Duke Power Company

Oconee Nuclear Station

OCONEE UNIT 1, CYCLE 14

- Reload Report-

DPC - RD - 2018

March 1991

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2. OPERATING HISTORY

The referenced fuel cycle for the nuclear and thermal-hydraulic analyses of Oconee Unit 1, Cycle 14, is the currently operating Cycle 13. Cycle 13 achieved initial criticality on June 5, 1990 and power escalation commenced on June 6, 1990. The fuel cycle design length for Cycle 14 - 390 ± 10 EFPD - is based on a Cycle 13 length of 410 ± 10 EFPD. No operating anomalies occurred during previous cycle operations that would adversely affect fuel performance in Cycle 14.

Cycle 14 will operate in a feed-and-bleed mode for its entire design length, as did Cycle 13.

3. GENERAL DESCRIPTION

The Oconee Unit 1 reactor core and fuel design basis are described in detail in Chapter 4, of the FSAR[1]. The Cycle 14 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 1, Cycle 14. The remaining 28 Batch 12C assemblies, along with 37 of the Batch 13C assemblies, will be discharged at the end of Cycle 13. The central portion of the core is primarily composed of the 4 Batch 11D assemblies (3.31 wt% ^{235}U), 8 Batch 13C assemblies (3.38 wt% ^{235}U), 52 Batch 15 assemblies (3.55 wt% ^{235}U), and 52 fresh Batch 16 assemblies (3.55 wt% ^{235}U). The Batch 10F and Batch 14 assemblies, with initial enrichments of 3.41 and 3.65 wt% ^{235}U, will primarily occupy the core periphery. Figure 3-2 is a quarter-core map showing the assembly burnup and enrichment distribution at the beginning of Cycle 14.

Cycle 14 will operate in a rods-out, feed-and-bleed mode. Core reactivity control is supplied mainly by soluble boron and supplemented by 61 full-length Ag-In-Cd control rods and 44 burnable poison rod assemblies (BPRAs). In addition to the full-length control rods, eight Inconel gray axial power shaping rods (APSRs) are provided for additional control of the axial power distribution. The Cycle 14 locations of the 69 control rods and the group designations are indicated in Figure 3-3. The Cycle 14 locations and enrichments of the BPRAs are shown in Figure 3-4.

4. FUEL SYSTEM DESIGN

4.1 Fuel Assembly Mechanical Design

The types of fuel assemblies and pertinent fuel design parameters for Oconee 1 Cycle 14 are listed in Table 4-1. All fuel assemblies are mechanically interchangeable. Two regenerative neutron sources will be used in Mark-B8 fuel assemblies.

The Batch 16 feed assemblies consist of 48 Mk-B9 assemblies and 4 Mk-B10[5] LTA's. The Mk-B10 assemblies are similar to Mk-B9 with the exception of the upper end fitting and the holddown spring contained therein. The holddown springs in the 4 Mk-B10 LTA's are the cruciform design as opposed to the traditional helical springs in the Mk-B9 assemblies. The new spring design is being implemented to improve holddown spring reliability. The cruciform springs have a significantly greater holddown capacity so that current fuel assembly lift analyses based on the helical spring design are bounding.

The Mk-B9 and Mk-B10 fuel assembly designs are an evolution from the existing Mk-B fuel assembly series presently in service. They are 15X15 fuel rod lattices with Zircaloy spacer grids and are identical in every regard to the earlier Mk-B designs with the exception of the following improvements:

1. Cruciform holddown spring (B10 only)[5]
2. Reduced pellet/clad gap fuel rod
3. Reduced bypass flow guide tubes
4. Skirtless, removable lower end fitting

These improvements are discussed below.

4.1.1 Reduced Pellet/Clad Gap Fuel Rod

SEE SECTION 4.2

4.1.2 Reduced Bypass Flow Guide Tubes

The guide tubes which provide means for control component insertion have been optimized in the Mk-B9 fuel assembly design to reduce bypass flow and improve fuel thermal hydraulic performance. This has been accomplished by a reduction in the number and size of the flow holes provided near the bottom of the guide tube. Reduced guide tube flow is favorable for thermal design considerations (see section 4.3) but too much reduction adversely affects control rod drop time and core component cooling. BWFC has performed control rod drop tests which have confirmed that Technical Specification limits are not exceeded. BWFC has also performed component analyses to ensure bulk boiling or centerline melt do not occur, and to ensure that core component internal pressures are acceptable. Those results also indicate that reduced guide tube flow is acceptable.

4.1.3 Removable Lower End Fitting

The Mk-B9/B10 fuel assemblies incorporate a removable lower end fitting to provide added flexibility in field maintenance. This improvement complements a removable upper end fitting presently in service which has been successfully utilized in field repair operations. The design includes the use of a lower end spacer grid identical to the grid used at the upper end of the fuel assembly. This spacer grid is made of Inconel 718 and is retained by the lower end plugs of the guide tube assemblies rather than the lower end fitting as before. The lower end plugs of the guide tube assemblies extend through the lower end grid and have restraining ledges that interface with the saddles of the spacer grid assembly preventing movement of the grid. This lower end spacer grid has better lead-in features facilitating fuel handling. The lock nut used to attach the lower end fitting has an integral crimping sleeve that secures the lock nut and permits remote removal and installation. It is crimped into grooves in the guide tube lower end plug. The lower end fitting has square recesses that receive a mating feature on the guide tube lower end plug in order to prevent guide tube rotation. The instrument tube has a restraint feature that interfaces with the upper end grid ensuring retention of the instrument tube in the fuel assembly during lower end fitting removal.

4.2 Fuel Rod Design

The fuel rod design for the Mk-B9/B10 is based on the previously used Mk-B8 design. The Mk-B9/B10 design has the same debris resistant configuration as the Mk-B8. Debris resistance is provided by extending the solid length of the lower end plug through the bottom spacer grid. This provides debris protection by using the bottom spacer grid as a debris filter. The differences between the Mk-B9/B10 fuel rod and the Mk-B8 fuel rod are:

- Pellet diameter increased from 0.3686 to 0.370 inches.
- Pellet-cladding diametral gap decreased from 0.0084 to 0.007 inches.
- Fuel stack length decreased from 141.8 to 140.595 inches.
- Slight changes in pellet dish geometry.

This results in a fuel rod with the same loading as the previous design, but with a smaller pellet-cladding gap and a larger plenum volume. The advantages of a decreased pellet gap and an increased plenum volume are lower fuel temperatures and higher pin pressure burnup limits. The mechanical evaluation of the Mk-B9/B10 rod design is discussed in this section.

4.2.1 Cladding Collapse

The fuel assemblies in Batch 13C are more limiting than the other batches due to their longer previous incore exposure time. This batch was analyzed for creep-collapse using the CROV computer code and procedures described in topical report BAW-10084, Rev. 2[3]. The CROV analyses are performed either in a generic fashion or on a batch specific basis, depending on available margins. The TACO2[4] code was used to calculate internal pin pressures and clad temperatures used as input to CROV. As shown in Table 4-1, the collapse time for the most limiting batches were conservatively determined to be greater than the maximum incore residence times.

4.2.2 Cladding Stress

As described in Reference 2, 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 2. 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 regard to stress

Duke's TACO2 analysis of the Mk-B9/B10 fuel designs demonstrate that the generic analyses are bounding.

4.2.3 Cladding Strain

Duke has performed a cladding strain calculation using TACO2 in accordance with approved methodology[2]. This analysis demonstrated that the uniform, circumferential strain of the cladding was within 1.0 %.

4.2.4 Cladding Fatigue

Duke has evaluated cladding fatigue for the Mk-B9/B10 fuel designs with TACO2. The conclusions indicate that the generic Mk-B8 fuel rod analysis envelopes the new fuel design because the pressures used in the generic analysis are much greater than those the Mk-B9/B10 fuel rod will experience.

4.3 Thermal Design

All fuel assemblies in the Cycle 14 core are thermally similar except the fresh Batch 16 fuel. The fresh Batch 16 fuel introduces significant differences in fuel thermal performance relative to the other fuel remaining in the core. Duke has performed the thermal evaluation with the TACO2 computer code. The results indicate that the current LHR limits (see Table 4-1) are bounding.

The linear heat rate to melt capability based on centerline fuel melt was assessed separately for each batch of fuel against Duke's generic linear heat rate to melt analysis, which is based on the TACO2 computer code. The fuel parameters used to determine the fuel melt limits are shown in Table 4-2.

The analysis 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 LHR capability and the average fuel temperature for each batch are shown in Table 4-2.

The maximum assembly and rod average burnups are predicted by Duke to be 47,688 MWD/MTU and 49,324 MWD/MTU, respectively, for Batch 14 fuel. Fuel rod internal pressure has been evaluated using TACO2 with conservative pin power history, and the maximum pressure is less than the nominal reactor coolant (RC) system pressure of 2200 psia.

The Mk-B9/B10 fuel designs have optimized guide tubes. These guide tubes have fewer and smaller flow holes than conventional guide tubes in the Mk-B design series. Therefore, with less guide tube bypass flow, more flow is available for heat removal in the core. The DNB and hydraulic lift force calculations have accounted for the presence of the optimized guide tubes in the fresh fuel.

4.4 Material Compatibility

The Batch 16 fuel assemblies do not utilize different component materials. Therefore, the chemical compatibility of all possible fuel-cladding-coolant-assembly interactions for the Batch 16 fuel assemblies is identical to those of the present fuel.

Table 4-1

Fuel Design Parameters and Dimensions

	Batch No.						
	10F	11D	13C	14	15	16	16
FA type	Mark B4	Mark B4Z	Mark B6	Mark B6	Mark B8	Mark B9	Mark B10 Demo
No. of FAs	9	4	8	52	52	48	4
Fuel rod OD, in.	0.430	0.430	0.430	0.430	0.430	0.430	0.430
Fuel rod ID, in.	0.377	0.377	0.377	0.377	0.377	0.377	0.377
Flex spacers, type	Spring	Spring	Spring	Spring	Spring	Spring	Spring
Rigid spacers, type	Zr-4	Zr-4	Zr-4	Zr-4	none	none	none
Undensified active fuel length,	143.5	141.8	141.8	141.8	141.8	140.6	140.6
Fuel pellet OD (mean spec), in.	0.3686	0.3686	0.3686	0.3686	0.3686	0.370	0.370
Fuel pellet initial density (mean spec), %TD	95.0	95.0	95.0	95.0	95.0	95.0	95.0
Initial fuel enrichment, wt % ^{235}U	3.41	3.31	3.38	3.65	3.55	3.55	3.55
Est. residence time, EOC 14, Hours	29,280	29,112	39,336	29,520	19,680	9600	9600
Cladding collapse time, Hours	>40,000	>40,000	>40,000	>40,000	>55,000	>55,000	>55,000

Table 4-2. Linear Heat Rate to Melt Analysis

	Batch No.						
	10F	11D	13C	14	15	16	16
Nominal initial density, % TD	95.0	95.0	95.0	95.0	95.0	95.0	95.0
Nominal initial pellet diameter, in.	0.3686	0.3686	0.3686	0.3686	0.3686	0.370	0.370
Nominal initial clad ID, in.	0.377	0.377	0.377	0.377	0.377	0.377	0.377
Nominal initial clad OD, in.	0.430	0.430	0.430	0.430	0.430	0.430	0.430
Average linear heat rate @ 100% of 2568 MW, kW/ft	5.74	5.74	5.74	5.74	5.74	5.79	5.79
Linear heat rate capability[b] from 0-1000 MWD/MTU, kW/ft	≥20.15	≥20.15	≥20.15	≥20.15	≥20.15	≥20.15	≥20.15
Linear heat rate capability[b] [illegible]000 MWD/MTU, kW/ft	≥21.20	≥21.20	≥21.20	≥21.20	≥21.20	≥21.20	≥21.20
Average fuel temp. @ nominal linear heat rate, °F	1240[a]	1240[a]	1240[a]	1240[a]	1240[a]	≤1240[a]	≤1240[a]

(a) Basis: TACO2[4], 96.5% TD @ 4000 MWD/MTU, nominal pellet and cladding dimensions.
(b) These values are utilized as fuel design limits for Cycle 14.

5. NUCLEAR DESIGN

5.1 Physics Characteristics

Table 5-1 compares the core physics parameters of design Cycle 14 with those of the reference Cycle 13. The Cycle 13 and 14 values were generated by Duke Power Company using the CASMO-2 based reload design methods described in Reference 2. Since the core has not yet reached an equilibrium cycle, differences in core physics parameters are to be expected between the cycles. Figure 5-1 illustrates a representative relative power distribution for the beginning of Cycle 14 at full power with equilibrium xenon and normal rod positions.

The primary reasons for the differences in the physics parameters between Cycles 13 and 14 are the different BPRA loadings and different core loading patterns. Differences in ejected and stuck rod worths between cycles are due to changes in the radial flux and burnup distributions. All safety criteria associated with these rod worths are met. The adequacy of the shutdown margin with Cycle 14 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 1, Cycle 13 Reload Report[6].

5.2 Analytical Input

The Cycle 14 incore measurement calculation constants to be used to compute core power distributions were obtained in a similar manner for Cycle 14 as for the reference cycle.

5.3 Changes in Nuclear Design

There are no changes in design methodology between Oconee 1 Cycle 13 and Oconee 1 Cycle 14.

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

	Cycle 13[b]	Cycle 14[c]
Cycle length, EFPD	410	390
Cycle burnup, MWd/mtU	12831	12205
Average core burnup, EOC, MWd/mtU	29523	29862
Initial core loading, mtU	82.1	82.1
Critical boron - BOC (no xenon), ppm		
HZP, groups 7 and 8 at nominal positions[d]	1719	1671
HFP, groups 7 and 8 at nominal positions[d]	1494	1450
Critical boron - EOC (equilibrium xenon), ppm		
HZP, groups 7 and 8 at nominal positions[d]	378	391
HFP, groups 7 and 8 at nominal positions[d]	0	47
Control rod worths - HFP, BOC, % dk/k		
Group 7	1.01	0.87
Group 8[e]	0.14	0.14
Control rod worths - HFP, EOC, % dk/k		
Group 7	1.12	1.03
Group 8[e]	(f)	(f)
Max ejected rod worth - HZP, % dk/k		
BOC, groups 5-8 inserted	0.38(L10)	0.40(L10)
EOC, groups 5-8 inserted	0.51(N12)	0.38(N12)
Max stuck rod worth - HZP, % dk/k		
BOC	1.73(N12)	1.16(M13)
EOC	1.94(N12)	1.53(M13)
Power deficit, HFP to HZP, % dk/k		
BOC	1.84	1.90
EOC	3.19	3.16
Doppler coeff - HFP, 10^{-5} (dk/k-°F)		
BOC (equilibrium xenon)	-1.21	-1.22
EOC (equilibrium xenon)	-1.56	-1.55

Table 5-1. (cont'd)

	Cycle 13(b)	Cycle 14(c)
Moderator coeff - HFP, 10^{-4} (dk/k-°F)		
BOC (no xenon)	-0.95	-0.98
EOC(equilibrium xenon)	-3.40	-3.37
Boron worth - HFP, ppm/% dk/k		
BOC	130	130
EOC	117	118
Xenon worth - HFP, % dk/k		
BOC (4 days)	2.58	2.58
EOC (equilibrium)	2.73	2.71
Effective delayed neutron fraction - HFP		
BOC	0.00603	0.006007
EOC	0.00516	0.005178

(a) Cycle 14 data are for the conditions stated in this report. The Cycle 13 core conditions are identified in Reference 6.

(b) Based on a 410 ± 10 EFPD Cycle 12 (Actual Cycle 12 length of 409.7 EFPD).

(c) Based on a Cycle 13 length of 410 ± 10 EFPD.

(d) Nominal positions are as follows:

	Cycle 13	Cycle 14
HZP (BOC)	CRGP7,8 = 100,35% WD	CRGP7,8 = 100,35% WD
HZP (EOC)	CRGP7,8 = 100,35% WD	CRGP7,8 = 100,100% WD
HFP (BOC)	CRGP7,8 = 92,35% WD	CRGP7,8 = 92,35% WD
HFP (EOC)	CRGP7,8 = 92,35% WD	CRGP7,8 = 92,100% WD

(e) Worth is calculated from 35% to 100% WD for Cycles 13 and 14.

(f) CRGP8 = 100% WD, therefore, there is no CRGP8 worth at EOC.

Table 5-2. Shutdown Margin Calculation for Oconee 1, Cycle 14

	BOC, % dk/k	EOC, % dk/k
Available Rod Worth		
Total rod worth, HZP	8.40	8.82
Worth reduction due to poison burnup	−0.42	−0.42
Maximum stuck rod, HZP	−1.16	−1.53
Net worth	6.82	6.87
Less 10% uncertainty	−0.68	−0.69
Total available worth	6.14	6.18
Required Rod Worth		
Power deficit, HFP to HZP	1.90	3.16
Max inserted rod worth, HFP	0.33	0.49
Total required worth	2.23	3.65
Shutdown Margin		
Total available worth minus total required worth	3.91	2.53

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

6. THERMAL-HYDRAULIC DESIGN

The generic thermal-hydraulic design analyses supporting Cycle 14 operation were performed by Duke Power Company using the methods described in References 1, 2, 6, 7, and 8. Cycle 13 and Cycle 14 maximum design conditions are summarized in Table 6-1 and changes are discussed below.

O1C14 is the first Duke reload core to employ B&W's MK-B9 fuel assembly design, which incorporates the skirtless removable lower end fitting (LEF), optimized fuel rod design (larger pellet diameter), and optimized guide tube (OGT) flow conditions. Batch 16 also includes 4 demonstration assemblies of the MK-B10 design, which includes all the MK-B9 features, plus the improved upper end fitting (UEF) with the cruciform holddown leaf spring.

O1C14 was evaluated using an RCS flow of 107.5%, which is less than the measured flow for Unit 1 including measurement uncertainties. The inlet temperature was increased slightly to account for the increase in design flow, and the outlet temperature was adjusted to maintain an average RCS temperature of 579°F. The reduction in bypass flow is due to the MK-B9 OGT flow holes. Since the MK-B9 has a larger pellet diameter, the stack height was reduced in order to maintain the 463.6 kg of Uranium loading. References 18 and 19 show that small local heat flux spikes have no effect on CHF; therefore, the local heat flux hot channel factor was removed from all DNBR calculations.

A flux to flow setpoint of 1.07 will be used for Cycle 14 operation. A conservative pump coastdown analysis was performed based on a 1.07 flux to flow setpoint and the reference design radial-local peaking factor, $F_{\Delta H}$ = 1.714. The two pump coastdown is initiated from a power level of 102% full power. This initial condition is based on 100% indicated power plus a 2% allowance for the secondary side heat balance uncertainty. The reactor trip time was conservatively determined based on an NI indicated power of 98%, 100% minus 2% NI calibration error. The minimum DNBR determined in the Mark-BZ flux to flow analysis is greater than the BWC CHF correlation limit of 1.18, Reference 9. No fuel rod bow penalty was included in the DNBR limit used in the generic Mark-BZ analyses, as justified in Reference 10. The rod bow topical report concludes that a DNBR penalty is no longer required for thermal-hydraulic analyses.

Table 6-1. Thermal Hydraulic Design Conditions

	Cycle 13	Cycle 14
Design power level, MWt	2568	2568
System pressure, psia	2200	2200
Reactor coolant flow, % design flow	106.5	107.5
Core bypass flow, % total flow	8.6	8.3
Vessel inlet/outlet coolant temp at 100% power, °F	555.6/602.4	555.8/602.2
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	140.6
Avg heat flux at 100% power, 10^3 Btu/hr-ft^2	176[(a)]	176[(a)]
CHF correlation	BWC	BWC
Min DNBR with densification penalty	>1.74	>1.74
Hot channel factors: Enthalpy rise	1.011	1.011
Heat flux	1.014	1.000
Flow area	0.97	0.97

7. ACCIDENT AND TRANSIENT ANALYSIS

7.1 General Safety Analysis

Each FSAR[1] accident analysis has been examined with respect to changes in Cycle 13 parameters to determine the effect of the Cycle 14 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 the Batch 16 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 11 are characteristic for Oconee 1 Cycle 14 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 14 are given in Table 4-2. Table 6-1 compares the Cycle 13 and 14 thermal-hydraulic maximum design conditions. Table 7-1 compares the key kinetics parameters from the FSAR and Cycle 14. 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[12]. 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 given in BAW-10103, Rev. 3[13]. The LOCA kW/ft limits given in BAW-10103 have been impacted by TACO2, NUREG-0630, and FLECSET. The net effect of these factors is summarized by the kW/ft limits in BAW-1915[14]. Due to an error in Babcock & Wilcox's evaluation model, the LOCA linear heat rate analyses have been revised. It was found that this error impacted the 4 ft and 6 ft elevation LHRs such that little margin was left between the peak clad temperature (PCT) and the acceptance criterion of 2200°F. B&W re-evaluated the 4 ft elevation using the same LHR (16.1 kW/ft) with TACO3[15]. This evaluation[16] showed that adequate margin was gained with the use of TACO3[15] such that the 16.1 kW/ft limit at the 4 ft elevation was still acceptable. Rather than have B&W reanalyze the 6 ft elevation using TACO3[15] to gain margin, a 0.4 kW/ft reduction was applied to this elevation. This provides adequate margin in the PCT without impacting core designs.

The combination of average fuel temperature as a function of LHR and the lifetime pin pressure data used in the BAW-1915 LOCA limits analysis[14] is conservative compared to those calculated for this reload. Thus, the analysis and the LOCA limits reported in BAW-1915, including the re-evaluation of the 4 ft LHR[16], provide conservative results for the operation of Oconee 1 Cycle 14 fuel[17].

The LOCA kW/ft limits have been reduced for the first 25 EFPDs. The kW/ft limits for the first 25 EFPDs are shown in Table 7-2. Table 7-3 shows the bounding values for allowable LOCA peak LHRs for Oconee 1 Cycle 14 fuel after 25 EFPD.

From the examination of Cycle 14 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 1 during Cycle 14. Considering the previously accepted design basis used in the FSAR and subsequent cycles, the transient evaluation of Cycle 14 is considered to be bounded by previously accepted analyses. The initial conditions of the transients in Cycle 14 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 14 value
BOC Doppler coeff, HFP, 10^{-5}, dk/k/°F	−1.17	−1.22
EOC Doppler coeff, HFP, 10^{-5}, dk/k/°F	−1.33[a]	−1.55
BOC moderator coeff, HFP, 10^{-4},dk/k/°F	+0.5[b]	−0.98
EOC moderator coeff, HZP, 10^{-4},dk/k/°F	−3.0[c]	−2.88
EOC moderator coeff, HFP, 10^{-4},dk/k/°F	−3.5[c]	−3.37
All rod bank worth, HZP, % dk/k	10.0	8.82
Boron reactivity worth, 70°F ppm/1% dk/k	75	93
Max. ejected rod worth, HFP, % dk/k	0.65	0.22
Dropped rod worth, HFP, % dk/k	0.40	0.13
Initial boron conc, HFP, ppm	1400	1450[d]

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

(b) $+0.94 \times 10^{-4}$ dk/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.

(d) The combined effect of boron concentration and boron worth is conservative for Cycle 14 as compared to that for the FSAR.

Table 7-2. LOCA Limits, Oconee 1, Cycle 14 0-1000 Mwd/mtU[a]

Elevation ft	LHR Limits, kW/ft
2	14.0
4	16.1[b]
6	16.1[b]
8	17.0
10	16.0

Table 7-3. LOCA Limits, Oconee 1, Cycle 14, After 1000 Mwd/mtU

Elevation, ft	LHR limits, kW/ft
2	15.5
4	16.6[b]
6	16.1[b]
8	17.0
10	16.0

[a] 1000 MWd/mtU corresponds to approximately 25 EFPD for the most limiting assembly.

[b] Reference 16

8. PROPOSED MODIFICATIONS TO TECHNICAL SPECIFICATIONS

The Technical Specifications have been revised for Cycle 14 operation in accordance with the methods of References 2 and 7 to account for minor changes in power peaking and control rod worths. In addition, Technical Specification 5.3 is revised to account for the reduced stack length of the Batch 16 fuel assemblies.

The figures for the operating limits on rod index and axial power imbalance have been removed from Section 3 of the Technical Specifications and are included in the cycle specific Core Operational Limits Report (COLR). Both the operational power imbalance limits and rod index limits are revised and included in the COLR for Cycle 14.

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. These operational limits are provided in the Oconee 1 Cycle 14 Core Operational Limits Report.

2. Due to the lower worth of the gray APSRs, operational limits for the Group 8 control rods are not required for Cycle 14. Administrative position and maneuvering limits for the APSRs, however, will be utilized to ensure adequate fuel performance during Cycle 14 operation.

In addition to the proposed revision to Technical Specification 5.3, several administrative changes to the Technical Specifications are requested. It is proposed that the following items be removed from the Technical Specifications and included in the cycle specific Core Operational Limits Report (COLR):

(next page)

1. The pressure/temperature and flux/flow/imbalance safety limit figures (Figures 2.1-1 and 2.1-2).

2. The pressure/temperature and flux/flow/imbalance limiting safety system settings (Figures 2.3-1 and 2.3-2).

3. The peaking factors in the bases for Technical Specification 2.1.

4. The limits on quadrant power tilt (Table 3.5-1).

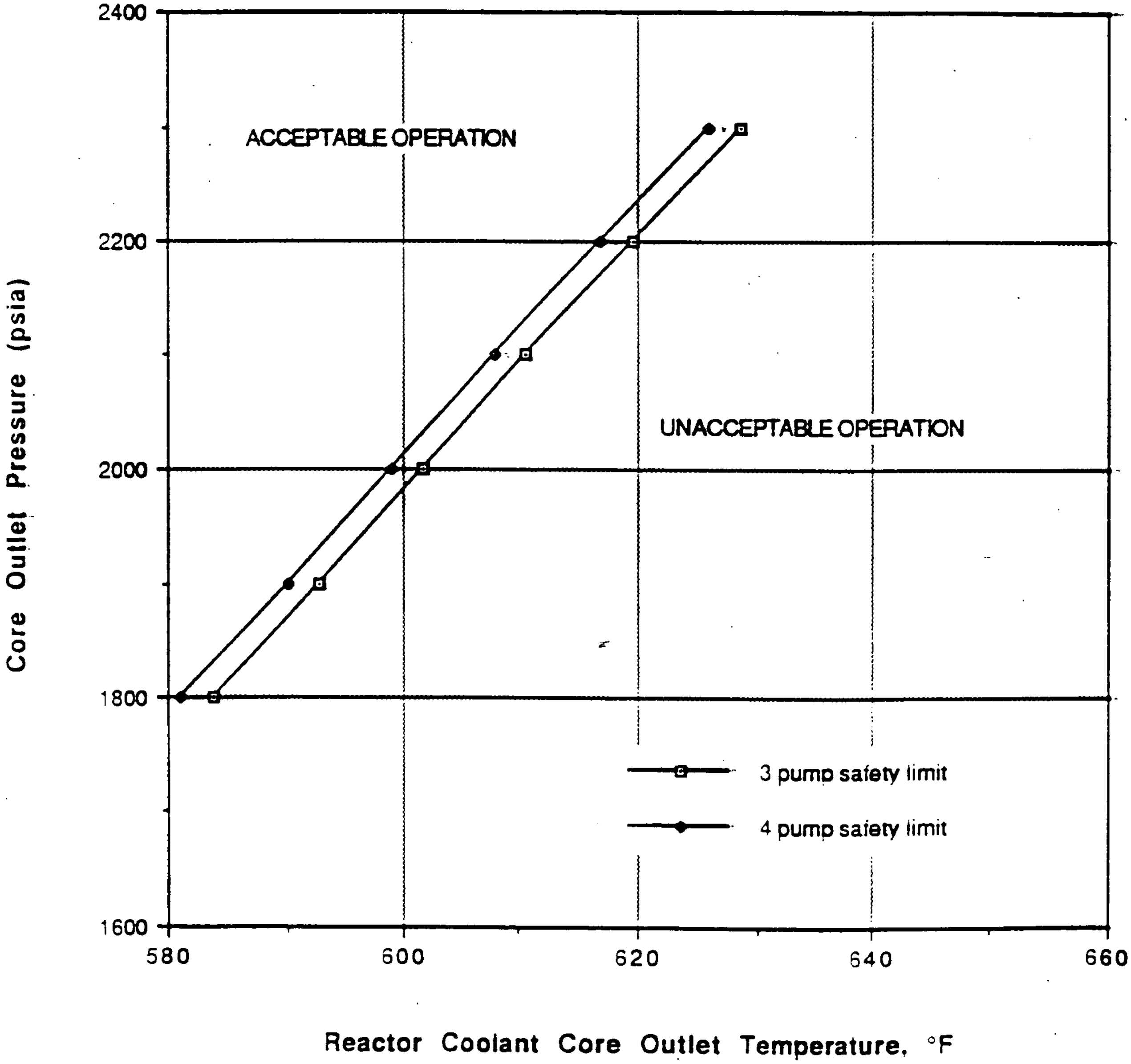
The non-cycle specific safety limits for fuel centerline temperature and DNBR are included in Section 2 of the Technical Specifications.

The variable low pressure core protection safety limits are revised for O1C14. Figure 8-1 shows the revised safety limits. In the past, the difference between the variable low pressure safety limit and the maximum allowable trip setpoint has significantly exceeded the uncertainty allowance for this trip function. The safety limit has been revised such that the difference between the safety limit and trip setpoint reflects the actual uncertainty allowance for the variable low pressure trip function. The uncertainty allowance for this trip function has been determined in the same manner as done in the past, using the square root of the sum of the squares methodology. Error adjustment of the variable low pressure safety limit falls outside the current Reactor Protective System (RPS) maximum allowable setpoint, thus making it unnecessary to revise the setpoint for this trip function.

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

Core Protection Safety Limits Unit 1



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