subcritical by at least 1%Δk/k until a steam bubble is formed and a water level between 80 and 396 inches is established in the pressurizer.

3.1.3.5 Except for physics tests and as limited by 3.5.2.1, safety rod groups shall be fully withdrawn prior to any other reduction in shutdown margin by deboration or regulating rod withdrawal during the approach to criticality. The regulating rods shall then be positioned within their acceptable operating position limits.

Bases

At the beginning of the initial fuel cycle, the moderator temperature coefficient is expected to be slightly positive at operating temperatures with the operating configuration of control rods.(1) Calculations show that above 525°F, the consequences are acceptable.

Since the moderator temperature coefficient at lower temperatures will be less negative more more positive than at operating temperature,(2) startup and operation of the reactor when reactor coolant temperature is less than 525°F is prohibited except where necessary for low power physics tests.

The potential reactivity insertion due to the moderator pressure coefficient(2) that could result from depressurizing the coolant from 2100 psia to saturation pressure of 900 psia is approximately 0.1% Δk/k.

During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient(1) and the small integrated Δk/k would limit the magnitude of a power excursion resulting from a reduction of moderator density.

The requirement that the reactor is not to be made critical below the limits of Specification 3.1.2.1 provides increased assurance that the proper relationship between primary coolant pressure and temperature will be maintained relative to the RT_{NDT} of the primary coolant system. Heatup to this temperature will be accomplished by operating the reactor coolant pumps.

If the shutdown margin required by Specification 3.5.2 is maintained, there is no possibility of an accidental criticality as a result of a decrease of coolant pressure.

The requirement for pressurizer bubble formation and specified water level when the reactor is less than 1% subcritical will assure that the reactor coolant system cannot become solid in the event of a rod withdrawal accident or a startup accident.(3)

The requirement that the safety rod groups be fully withdrawn before criticality ensures shutdown capability during startup. This does not prohibit rod latch confirmation, i.e., withdrawal by group to a maximum of 3 inches withdrawn of all seven groups prior to safety rod withdrawal.

The requirement for regulating rods being within their rod position limits ensures that the shutdown margin and ejected rod criteria at hot zero power are not violated. The acceptable operating position limits for the regulating rods for the appropriate unit and cycle are provided in the Core Operational Limits Report.

REFERENCES

(1) FSAR, Section 4.3.2

(2) FSAR, Section 4.3.2.4

(3) FSAR, Section 15.3

3.1.11 Shutdown Margin

Specification

The available shutdown margin during all system conditions except refueling shall be greater than 1% $\Delta k/k$ with the highest worth control rod fully withdrawn.

Bases

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

During power operation and startup the SHUTDOWN MARGIN is known to be within limits if all control rods are OPERABLE and withdrawn to or beyond the insertion limits provided in the Core Operational Limits Report, for the appropiate unit and cycle.

During refueling conditions equivalent protection is provided in the requirements of Specification 3.8.4.

c. If a control rod is declared inoperable by being immovable due to excessive friction or mechanical interference or known to be untrippable then:

   1. Within 1 hour verify that the shutdown margin requirement of Specification 3.5.2.1 is satisfied and,

   2. Within 12 hours place the reactor in the hot standby condition.

d. If a control rod is declared inoperable due to causes other than addressed in 3.5.2.2.c above then:

   1. Within 1 hour either restore the rod to operable status or,

   2. Continue power operation with the control rod declared inoperable and

      a. Within 1 hour verify the shutdown margin requirement of Specification 3.5.2.1 with an additional allowance for the withdrawn worth of the inoperable rod and,

      b. Either reactor thermal power shall be reduced to less than 60% of the allowable power for the reactor coolant pump combination within 1 hour and the Nuclear Overpower Trip Setpoints, based on flux and flux/flow/imbalance, shall be reduced within the next 4 hours to 65.5% of thermal power value allowable for the reactor coolant pump combination or,

      c. Position the remaining rods in the affected group such that the inoperable rod is maintained within allowable group average limits of Specification 3.5.2.2.a and within acceptable operating rod position withdrawal/ insertion limits.

e. If more than one control rod is inoperable or misaligned, the reactor shall be shut down to the hot standby condition within 12 hours.

3.5.2.3 The worths of single inserted control rods during criticality are limited by the restrictions of Specification 3.1.3.5 and the control rod position limits.

3.5.2.4 Quadrant Power Tilt

a. Except for physics tests, the maximum positive quadrant power tilt shall not exceed the Steady State Limit of Table 3.5-1 during power operation above 15% full power.

b. If the maximum positive quadrant power tilt exceeds the Steady State Limit but is less than or equal to the Transient Limit of Table 3.5-1, then:

1. Either the quadrant power tilt shall be reduced within 2 hours to within its Steady State Limit or,

2. The reactor thermal power shall be reduced below the power level cutoff and further reduced 2% thermal power for each 1% of quadrant power tilt in excess of the Steady State Limit, and the Nuclear Overpower Trip Setpoints, based on flux and flux/flow imbalance, shall be reduced within 4 hours by 2% thermal power for each 1% tilt in excess of the Steady State Limit. If less than four reactor coolant pumps are in operation, the allowable thermal power for the reactor coolant pump combination shall be reduced by 2% for each 1% excess tilt.

c. Quadrant power tilt shall be reduced within 24 hours to within its Steady State Limit or,

1. The reactor thermal power shall be reduced within the next 2 hours to less than 60% of the allowable power for the reactor coolant pump combination and the Nuclear Overpower Trip Setpoints, based on flux and flux/flow imbalance, shall be reduced within the next 4 hours to 65.5% of the thermal power value allowable for the reactor coolant pump combination.

d. If the quadrant power tilt exceeds the Transient Limit but is less than the Maximum Limit of Table 3.5-1 and if there is a simultaneous indication of a misaligned control rod then:

1. Reactor thermal power shall be reduced within 30 minutes at least 2% for each 1% of the quadrant power tilt in excess of the Steady State Limit.

2. Either quadrant power tilt shall be reduced within 2 hours to within its Transient Limit or,

3. The reactor thermal power shall be reduced within the next 2 hours to less than 60% of the allowable power for the reactor coolant pump combination and the Nuclear Overpower Trip Setpoints, based on flux and flux/flow imbalance, shall be reduced within the next 4 hours to 65.5% of the thermal power value allowable for the reactor coolant pump combination.

e. If the quadrant power tilt exceeds the Transient Limit but is less than the Maximum Limit of Table 3.5-1, due to causes other than simultaneous indication of a misaligned control rod then:

1. Reactor thermal power shall be reduced within 2 hours to less than 60% of the allowable power for the reactor coolant pump combination and the Nuclear Overpower Trip Setpoints, based on flux and flux/flow imbalance, shall be

reduced within the next 2 hours to 65.5% of the thermal power value allowable for the reactor coolant pump combination.

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 shall be maintained within acceptable operating limits for the particular number of operating reactor coolant pumps (4,3,2).

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.

3.5.2.6 Xenon Reactivity

Except for physics tests, reactor power shall not be increased above the power level-cutoff unless one of the following conditions is satisfied:

1. Xenon reactivity did not deviate more than 10 percent from the equilibrium value for operation at steady state power.

2. Xenon reactivity deviated more than 10 percent but is now within 10 percent of the equilibrium value for operation at steady state rated power and has passed its final maximum or minimum peak during its approach to its equilibrium value for operation at the power level cutoff.

3. Except for xenon free startup (when 2. applies), the reactor has operated within a range of 87 to 92 percent of rated thermal power for a period exceeding 2 hours.

3.5.2.7 Reactor power imbalance shall be monitored on a frequency not to exceed two hours during power operation above 40 percent rated power. Except for physics tests, imbalance shall be maintained within the acceptable operating envelope. If the imbalance is not within the acceptable operating envelope, corrective measures shall be taken to achieve an acceptable imbalance. If an acceptable imbalance is not achieved within two hours, reactor power shall be reduced until imbalance limits are met.

3.5.2.8 The control rod drive patch panels shall be locked at all times with limited access to be authorized by the manager or his designated alternate.

Bases

Operation at power with an inoperable control rod is permitted within the limits provided. These limits assure that an acceptable power distribution is maintained and that the potential effects of rod misalignment on associated accident analyses are minimized. For a rod declared inoperable due to misalignment, the rod with the greatest misalignment shall be evaluated first. Additionally, the position of the rod declared inoperable due to misalignment shall not be included in computing the average position of the group for determining the operability of rods with lesser misalignments. When a control rod is declared inoperable, boration may be initiated to achieve the existence of 1% $\Delta k/k$ hot shutdown margin.

The power-imbalance envelope defined in the Core Operational Limits Report is based on LOCA analyses which have defined the maximum linear heat rate (see Figure 3.5.2-5) such that the maximum clad temperature will not exceed the Final Acceptance Criteria. Corrective measures will be taken immediately should the indicated quadrant tilt, rod position, or imbalance be outside their specified boundary. Operation in a situation that would cause the Final Acceptance Criteria to be approached should a LOCA occur is highly improbable because all of the power distribution parameters (quadrant tilt, rod position, and imbalance) must be at their limits while simultaneously all other engineering and uncertainty factors are also at their limits.** Conservatism is introduced by application of:

a. Nuclear uncertainty factors
b. Thermal calibration
c. Fuel densification power spike factors
d. Hot rod manufacturing tolerance factors
e. Fuel rod bowing power spike factors

The 25% ± 5% overlap between successive control rod groups is allowed since the worth of a rod is lower at the upper and lower part of the stroke. Control rods are arranged in groups or banks defined as follows:

Group	Function
1	Safety
2	Safety
3	Safety
4	Safety
5	Regulating
6	Regulating
7	Regulating
8	APSR (axial power shaping rod)

** Actual operating limits depend on whether or not incore or excore detectors are used and their respective instrument calibration errors. The method used to define the operating limits is defined in plant operating procedures.

The rod position limits provided in the Core Operational Limits Report are based on the most limiting of the following three criteria: ECCS power peaking, shutdown margin, and potential ejected rod worth. Therefore, compliance with the ECCS power peaking criterion is ensured by the rod position limits. The minimum available rod worth, consistent with the rod position limits, provides for achieving hot shutdown by reactor trip at any time, assuming the highest worth control rod that is withdrawn remains in the full out position(1). The rod position limits also ensure that inserted rod groups will not contain single rod worths greater than 0.65% $\Delta k/k$ at rated power. These values have been shown to be safe by the safety analysis (2,3,4, 5) of hypothetical rod ejection accident. A maximum single inserted control rod worth of 1.0% $\Delta k/k$ is allowed by the rod position limits at hot zero power. A single inserted control rod worth of 1.0% $\Delta k/k$ at beginning-of-life, hot zero power would result in a lower transient peak thermal power and, therefore, less severe environmental consequences than a 0.65% $\Delta k/k$ ejected rod worth at rated power.

Control rod groups are withdrawn in sequence beginning with Group 1. Groups 5, 6, and 7 are overlapped 25 percent. The normal position at power is for Group 7 to be partially inserted.

The quadrant power tilt limits set forth in Specification 3.5.2.4 have been established to prevent the linear heat rate peaking increase associated with a positive quadrant power tilt during normal power operation from exceeding
7.50% for Unit 1. The limits shown in Specification 3.5.2.4
7.50% for Unit 2
7.50% for Unit 3
are measurement system independent. The actual operating limits, with the appropriate allowance for observability and instrumentation errors, for each measurement system are defined in the station operating procedures.

The quadrant tilt and axial imbalance monitoring in Specification 3.5.2.4 and 3.5.2.7, respectively, normally will be performed in the process computer. The two-hour frequency for monitoring these quantities will provide adequate surveillance when the computer is out of service.

Allowance is provided for withdrawal limits and reactor power imbalance limits to be exceeded for a period of two hours without specification violation. Acceptable rod positions and imbalance must be achieved within the two-hour time period or appropriate action such as a reduction of power taken.

Technical Specification 3.5.2.6 provides the ability to prevent excessive power peaking by transient xenon at rated power.

Operating restrictions resulting from xenon transients and power maneuvers are inherently included in the limits provided in the Core Operational Limits Report.

6.6.1.6 Core Operational Limits Report

A report providing the following core operational limits shall be provided to the Director, ONRR, Attention: Project Director, PWR Project Directorate No. 6 with a copy to the Regional Administrator of the Regional office of the NRC at least 60 days prior to each cycle initial criticality unless otherwise approved by the Commission by letter:

1. Regulating Control Rod Insertion/Withdrawal Limits
2. Axial Power Shaping Control Rod Insertion/Withdrawal Limits
3. Power Level Cutoff
4. Power/Power Imbalance Operational Limits

In addition, in the event that the limits should change requiring a new submittal or amended submittal of the Core Operational Limits Report, it shall be submitted at least 60 days prior to the date the limit would become effective unless otherwise approved by the Commission by letter. Any information needed to support the Core Operational Limits Report will be requested from the NRC and need not be included in the report.

The Core Operational Limits Report is valid for a nominal design cycle length, as defined for the appropriate unit and cycle. Operation beyond the nominal design cycle length is permitted provided that an evaluation is performed to verify that the operational limit curves are valid for extended operation. If the operational limit curves are not valid for the extended period of the operation, appropriate limits will be established and the Core Operational Limits Report will be revised.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

ATTACHMENT 2

TECHNICAL JUSTIFICATION

Technical Justification

The proposed revision to the Oconee Technical Specifications and creation of the cycle specific Core Operational Limits Report is strictly an administrative mechanism to reduce the volume of licensing material which is required to be prepared by Duke and subsequently reviewed and approved by the NRC. The preparation of the power-imbalance limits and control rod position limits will continue to be performed by NRC approved methods. Requirements for plant operation within the above limits and actions required if limits are exceeded remain unchanged. The NRC shall continue to receive the cycle specific limits defined by the Core Operational Limits Report and shall be given ample time to examine the document and make inquiries if necessary. Therefore, the proposed revisions do not in any way impact the operation or safety of the Oconee Nuclear Station.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

ATTACHMENT 3

NO SIGNIFICANT HAZARDS CONSIDERATION

No Significant Hazards Consideration Evaluation
for Oconee Unit 2, Cycle 9 Reload

Duke Power has determined that the present amendment request poses no significant hazards as defined by NRC regulations in 10 CFR 50.92. This ensures that operation of the facility in accordance with the proposed amendment would not:

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

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

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

The commission has provided guidelines pertaining to the application of the three standards by listing specific examples in 48 FR 14870. Example (iii) of the types of amendments not likely to involve significant hazards considerations expressly applies inasmuch as the proposed amendment involves a nuclear power reactor core reload.

Example (iii) of amendments <u>not</u> likely to involve a significant hazards consideration concerns a core reload, assuming that:

(1) no fuel assemblies significantly different from those found previously acceptable e to the NRC or a previous core at the facility in question are involved,

(2) no significant changes are made to the acceptance criteria for the technical specifications,

(3) the analytical methods used to demonstrate conformance with the technical specifications and regulations are not significantly changed, and

(4) the NRC has previously found such methods acceptable.

The Mark BZ fuel assembly, 60 of which comprise Batch 11, is an improved version of the Mark B fuel assembly, in that the six Inconel Intermediate Spacer grids are replaced with Zircaloy grids. The Mark BZ spacer grids have the same functional design a the Mark B grids, but are slightly different dimensionally to accommodate Zircaloy's material properties. The other fuel assembly components such as the fuel rods, end grid, end fittings, and guide tubes are the same for both designs. The interfaces with the control rods and fuel handling equipment are unchanged, ensuring compatibility with the present reactor site operational procedures. The Mark BZ fuel assembly has been analyzed to ensure conformance with the standard review plan acceptance criteria. Therefore, on the whole, the Mark BZ fuel assembly is a minor design change from the Mark B -- with the Zircaloy spacer grids and related changes being the principal differences. The use of the Mark BZ fuel assembly in the Oconee Unit 1, Cycle 9 and Oconee 2 Cycle 8 core reload was accepted by the NRC via the Staff's approval of the Unit 1, Cycle 9 amendment request dated November 23, 1984 and the Unit 2, Cycle 8 amendment request dated April 18, 1985.

As was used in Cycle 8, gray (less-absorbing) axial power shaping rods (APSR's) are to be utilized. The staff approved the use of gray APSRs in the Oconee Unit 1, Cycle 9 and Oconee Unit 2, Cycle 8 reload amendment request.

The present reload involves no significant changes to the acceptance criteria for the Technical Specifications. Revisions of the Technical Specifications required for Cycle 9 operation were made in accordance with methods and procedures found acceptable in connection with previous reloads. The final acceptance criteria of the ECCS limits will not be exceeded, and thermal design criteria will be satisfied.

The Oconee Unit 2, Cycle 9 Reload Report (Attachment 4) justifies the operation of the ninth cycle at the rated core power of 2568 MWt. Included are the required analyses as outlined in the USNRC document "Guidance for Proposed License Amendments Relating to Refueling", June 1985. The Reload Report employs analytical techniques and design bases established in reports submitted for previous reloads which were accepted by USNRC and its predecessor. These techniques are described in the Reload Report references.

Example (vi) of 48 FR 14870 is applicable to the deletion of the rod position limits and operation imbalance envelope curves. This example involves a change which either may result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria.

The removal of Specific Cycle dependent figures from the Technical Specification and the use of the Core Operational Limits Report has no impact upon plant operation or safety. The Technical Specification shall continue to require operation within rod position and operational imbalance limits provided by the Core Operational Limits Report. Appropriate actions to be taken if limits are violated shall also remain in the Technical Specifications.

The development of the rod position and operational imbalance limits shall continue to be performed by methods reviewed and approved by the NRC. The NRC shall be provided a copy of the Core Operational Limits Report at least 60 days prior to initial criticality of the cycle. This should be sufficient time to allow the NRC to review the document and to make inquiries if necessary.

Based on the above considerations, Duke contends that the removal of these particular curves (rod position limits and operational imbalance limits) will not involve a significant increase in the probability or consequences of accidents previously considered, nor create the possibility of a new or different kind of accident and will not involve a significant reduction in a safety margin. Therefore, Duke concludes that there is no significant hazards considerations involved with the deletion of these curves.

With supporting reference to previously performed analyses, the following evaluation measures aspects of the Unit 2, Cycle 9 reload against the Part 50.92 (c) requirements to demonstrate that all three standards are satisfied.

First Standard

(Amendment would not) involve a significant increase in the probability or consequences of an accident previously evaluated.

Each accident analysis addressed in the Oconee Final Safety Analysis Report (FSAR) has been examined with respect to the removal of the rod position and Operational Imbalance envelop curves from the Technical Specification and the incorporation of these curves within the Core Operational Limits Report (COLR). The probability of any Design Based Accident (DBA) is not affected by this change. Where these curves are actually located has no influence or impact on the probability of a DBA occurring. The location of these curves are not considered to be an initiator or part of th sequence of events of a DBA.

In the same sense, the consequences of a DBA are not affected by this change. Where these curves are actually located has no influence or impact on the consequences of a DBA, since this proposed amendment still requires exactly the same actions to be taken when or if limits are exceeded as is required by current Technical Specifications. The location of these curves are not considered to be a contributor to the events of a DBA.

Additionally, each accident analysis addressed in the Oconee Final Safety Analysis Report (FSAR) has been examined with respect to changes in Cycle 8 parameters to determine the effect of the Cycle 9 reload and to ensure that thermal performance during hypothetical transients is not degraded. The transient evaluation of Cycle 9 is considered to be bound by previously accepted analyses. Section 7 of the Reload Report addresses "Accident and Transient Analysis" for this core reload. This analysis ensures that the proposed reload will not involve a significant increase in the probability or consequences of an accident previously evaluated.

Second Standard

(Amendment would not) create the possibility of a new or different kind of accident form any accident previously evaluated.

As stated in the discussion of the First Standard, the actual location of these curves has no influence, impact nor does it contribute in any way to the probability or consequences of an accident. The actual location of these curves does not in any way create the possibility of a new or different kind of accident from any accident previously evaluated. Principally, due to the fact that the preparation of the power-imbalance limits and control rod position limits are performed by NRC approved methods. In addition, the Technical Specification will continue to require operation within rod position and operational imbalance limits and appropriate actions to be taken when or if limits are exceeded.

The analyses performed in support of this reload are in accordance with the USNRC document "Guidance for Proposed License Amendments Relating to Refueling", June 1975. The conclusion of the overall analysis is that the proposed reload does not in any way create the possibility of a new or different kind of accident from any accident previously evaluated.

Third Standard

(Amendment would not) involve a significant reduction in a margin of safety.

The margin of safety is not affected by the use of the COLR and removal of power-imbalance and control rod position unit curves from the Technical Specification. The margin of safety presently provided by current Technical Specification remains unchanged.

That is, the proposed amendment still requires operation within rod position and operational imbalance limits, as provided by COLR, and appropriate actions to be taken when or if limits are violated remain unchanged.

The development of the limits and the preparation of the curves will continue to conform to those methods described in NRC approved documentation. In addition, there exist provisions within the Technical Specification requiring that the information provided by the COLR be submitted to the NRC at least 60 days prior to the implementation of the limits provided by the COLR, unless otherwise approved by the commission. This shall assure ample time for the NRC staff to review the document and, if necessary, to make any inquiries.

The issue of margin of safety for a reload modification involves the following areas:

1. Fuel System Design considerations,

2. Nuclear Design considerations, and

3. Thermal-Hydraulic Design considerations.

Sections 4, 5, and 6 of the Oconee Unit 2, Cycle 9 Reload Report addresses the above areas, respectively. The value limits and margins discussed in these areas are well within the allowable limits and requirements, and reflect no significant reductions to any margins of safety. One can conclude from the examination of these sections, and the Cycle 9 core thermal and kinetic properties (with respect to previous cycle values), that this core reload will not significantly reduce the ability of Oconee Unit 2 to operate safely during Cycle 9.

The above evaluation, with its accompanying references, shows that the three Part 50.92 (c) standards are satisfied. In summary, Duke has determined and submits that the removal of rod position limits and operational imbalance limits curves, as well as the proposed reload described herein does not represent any significant hazards.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

ATTACHMENT 4

CORE OPERATIONAL LIMITS REPORT

OCONEE 1, CYCLE 10

OCONEE 2, CYCLE 9

OCONEE 3, CYCLE 9

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

OCONEE UNIT 1, CYCLE 10

CORE OPERATIONAL LIMITS REPORT

SRC-OS1-ND-85-007-0

OCTOBER 1985

8512060280

Oconee Unit 1, Cycle 10

Core Operational Limits Report

Per the requirements of Technical Specification 6.6.1.4, this Core Operational Limits Report has been prepared to provide the necessary limitations on reactor power, imbalance, and control rod position for operation of Oconee 1, Cycle 10. Curves presented in this report are based upon a cycle length of 400 EFPD. If the cycle length is expected to exceed 410 EFPD, an evaluation shall be performed in accordance with Technical Specification 3.5.2.4 in order to verify the continued validity of the curves presented in this report. Any required changes to the operational limit curves due to extended operation or other causes shall be implemented in accordance with Technical Specification 6.6.1.1.

Figure 1 provides the operational limits upon power and power imbalance. The power-imbalance envelope is determined by the most limiting power distribution criteria of either the loss of coolant accident (LOCA) analyses or the loss of flow accident (LOFA) analyses. Requirements on surveillance and actions required to respond to plant conditions outside of the acceptable power-imbalance envelope are provided in Technical Specification 3.5.2.

Figures 2 thru 4 provide the control rod position limits for operation with 4, 3, and 2 reactor coolant pumps in operation. The rod insertion limits ensure the shutdown margin requirements of Technical Specification 3.1.11 are satisfied and therefore provides for achieving hot shutdown by reactor trip at any time (assuming the highest worth control rod remains in the full out position). Rod position limits also ensure that power peaking criteria associated with LOCA and LOFA analyses are not exceeded. In addition, the limits preclude the insertion of rod groups which could result in any single rod worth greater than the safety analysis assumption for the rod ejection transient. Requirements on surveillance and actions required to respond to plant conditions outside the acceptable restricted operation regions are provided in Technical Specification 3.5.2. The power-level-cutoff values associated with the Technical Specification 3.5.2.6, Xenon Reactivity, are provided by the rod position limit curves of Figure 2.

Due to the low worth of Inconel-600 axial power shaping rods (APSR's) no position limits on the APSR's are required for this cycle.

Table 1 provides the bounding values of allowable LOCA peak linear heat rates used to determine the operational power-imbalance envelope and control rod position limits.

Oconee Unit 1, Cycle 10

Table 1

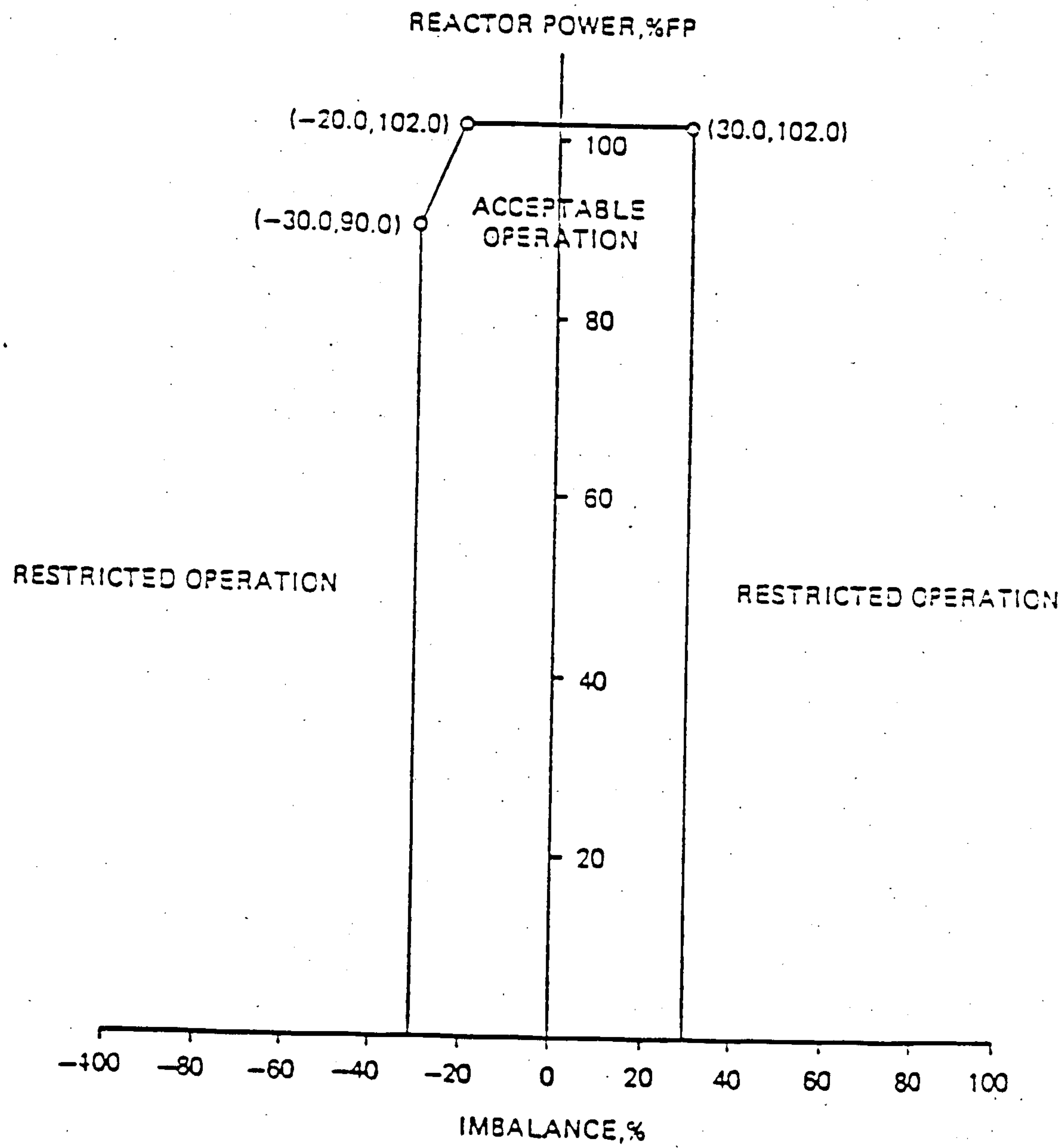
Limiting LOCA Linear Heat Rates

Elevation ft	LOCA LHR Limits, kW/ft 0 - 1000 MWD/MTU	1000 - 2600 MWD/MTU	After 2600 MWD/MTU
2	13.5	15.0	15.5
4	15.1	15.6	15.5
6	16.5	18.0	18.0
8	17.0	17.0	17.0
10	16.0	16.0	15.0

Oconee Unit 1, Cycle 10

FIGURE 1

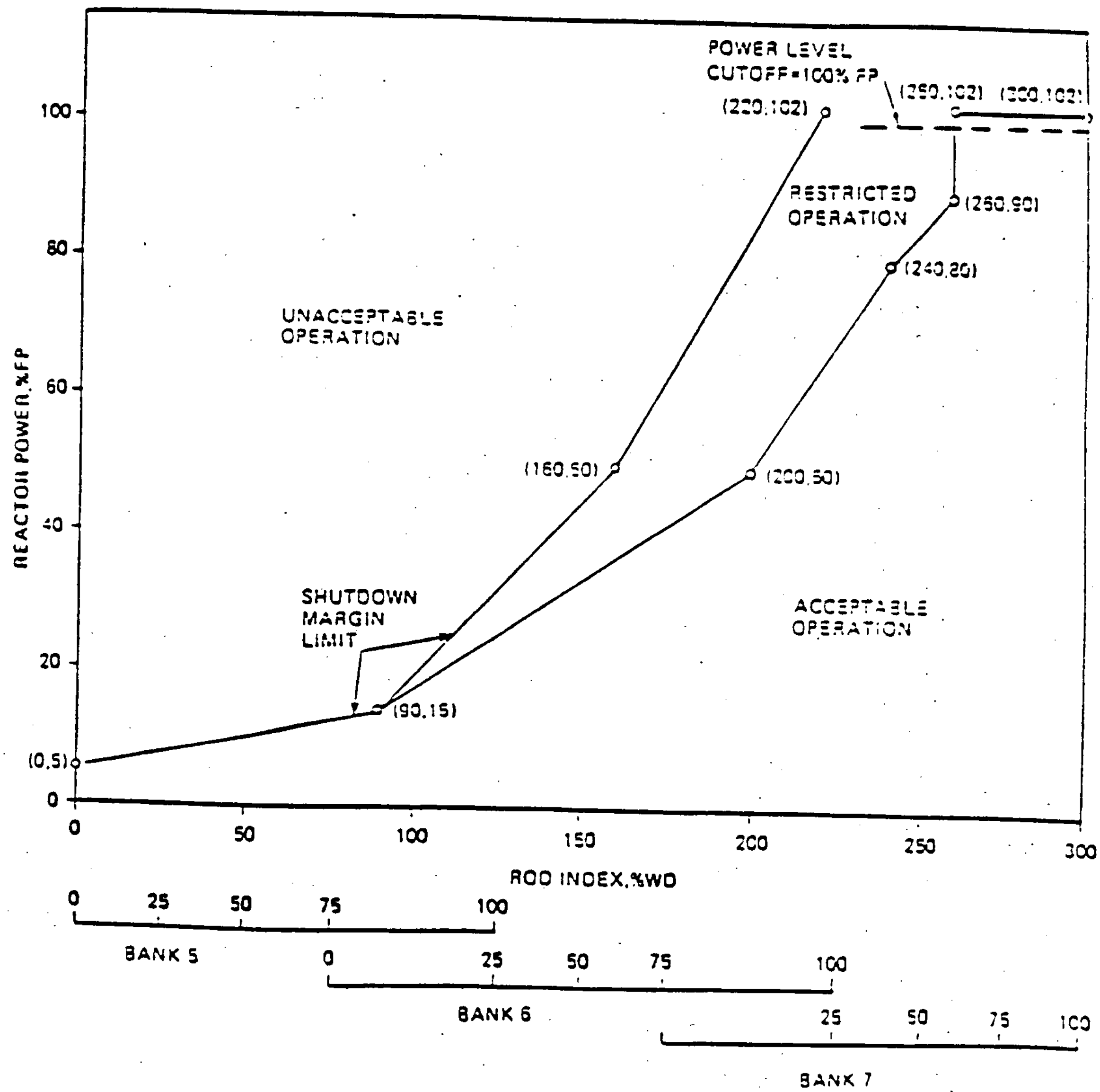
OPERATIONAL POWER-IMBALANCE LIMITS, 0 EFPD TO EOC



Oconee Unit 1, Cycle 10

FIGURE 2

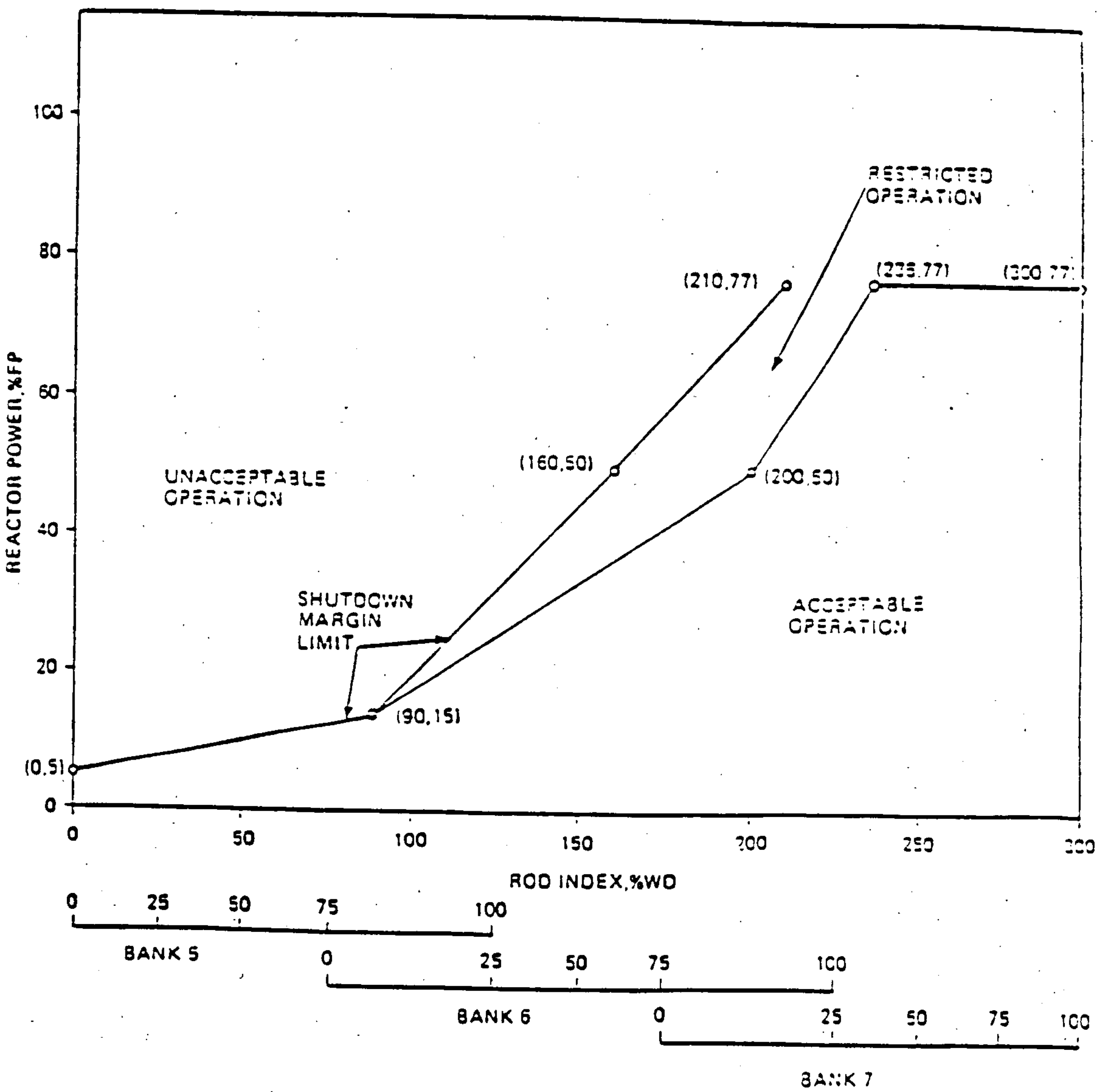
CONTROL ROD POSITION LIMITS, 4 PUMPS, 0 EFPD TO EOC



Oconee Unit 1, Cycle 10

FIGURE 3

CONTROL ROD POSITION LIMITS, 3 PUMPS, 0 EFPD TO EOC



Oconee Unit 1, Cycle 10

FIGURE 4

CONTROL ROD POSITION LIMITS, 2 PUMPS, 0 EFPD TO EOC