Duke Power Company
Oconee Nuclear Station

Attachment 1

Proposed Technical Specification Revision
Oconee 3, Cycle 10

Remove Page	Insert Page
3.5-26	3.5-26
3.5-8	3.5-8
3.5-10	3.5-10
3.5-12	3.5-12
3.5-24	3.5-24
3.5-26	3.5-26

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 100% full power by 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.

3.5.2.6 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 envelope defined by Figures 3.5.2-10 (Unit 1). If the imbalance
3.5.2-11 (Unit 2)
3.5.2-12 (Unit 3)

is not within the envelope defined by these figures, 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.

3.5.2.7 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.

3.5.2.8 The operational limit curves of Technical Specifications 3.5.2.5.c and 3.5.2.6 are valid for a nominal design cycle length, as defined in the Safety Evaluation Report 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 <u>not</u> valid for the extended period of the operation, appropriate limits will be established and the Technical Specification curves will be modified as required.

The rod position limits 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% Δk/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% Δk/k is allowed by the rod position limits at hot zero power. A single inserted control rod worth of 1.0% δk/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% Δk/k ejected rod worth at rated power.

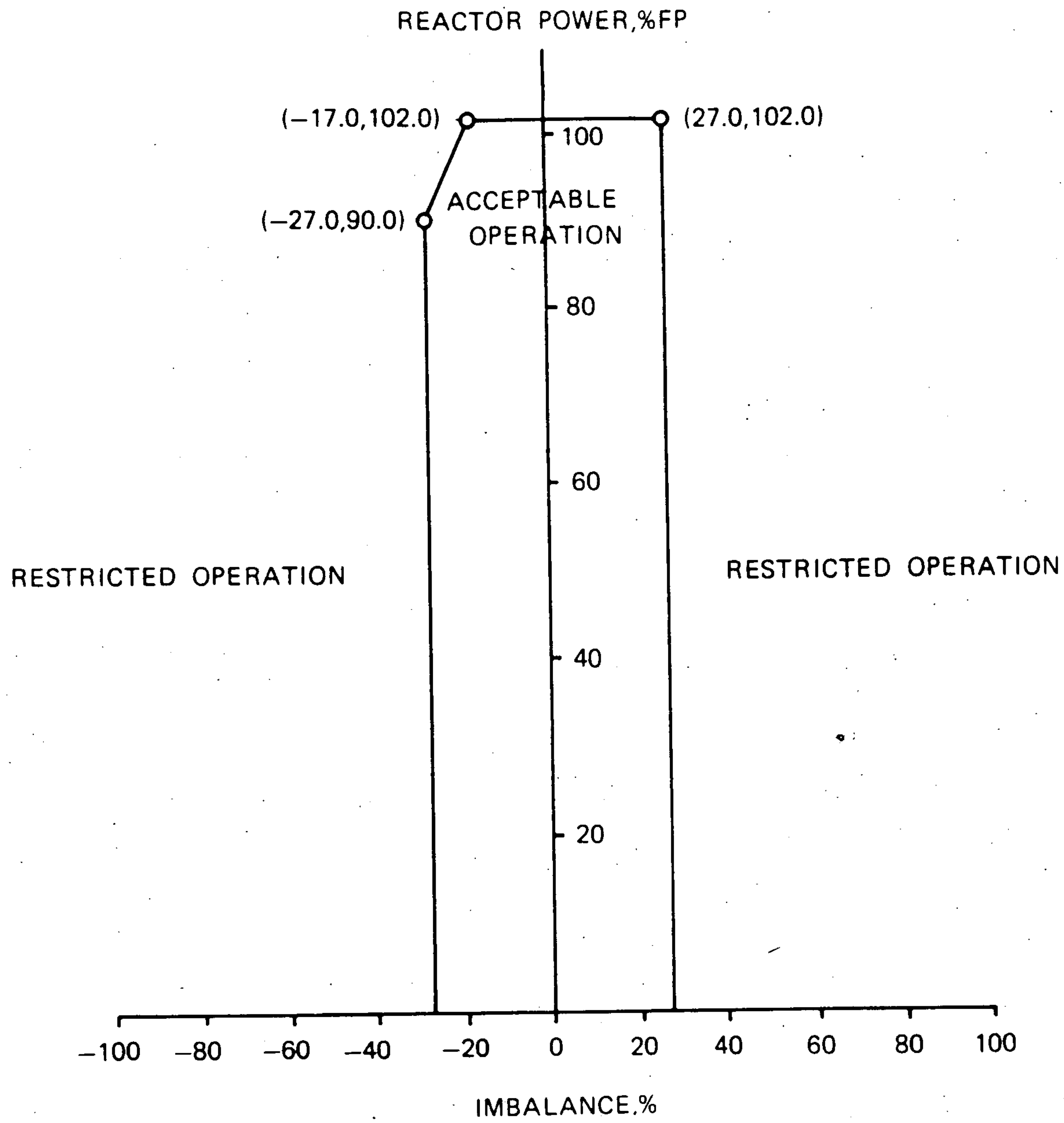
Control rod groups are withdrawn in sequence beginning with Group 1. Groups 5, 6, and 7 are overlapped 25 percent. The normal position at power is for Group 7 to be partially inserted.

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.

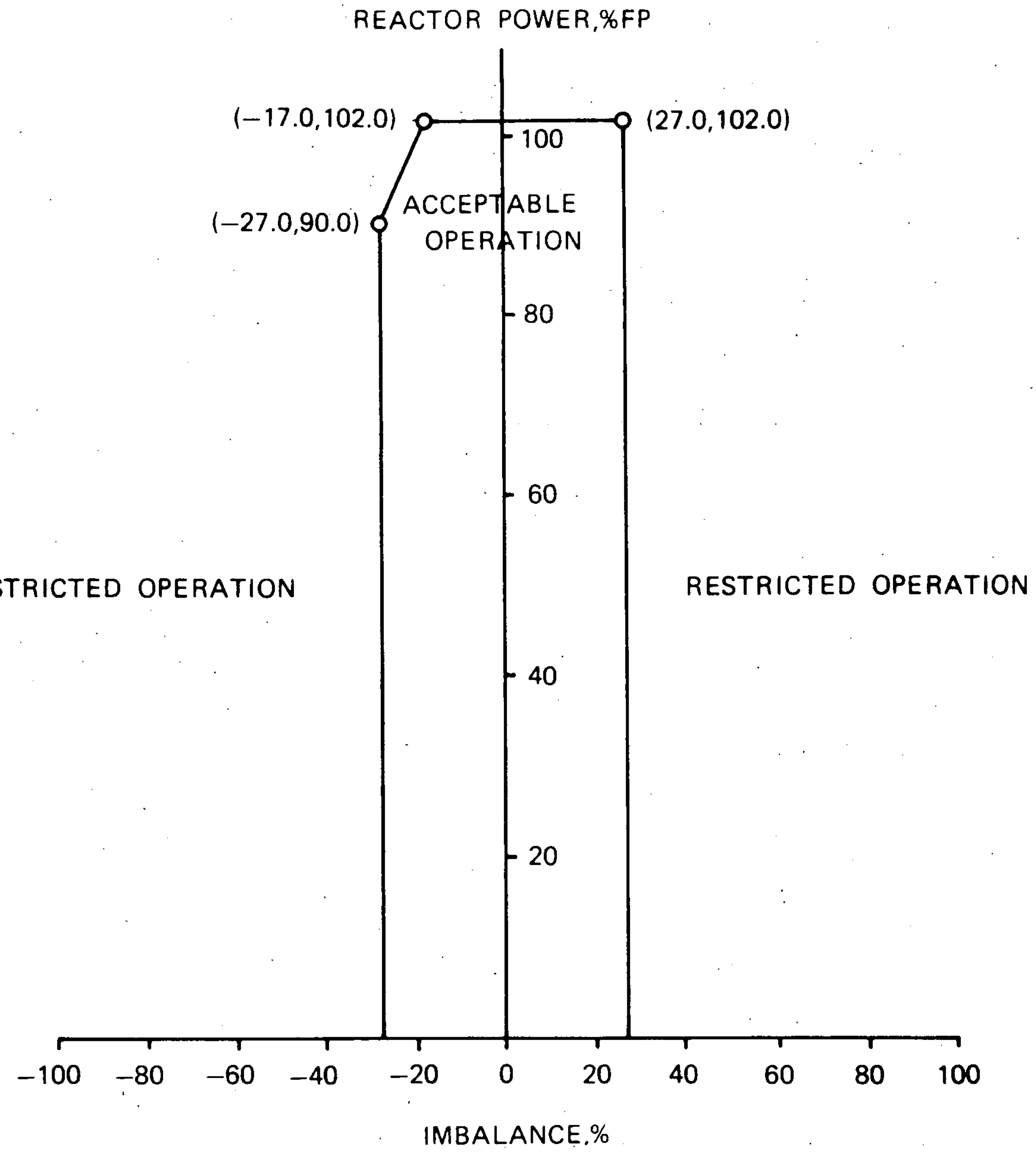
The quadrant tilt and axial imbalance monitoring in Specification 3.5.2.4 and 3.5.2.6, respectively, normally will be performed in the process computer. The two-hour frequency for monitoring these quantities will provide adequate surveillance when the computer is out of service.

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.

Operating restrictions resulting from transient xenon power peaking are implicitly included in the limits of Section 3.5.2.5 (control rod positions) and 3.5.2.6 (reactor power imbalance). Since these limits are set during the cycle-specific maneuvering analysis to prevent excessive power peaking by transient xenon at all power levels, there is no need for any hold at a power level cutoff below 100% FP.



OPERATIONAL POWER
IMBALANCE ENVELOP
FROM ... EFPD TO EOC UNIT 1
OC... ...LEAR STATION
...URE 3.5.2-10



OPERATIONAL POWER
IMBALANCE ENVELOP
FROM 0 EFPD TO [illegible] UNIT 3
OCONEE NUC[illegible]ATION
Figure 3.[illegible]-12

Duke Power Company
Oconee Nuclear Station

Attachment 2

No Significant Hazards Consideration

No Significant Hazards Consideration Evaluation
for Oconee Unit 3, Cycle 10 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 from 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 not likely to involve a significant hazards consideration concerns a core reload, assuming that:

(1) no fuel assemblies significantly different from those found previously acceptable 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 Batch 12 Mark B5Z fuel assemblies are a Mark B5 design with zircaloy intermediate spacer grids. The B5Z design has been previously demonstrated in the Oconee 3, Cycle 9 reload. There are no new features not previously demonstrated. Additionally, the Batch 12 fuel rods have a slightly reduced prepressurization level to provide a small increase in fuel rod burnup. This level of prepressurization has also been previously implemented (03C9). The use of the Mark B5Z fuel assembly in the Oconee Unit 3, Cycle 9 core reload was accepted by the NRC via the Staff's approval of the Unit 3, Cycle 9 amendment request dated September 19, 1985.

As was used in Cycle 9, 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 10 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 3, Cycle 10 Reload Report (Attachment 3) justifies the operation of the tenth 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 (ii) of the types of amendments not likely to involve significant hazards considerations applies in that the proposed Figure 3.5.2-10 provides a more conservative operational power imbalance envelope for Unit 1. Example (i) applies in that changes to T.S. 3.5.2.4.b.2, 3.5.2.6, 3.5.2.7, 3.5.2.8, and 3.5.2.9 are administrative in nature.

With supporting reference to previously performed analyses, the following evaluation measures aspects of this amendment request 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 changes in Cycle 9 parameters to determine the effect of the Cycle 10 reload and to ensure that thermal performance during hypothetical transients is not degraded. The transient evaluation of Cycle 10 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.

Changes to the Operational Power Imbalance Envelope for Unit 1 (Figure 3.5.2-10) provide a more conservative envelope for Unit 1. As such, this change will not involve a significant increase in the probability or consequences of accidents previously evaluated.

The update of T.S. 3.5.2.4.b.2 reflects the fact that power level cutoffs (other than 100%) are no longer applicable to Oconee. Thus, each accident analysis addressed in the Oconee FSAR is not affected by this change.

Each accident analysis addressed in the Oconee FSAR has been examined with respect to the deletion of T.S. 3.5.2.6 (Xenon Reactivity). Operating restrictions due to transient xenon power peaking are implicitly included in the limits of T.S. 3.5.2.5 (Control Rod Positions) and new T.S. 3.5.2.6 (Reactor Power Imbalance), as a result deletion of the present T.S. 3.5.2.6 is purely administrative in nature, and thus will not involve a significant increase in the probability or consequences of previously evaluated accidents. Changes to the bases of T.S. 3.5 note the justification for deletion of this specification.

T.S. 3.5.2.7, 3.5.2.8, and 3.5.2.9 have been renumbered to reflect the deletion of T.S. 3.5.2.6. This change is purely administrative in nature, and as such will not involve a significant increase in the probability or consequences of previously evaluated accidents.

Second Standard

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

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.

Analysis of the changes to the operational Power Imbalance Envelope for Unit 1 (Figure 3.5.2-10) has indicated that the proposed envelope is more conservative than the envelope presently included in the Technical Specifications. As a result, this change will not create the possibility of a new or different kind of accident from accidents previously evaluated.

A new or different kind of accident from any accident previously evaluated is not possible due to the update of T.S. 3.5.2.4.b.2 since power level cutoffs are no longer applicable to Oconee.

The deletion of T.S. 3.5.2.6 (Xenon Reactivity) is purely administrative in nature in that operating restrictions due to transient xenon power peaking are implicitly included in the limits of new T.S. 3.5.2.6 (Reactor Power Imbalance) and T.S. 3.5.2.5 (Control Rod Positions). As such, since operating restrictions are presently and implicitly included in other Technical Specifications, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

T.S. 3.5.2.7, 3.5.2.8, and 3.5.2.9 have been renumbered to reflect the deletion of T.S. 3.5.2.6 (Xenon Reactivity). This change is administrative in nature and, thus will not 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 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 3, Cycle 10 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 10 core thermal and kinetic properties (with respect to previous cycle values), that this core reload will not reduce the ability of Oconee Unit 3 to operate safely during Cycle 10.

Changes to the Operational Power Imbalance Envelope for Unit 1 (Figure 3.5.2-10) provide a more conservative envelope. As such, the margin of safety due to this change will be increased.

As power level cutoffs are no longer applicable to Oconee, there will be no reduction to any margin of safety due to the change to T.S. 3.5.2.4.b.2.

Deletion of T.S. 3.5.2.6 (Xenon Reactivity) has been determined to be administrative in nature. As such, no margin of safety will be decreased due to this change.

Renumbering of T.S. 3.5.2.7, 3.5.2.8, and 3.5.2.9 will have no impact on any margin of safety, since this change is purely administrative.

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 proposed reload described herein does not represent any significant hazards.

Duke Power Company
Oconee Nuclear Station

Attachment 3

Oconee Unit 3, Cycle 10 Reload Report

OCONEE UNIT 3, CYCLE 10

\- Reload Report -

DPC - RD - 2008

December 1986

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Design Engineering Department
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CONTENTS

List of Tables

List of Figures

1. INTRODUCTION AND SUMMARY

This report justifies the operation of the tenth cycle of Oconee Nuclear Station, Unit 3, 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 1975.

To support Cycle 10 operation of Oconee Unit 3, this report employs analytical techniques and design bases established in reports that were previously submitted and accepted by the USNRC and its predecessor (see references).

A brief summary of Cycle 9 and 10 reactor parameters related to power capability is included in Section 5 of this report. All of the accidents analyzed in the FSAR[1] have been reviewed for Cycle 10 operation. In those cases where Cycle 10 characteristics were conservative compared to those analyzed for previous cycles, no new accident analyses were performed.

The Technical Specifications have been reviewed, and the modifications required for Cycle 10 operation are justified in this report.

Based on the analyses performed, which take into account the postulated effects of fuel densification and the Final Acceptance Criteria for Emergency Core Cooling Systems, it has been concluded that Oconee Unit 3 can be operated safely for Cycle 10 at the rated power level of 2568 MWt.

2. OPERATING HISTORY

The referenced fuel cycle for the nuclear and thermal-hydraulic analyses of Oconee Unit 3, Cycle 10, is the currently operating Cycle 9. Cycle 9 achieved initial criticality on October 6, 1985 and power escalation commenced on October 7, 1985. The fuel cycle design length for Cycle 10 - 400 EFPD - is based on a Cycle 9 length of 421 EFPD. No operating anomalies occurred during previous cycle operations that would adversely affect fuel performance in Cycle 10.

Cycle 10 will operate in a feed-and-bleed mode for its entire design length, as did Cycle 9.

3. GENERAL DESCRIPTION

The Oconee Unit 3 reactor core and fuel design basis are described in detail in Chapter 4, of the FSAR.[1] The Cycle 10 core consists of 177 fuel assemblies, each of which is a 15 by 15 array containing 208 fuel rods, 16 control rod guide tubes, and one incore instrument guide tube. The fuel consists of dished-end, cylindrical pellets of uranium dioxide clad in cold-worked Zircaloy-4. The fuel assemblies in all batches have an average nominal fuel loading of 463.6 kg uranium. The undensified nominal active fuel lengths, theoretical densities, fuel and fuel rod dimensions, and other related fuel parameters are given in Table 4-1.

Figure 3-1 is the core loading diagram for Oconee 3, Cycle 10. Nineteen of the Batch 10 assemblies will be discharged at the end of Cycle 9 along with Batch 9B. The remaining 49 Batch 10 assemblies, designated "10B," and the fresh Batch 12 FAs - with initial enrichments of 3.28 and 3.22 wt % ^{235}U, respectively - will be loaded into the central portion of the core. Batch 11, with an initial enrichment of 3.22 wt % ^{235}U, will occupy primarily the core periphery. Figure 3-2 is a quarter-core map showing the assembly burnup and enrichment distribution at the beginning of Cycle 10.

Cycle 10 will operate in a rods-out, feed-and-bleed mode. Core reactivity control is supplied mainly by soluble boron and supplemented by 61 full-length Ag-In-Cd control rods and 60 burnable poison rod assemblies (BPRAs). In addition to the full-length control rods, eight Inconel gray axial power shaping rods (APSRs) are provided for additional control of axial power distribution. The Cycle 10 locations of the 69 control rods and the group designations are indicated in Figure 3-3. The Cycle 10 locations and enrichments of the BPRAs are shown in Figure 3-4.

FIGURE 3.1. CORE LOADING DIAGRAM FOR OCONEE 3, CYCLE 10

X

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						L15 10B	P12 10B	O08 11	P04 10B	L01 10B					
B				O06 11	G14 11	B11 11	12	L11 11	12	B05 11	G02 11	O10 11			
C			N05 11	12	N03 11	12	N07 11	K08 11	N09 11	12	N13 11	12	E04 11		
D		F13 11	12	F11 11	12	F07 10B	12	M08 11	12	F09 10B	12	M10 11	12	F03 11	
E		P07 11	C12 11	12	B06 10B	12	A07 10B	B10 10B	A09 10B	12	F14 10B	12	C04 11	P09 11	
F	R10 10B	M02 11	12	G06 10B	12	N11 11	12	A08 10B	12	M04 11	12	G10 10B	12	M14 11	R06 10B
G	N14 10B	12	G12 11	12	G01 10B	12	O13 10B	12	O03 10B	12	G15 10B	12	G04 11	12	N02 10B
H	H13 11	E10 11	H09 11	H11 11	F02 10B	H01 10B	12	H14 10B	12	H15 10B	L14 10B	H05 11	H07 11	M06 11	H03 11
K	D14 10B	12	K12 11	12	K01 10B	12	C13 10B	12	C03 10B	12	K15 10B	12	K04 11	12	D02 10B
L	A10 10B	E02 11	12	K06 10B	12	E12 11	12	R08 10B	12	D05 11	12	K10 10B	12	E14 11	A06 10B
M		B07 11	O12 11	12	L02 10B	12	R07 10B	P06 10B	R09 10B	12	P10 10B	12	O04 11	B09 11	
N		L13 11	12	E06 11	12	L07 10B	12	E08 11	12	L09 10B	12	L05 11	12	L03 11	
O			M12 11	12	D03 11	12	D07 11	G08 11	D09 11	12	D13 11	12	D11 11		
P				C06 11	K14 11	P11 11	12	F05 11	12	P05 11	K02 11	C10 11			
R						F15 10B	B12 10B	C08 11	B04 10B	F01 10B					

Y

Z

XX	PREVIOUS CYCLE LOCATION
X	BATCH NO.

FIGURE 3.2 ENRICHMENT & BURNUP FOR OCONEE 3, CYCLE 10

	8	9	10	11	12	13	14	15
H	3.28 29011	3.22 0	3.28 22580	3.28 26064	3.22 16204	3.22 14330	3.22 16193	3.22 16416
K	3.22 0	3.28 23303	3.22 0	3.28 22824	3.22 0	3.22 16705	3.22 0	3.28 17842
L	3.28 22597	3.22 0	3.22 16538	3.22 0	3.28 26526	3.22 0	3.22 11454	3.28 21280
M	3.28 26064	3.28 22824	3.22 0	3.28 26070	3.22 0	3.22 13440	3.22 14199	
N	3.22 16200	3.22 0	3.28 26508	3.22 0	3.22 16192	3.22 0	3.22 16014	
O	3.22 14327	3.22 16704	3.22 0	3.22 13449	3.22 0	3.22 16522		
P	3.22 16193	3.22 0	3.22 11449	3.22 14199	3.22 16009			
R	3.22 16420	3.28 17846	3.28 21257					

S_L

X.XX	INITIAL ENRICHMENT, wt% ^{235}U
XXXXX	BOC BURNUP, MWd/mtU

FIGURE 3.3 CONTROL ROD LOCATIONS FOR OCONEE 3, CYCLE 10

X

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

Y

Z

X GROUP NO.

GROUP	NO. OF RODS	FUNCTION
1	8	SAFETY
2	8	SAFETY
3	8	SAFETY
4	9	SAFETY
5	12	CONTROL
6	8	CONTROL
7	8	CONTROL
8	8	APSRs
TOTAL	69	

FIGURE 3.4 BPRA ENRICHMENT & DISTRIBUTION FOR OCONEE 3, CYCLE 10

	8	9	10	11	12	13	14	15
H		1.1						
K	1.1		1.1		1.1		0.2	
L		1.1		1.1		0.8		
M			1.1		1.1			
N		1.1		1.1		0.2		
O			0.8		0.2			
P		0.2						
R								

X.X	BPRA CONCENTRATION, wt % B_4C IN Al_2O_3

4. FUEL SYSTEM DESIGN

4.1 Fuel Assembly Mechanical Design

The types of fuel assemblies and pertinent fuel design parameters for Oconee 3 Cycle 10, are listed in Table 4-1. All fuel assemblies are identical in concept and are mechanically interchangeable. Two regenerative neutron sources will be used in Mark B5 fuel assemblies.

The Batch 12 Mark B5Z fuel assemblies are a Mark B5 design with zircaloy intermediate spacer grids. The B5Z design has been previously demonstrated in the Oconee 3 Cycle 9 reload (Reference 5). There are no new features not previously demonstrated. Additionally, the Batch 12 fuel rods have a slightly reduced prepressurization level to provide a small increase in fuel rod burnup. This level of prepressurization has also been previously implemented[5]. All 60 BPRAs will be inserted into Batch 12 fuel assemblies.

Other results presented in the FSAR[1] fuel assembly mechanical discussions and in previous reload reports, are applicable to the reload fuel assemblies. Duke has performed generic mechanical analyses, as described below, which envelope the Cycle 10 design. All methods are consistent with the approved methodologies of Reference 10 except where specifically stated.

4.2 Fuel Rod Design

The mechanical evaluation of the fuel rod is discussed below.

4.2.1 Cladding Collapse

The fuel of Batch 10B is more limiting than other batches due to its longer previous incore exposure time. The Batch 10B assembly power histories were assessed against Duke's generic creep collapse analysis which is based on the CROV computer code and procedures described in topical report BAW-10084, Rev. 2[2]. The TACO2[6] code was used to calculate internal pin pressure and clad temperatures used as input to CROV. The collapse time for the most limiting

assembly was conservatively determined to be 32,900 EFPH, which is greater than the maximum projected residence time of Cycle 10 fuel (Table 4-1).

4.2.2 Cladding Stress

As described in Reference 10, Duke has performed a generic and conservative fuel rod cladding stress analysis in accordance with the guidelines set forth in Section III, Division 1 - Subsection NB, of the ASME Boiler and Pressure Vessel Code. All methods are consistent with Reference 10 except that the static stress analysis uses design stress intensity limits on mechanical properties based on the requirements of ASME Code Article III-2000. Compliance with ASME Code criteria verify the structural integrity of the cladding throughout the most limiting design conditions.

The following conservatisms exist in the generic cladding stress calculation:

- high external cladding pressure (110% of design system pressure)
- low internal pressure (HZP - min. specified pre-pressure)
- maximum possible radial temperature gradient through clad (fuel melt conditions)
- conservative cladding dimensions with regards to stress

4.2.3 Cladding Strain

Duke has performed a cladding strain calculation using TACO2[6] in accordance with the approved methodology [10]. This analysis demonstrated that the uniform, circumferential strain of the cladding was within 1.0%.

4.3 Thermal Design

All fuel in the Cycle 10 core is thermally similar. The fresh Batch 12 fuel inserted for Cycle 10 operation introduces no significant differences in fuel thermal performance relative to the other fuel remaining in the core. The linear heat rate to melt capability based on centerline fuel melt was determined separately for each batch of fuel using the TACO2[6] computer code. The fuel parameters used to determine the fuel melt limits are shown in Table 4-2.

The input shown includes the following bounding, generic conservatisms:

1. A maximum gap based on as-fabricated pellet and cladding data.
2. Maximum incore densification based on resinter test results.

The burnup dependent linear heat rate (LHR) capability and the average fuel temperature for each batch are shown in Table 4-2.

The maximum assembly average burnup is predicted to be 42,222 MWd/MtU and the maximum fuel rod burnup is predicted to be 43,725 MWd/mtU. Fuel rod internal pressure has been evaluated using TACO2[6] with a conservative pin power history, and the maximum pressure is less than the nominal reactor coolant (RC) system pressure of 2200 psia.

4.4 Material Design

The Batch 12 fuel assemblies are not unique in concept, nor do they utilize different component materials. Therefore, the chemical compatibility of all possible fuel-cladding-coolant-assembly interactions for the Batch 12 fuel assemblies is identical to those of the present fuel.

Table 4-1.

Fuel Design Parameters and Dimensions

	Batch No.		
	10B	11	12
FA type	Mark B5	Mark B5Z	Mark B5Z
No. of FAs	49	68	60
Fuel rod OD, in.	0.430	0.430	0.430
Fuel rod ID, in.	0.377	0.377	0.377
Flex spacers, type	Spring	Spring	Spring
Rigid spacers, type	Zr-4	Zr-4	Zr-4
Undensified active fuel length, in.	141.8	141.8	141.8
Fuel pellet OD (mean spec), in.	0.3686	0.3686	0.3686
Fuel pellet initial density (mean spec), %TD	95.0	95.0	95.0
Initial fuel enrichment, wt % ^{235}U	3.28	3.22	3.22
Est. residence time, EOC 10, EFPH	29,544	19,944	9,840
cladding collapse time, EFPH	>32,900	>27,400	>27,400

Table 4-2. Linear Heat Rate to Melt Analysis

	Batch No.		
	10B	11	12
Nominal initial density, % TD	95.0	95.0	95.0
Nominal initial pellet diameter, in.	0.3686	0.3686	0.3686
Nominal initial clad ID, in.	0.377	0.377	0.377
Nominal initial clad OD, in.	0.430	0.430	0.430
Average linear heat rate @ 100% of 2568 MW, kW/ft	5.74	5.74	5.74
Linear heat rate capability[b] from 0-1000 MWD/MTU, kW/ft	20.15	20.15	20.15
Linear heat rate capability[b] >1000 MWD/MTU, kW/ft	21.20	21.20	21.20
Average fuel temp. @ nominal linear heat rate, °F	1240[a]	1240[a]	1240[a]

(a) Basis: TACO2, 96.5% TD @ 4000 MWD/MTU, nominal pellet and cladding dimensions

(b) These values are utilized as fuel design limits for Cycle 10.

5. NUCLEAR DESIGN

5.1 Physics Characteristics

Table 5-1 compares the core physics parameters of design Cycles 9 and 10; the values for Cycles 9 and 10 were generated by Duke Power Company using methods described in Reference 10. Since the core has not yet reached an equilibrium cycle, differences in core physics parameters are to be expected between the cycles. Figure 5-1 illustrates a representative relative power distribution for the beginning of the tenth cycle at full power with equilibrium xenon and normal rod positions.

The primary reasons for the differences in the physics parameters between Cycles 9 and 10 are the number of feed assemblies and the different shuffle patterns. Differences in ejected and stuck rod worths between cycles are due to changes in the radial flux and burnup distributions. Calculated ejected rod worths and their adherence to criteria are considered at all times in life and at all power levels in the development of the rod position limits presented in Section 8. All safety criteria associated with these rod worths are met. The adequacy of the shutdown margin with Cycle 10 stuck rod worths is demonstrated in Table 5-2. The following conservatisms were applied for the shutdown calculations:

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

Flux redistribution was explicitly accounted for since the shutdown analysis was calculated using a three-dimensional model. The reference fuel cycle shutdown margin is presented in the Oconee 3, Cycle 9 Reload Report.[5]

5.2 Analytical Input

The Cycle 10 incore measurement calculation constants to be used to compute core power distributions were obtained in a similar manner for Cycle 10 as for the reference cycle.

5.3 Changes in Nuclear Design

There are no changes in design methodology between Oconee 3 Cycle 10 and Oconee 3 Cycle 9.

Table 5-1. Oconee 3 Physics Parameters[a]

	Cycle 9[b]	Cycle 10[c]
Cycle length, EFPD	400	400
Cycle burnup, MWd/mtU	12,349	12,327
Average core burnup, EOC, MWd/mtU	23,035	24,490
Initial core loading, mtU	82.1	82.1
Critical boron - BOC (no xenon), ppm		
HZP, Groups 7 and 8 at nominal positions[d]	1577	1589
HFP, Groups 7 and 8 at nominal positions	1395	1330
Critical boron - EOC (equilibrium xenon), ppm		
HZP, Groups 7 and 8 at nominal positions	402	409
HFP, Groups 7 and 8 at nominal positions	64	48
Control rod worths - HFP, BOC, % $\Delta k/k$		
Group 7	1.19	1.02
Group 8 (e)	0.16	0.17
Control rod worths - HFP, EOC, % $\Delta k/k$		
Group 7	1.24	1.11
Group 8 (e)	0.16	0.22
Max ejected rod worth - HZP, % $\Delta k/k$		
BOC, (L10) groups 5-8 inserted	0.69	0.39
EOC, (N12) groups 5-8 inserted	0.51	0.43
Max stuck rod worth - HZP, % $\Delta k/k$		
BOC (N12)	1.63	1.57
EOC (N12)	1.92	1.98
Power deficit, HFP to HZP, % $\Delta k/k$		
BOC	1.77	1.88
EOC	3.04	3.17
Doppler coeff - HFP, 10^{-5} ($\Delta k/k$-°F)		
BOC (equilibrium xenon)	-1.31	-1.39
EOC (equilibrium xenon)	-1.63	-1.73

Table 5-1. (Cont'd)

	Cycle 9[(b)]	Cycle 10[(c)]
Moderator coeff - HFP, 10^{-4} ($\Delta k/k$-°F)		
BOC (equilibrium xenon)	-0.58	-0.70
EOC (equilibrium xenon)	-2.91	-3.27
Boron worth - HFP, ppm/% $\Delta k/k$		
BOC	123	123
EOC	110	110
Xenon worth - HFP, % $\Delta k/k$		
BOC (4 days)	2.43	2.45
EOC (equilibrium)	2.65	2.59
Effective delayed neutron fraction - HFP		
BOC	0.00618	0.00611
EOC	0.00523	0.00519

(a) Cycle 10 data are for the conditions stated in this report. The Cycle 9 core conditions are identified in Reference 5.

(b) Based on a 400-EFPD Cycle 8. (Actual Cycle 8 length 396.59 EFPD).

(c) Based on a Cycle 9 length of 421-EFPD.

(d) Nominal positions are as follows:

	Cycle 9	Cycle 10
HZP (BOC)	Group 7 at 100% WD, 8 at 25.5% WD	Group 7 at 100% WD, 8 at 25.0% WD
HFP (BOC)	Group 7 at 92% WD, 8 at 15% WD	Group 7 at 92% WD, 8 at 35% WD
HZP (EOC)	Group 7 at 100% WD, 8 at 25.5% WD	Group 7 at 100% WD, 8 at 25.0% WD
HFP (EOC)	Group 7 at 100% WD, 8 at 15% WD	Group 7 at 100% WD, 8 at 35% WD

(e) (15% to 100% WD for Cycle 9, 35% to 100% WD for Cycle 10)

Table 5-2. Shutdown Margin Calculation for Oconee 3, Cycle 10

	BOC, % Δk/k	EOC, % Δk/k
Available Rod Worth		
Total rod worth, HZP	8.90	9.50
Worth reduction due to poison burnup	-0.42	-0.42
Maximum stuck rod, HZP	-1.57	-1.98
Net worth	6.91	7.10
Less 10% uncertainty	-0.69	-0.71
Total available worth	6.22	6.39
Required Rod Worth		
Power deficit, HFP to HZP	1.88	3.17
Max inserted rod worth, HFP	0.37	0.54
Total required worth	2.25	3.71
Shutdown Margin		
Total available worth minus total required worth	3.97	2.68

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

FIGURE 5-1
OCONEE 3, CYCLE 10
TWO DIMENSIONAL
RELATIVE POWER DISTRIBUTION

HFP, 004 EFPD, EQXE
NOMINAL ROD POSITIONS

	8	9	10	11	12	13	14	15
H	1.007	1.240	1.074	0.924	1.146	1.208	1.062	0.554
K	1.240	1.130	1.284	1.062	1.257	1.220	1.246	0.529
L	1.074	1.284	1.254	1.236	0.983	1.280	0.968	0.366
M	0.924	1.062	1.236	1.022	1.199	1.086	0.667	
N	1.146	1.257	0.983	1.199	1.109	1.058	0.431	
O	1.208	1.220	1.280	1.086	1.058	0.534		
P	1.062	1.246	0.968	0.667	0.431			
R	0.554	0.529	0.366					

6. THERMAL-HYDRAULIC DESIGN

The generic Mark-B and Mark-BZ thermal-hydraulic design analyses supporting Cycle 10 operation were performed by Duke Power Company using the methods described in References 1, 5, 8, and 10. The Cycle 9 and Cycle 10 maximum design conditions are summarized in Table 6-1.

The Cycle 10 transition core will include 60 fresh Mark-BZ Batch 12 fuel assemblies, all of which will contain BPRAs. Two assemblies will contain regenerative neutron sources, leaving 46 fuel assemblies with open guide tubes. This results in a core bypass flow of 7.9% of the total system flow. This bypass flow is less than that assumed in the generic thermal-hydraulic analyses and the consequent increase in core flow establishes the generic analyses as conservative for Cycle 10 operation.

The Mark-BZ fuel assembly has a slightly higher pressure drop than the Mark-B assembly as a result of the increased flow resistance of the Zircaloy spacer grids. The presence of Mark-BZ and Mark-B assemblies in a core results in less coolant flow in the Mark-BZ fuel then would occur in an all Mark-BZ core. The generic Mark-BZ analyses conservatively account for this transition core effect.

In a Mark-BZ transition core the limiting Mark-B hot channel will receive more coolant and yield better DNB performance than would be predicted for a full Mark-B core. Thus, the generic Mark-B analyses, based on the B&W-2 CHF correlation, are bounding and are applicable to the Cycle 10 transition core.

No fuel rod bow penalty was included in the DNBR limit used in the generic Mark-BZ analyses, as justified in Reference 9. The rod bow topical report concludes that a DNBR penalty is no longer required for thermal-hydraulic analyses. Nevertheless, to account for fuel rod bow, the generic Mark-B analyses used for determining plant operating limits (except the flux to flow setpoint analysis) were based on a DNBR criteria including 10.2% margin from

the 1.30 design limit. Primarily due to this conservatism, the current pressure-temperature envelope and design radial x local peaking have been shown to be conservative for a full and transition Mark-BZ core.

A flux to flow setpoint of 1.07 will be used for Cycle 10 operation. A conservative transition core pump coastdown analysis was performed based on a 1.08 flux to flow setpoint and the reference design radial-local peaking factor, $F_{\Delta H}$ = 1.714. The minimum DNBR determined in the Mark-BZ transition core flux to flow analysis is greater than the BWC CHF correlation limit of 1.18, Reference 11. The minimum DNBR determined in the generic Mark-B flux to flow analysis, also based on a 1.08 flux to flow setpoint, is greater than the BAW-2 CHF correlation limit of 1.30.

Table 6-1. Thermal Hydraulic Design Conditions

	Cycle 9	Cycle 10
Design power level, MWt	2568	2568
System pressure, psia	2200	2200
Reactor coolant flow, % design flow	106.5	106.5
Core bypass flow, % total flow[(a)]	7.9	7.9
Vessel inlet/outlet coolant temp at 100% power, °F	555.6/602.4	555.6/602.4
Ref design radial-local power peaking factor	1.71	1.71
Ref design axial flux shape	1.5 cosine	1.5 cosine
Active fuel length, in.	141.8	141.8
Avg heat flux at 100% power, 10^3 Btu/h-ft^2	176[(b)]	176[(b)]
CHF correlation	BAW-2/BWC	BAW-2/BWC
Min DNBR with densification penalty	2.05/>1.74	>2.05/>1.74
Hot channel factors: Enthalpy rise	1.011/1.011	1.011/1.011
Heat flux	1.014/1.014	1.014/1.014
Flow area	0.98/0.97	0.98/0.97

[(a)]Generic analyses based on $\geq$8.0% core bypass flow.

[(b)]Heat flux based on a conservative minimum densified length of 140.3 in.

7. ACCIDENT AND TRANSIENT ANALYSIS

7.1 General Safety Analysis

Each FSAR[1] accident analysis has been examined with respect to changes in Cycle 9 parameters to determine the effect of the Cycle 10 reload and to ensure that thermal performance during hypothetical transients is not degraded. The effects of fuel densification on the FSAR accident results have been evaluated and are reported in Reference 8. Since Batch 12 reload fuel assemblies contain fuel rods with a theoretical density higher than those considered in Reference 8, the conclusions in that reference are still valid.

No new dose calculations were performed for the reload report. The dose considerations in Reference 13 are characteristic for Oconee 3 Cycle 10 based upon comparisons of key parameters which determine radionuclide inventories.

7.2 Accident Evaluations

The key parameters that have the greatest effect on determining the outcome of a transient can typically be classified in three major areas: core thermal parameters, thermal-hydraulic parameters, and kinetics parameters, including the reactivity feedback coefficients and control rod worths.

Fuel thermal analysis parameters for each batch in Cycle 10 are given in Table 4-2. Table 6-1 compares the Cycle 9 and 10 thermal-hydraulic maximum design conditions. Table 7-1 compares the key kinetics parameters from the FSAR and Cycle 10. The effect of a more negative hot full power end-of-cycle moderator temperature coefficient on the FSAR accident analyses has been analyzed for Oconee Nuclear Station.[16] Table 7-1 has been revised to include the new values for end-of-cycle moderator temperature coefficient and dropped rod worth assumed in these analyses.

A generic LOCA analysis for the B&W 177-FA, lowered-loop NSSS has been performed using the Final Acceptance Criteria ECCS Evaluation Model. This study is reported in BAW-10103, Rev. 3.[12] The analysis in BAW-10103 is generic since the limiting [illegible] parameters [illegible] all plants in this category

were used. Furthermore, the combination of average fuel temperature as a function of LHR and the lifetime pin pressure data used in the BAW-10103 LOCA limits analysis[7,12] is conservative compared to those calculated for this reload. In addition, it has been determined that the slightly lower prepressurization of the Batch 12 fuel rods has a negligible impact on the LOCA analyses[14]. Thus, the analysis and the LOCA limits reported in BAW-10103 provide conservative results for the operation of Oconee 3, Cycle 10 fuel.

The LOCA kW/ft limits have been reduced for the first 65 EFPDs. This reduction will ensure conservative limits based upon an interim bounding analytical assessment of NUREG 0630 on LOCA and operating kW/ft limits performed by Babcock and Wilcox[4,15]. The LOCA kw/ft limits for the first 65 EFPD are shown in Table 7-2. Table 7-3 shows the bounding values for allowable LHRs for Oconee 3 Cycle 10 fuel after 65 EFPD.

From the examination of Cycle 10 core thermal properties and kinetics properties with respect to acceptable previous cycle values, it is concluded that this core reload will not adversely affect the safe operation of Oconee 3 during Cycle 10. Considering the previously accepted design basis used in the FSAR and subsequent cycles, the transient evaluation of Cycle 10 is considered to be bounded by previously accepted analyses. The initial conditions of the transients in Cycle 10 are bounded by the FSAR and/or the fuel densification report.[8]

Table 7.1. Comparison of Key Parameters for Accident Analysis

Parameter	FSAR[1] value	Predicted Cycle 10 value
BOC Doppler coeff, 10^{-5}, $\Delta k/k/°F$	-1.17	-1.39
EOC Doppler coeff, 10^{-5} $\Delta k/k/°F$	-1.33[(a)]	-1.73
BOC moderator coeff, 10^{-4}, $\Delta k/k°F$	+0.5[(b)]	-0.79
EOC moderator coeff, HZP, 10^{-4}, $\Delta k/k/°F$	-3.0[(c)]	-2.71
, HFP, 10^{-4}, $\Delta k/k/°F$	-3.5[(c)]	-3.27
All rod bank worth, HZP, % $\Delta k/k$	10.0	9.50
Boron reactivity worth, 70°F ppm/1% $\Delta k/k$	75	87
Max. ejected rod worth, HFP, % $\Delta k/k$	0.65	0.24
Dropped rod worth, HFP, % $\Delta k/k$	0.46	0.13
Initial boron conc, HFP, ppm	1400	1330

(a) -1.2×10^{-5} $\Delta k/k/F$ was used for steam line break analysis.
-1.3×10^{-5} $\Delta k/k/F$ was used for cold water accident (pump start-up).

(b) $+0.94 \times 10^{-4}$ $\Delta k/k/F$ was used for the moderator dilution accident.

(c) The HZP moderator temperature coefficient is one of the key parameters assumed in the steam line break analysis. The HFP moderator temperature coefficient is included since it is one of the parameters assumed in the cold water, rod ejection, and control rod misalignment accident analyses, although none of these accidents are very sensitive to changes in the coefficient.

Table 7-2. LOCA Limits, Oconee 3, Cycle 10

Elevation, ft	LHR Limits, kW/ft	
	0-1000 MWd/mtU[a]	1000-2600 MWd/mtU[b]
2	13.5	15.0
4	16.1	16.6
6	16.5	18.0
8	17.0	17.0
10	16.0	16.0

Table 7-3. LOCA Limits, Oconee 3, Cycle 10,
After 2600 MWD/mtU(b)

Elevation, ft	LHR limits, kW/ft
2	15.5
4	16.6
6	18.0
8	17.0
10	16.0

(a) 1000 MWd/mtU corresponds to approximately 25 EFPD for the most limiting assembly

(b) 2600 MWd/mtU corresponds to approximately 65 EFPD for the most limiting assembly

8. PROPOSED MODIFICATIONS TO TECHNICAL SPECIFICATIONS

The Technical Specifications have been revised for Cycle 10 operation in accordance with the methods of Reference 10 to account for minor changes in power peaking and control rod worths.

In addition:

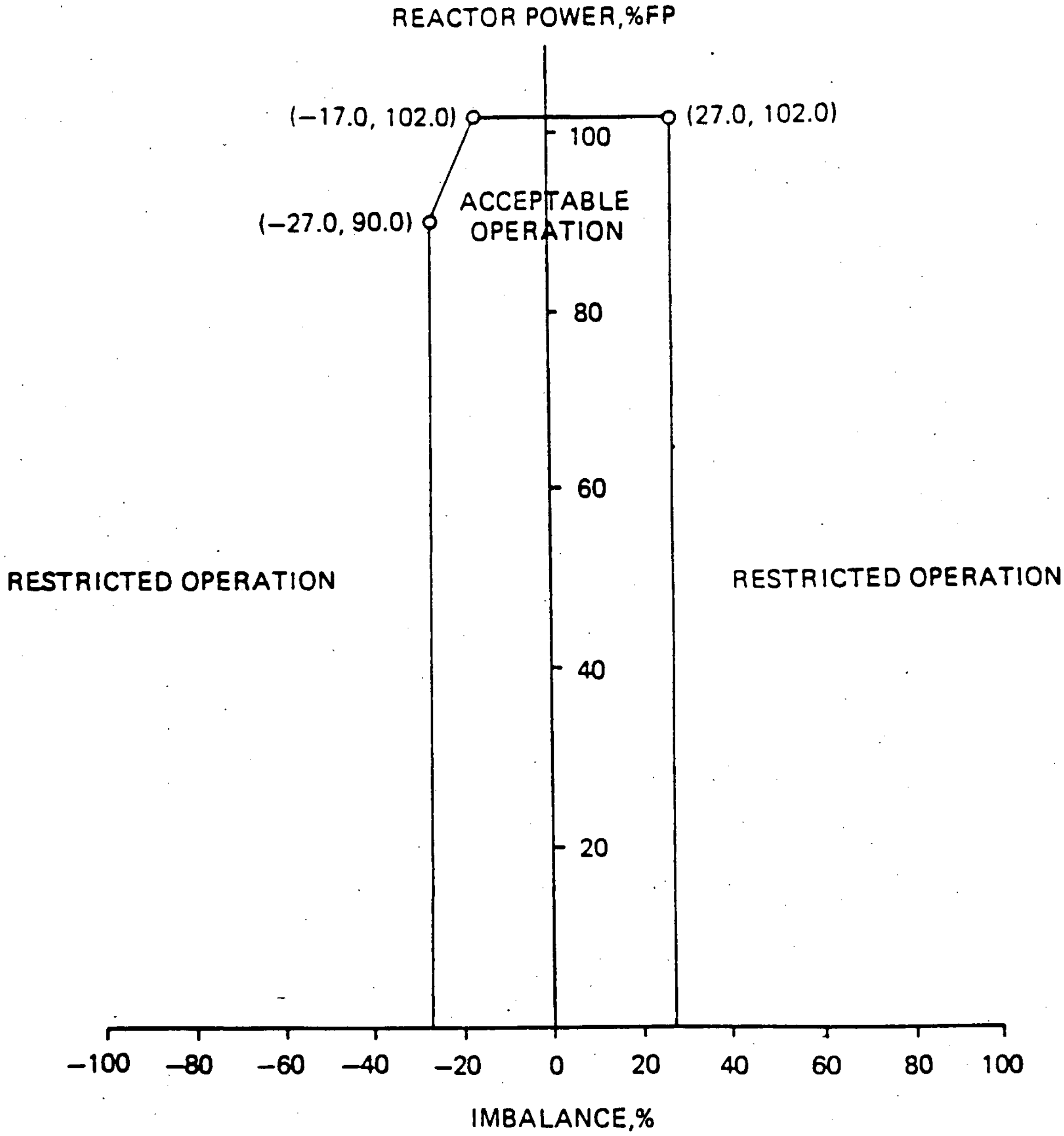
1. The operating limits on rod index and axial power imbalance were developed in accordance with the LOCA linear heat rate limits discussed in Chapter 7.

2. Due to the lower worth of the gray APSRs, operational limits for the Group 8 control rods are not required for Cycle 10. Administrative position and maneuvering limits for the APSRs, however, will be utilized to ensure adequate fuel performance during Cycle 10 operation.

Based on the Technical Specifications derived from the analyses presented in this report, The Final Acceptance Criteria ECCS limits will not be exceeded, nor will the thermal design criteria be violated. Figure 8-1 is a revision to a previous Technical Specification limit.

Figure 8-1

Operational Power-Imbalance Limits, 0 EFPD to EOC



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