APR 18 1985

DO NOT REMOVE

Posted
amdt. 134
to DPR-55

LICENSE AUTHORITY FILE COPY

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

Distribution
Docket File
LPDR
NRC PDR
ORB#4 Rdg
Gray Files+4
HThompson
CMiles
OELD
BGrimes
EBlackwood
MDunenfeld
WJones
LHarmon
TBarnhart+12
HNicolaras
RIngram
ACRS-10
EButcher
JPartlow
EJordan
RDiggs
HOrnstein

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

Dear Mr. Tucker:

The Commission has issued the enclosed Amendments Nos. 137 , 137 , and 134 to Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units Nos. 1, 2 and 3. These amendments consist of changes to the Station's common Technical Specifications (TSs) in response to your request dated December 19, 1984, as supplemented on March 8, 1985.

These amendments revise the TSs to support the operation of Oconee Unit 2 at full rated power during the upcoming Cycle 8. The amendments change the following areas:

1. Rod Position Limits of TS 3.5.2; and
2. Power Imbalance Limits of TS 3.5.2.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance of the enclosed amendments will be included in the Commission's next monthly FEDERAL REGISTER notice.

Sincerely,

Original signed by

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

Enclosures:
1. Amendment No. 137 to DPR-38
2. Amendment No. 137 to DPR-47
3. Amendment No. 134 to DPR-55
4. Safety Evaluation

cc w/enclosures:
See next page

*See previous white for concurrences:

ORB#4:DL	ORB#4:DL	ORB#4:DL	OELD	AD:OR:DL
RIngram*	HNicolaras;	JStolz*	GJohnson*	GLainas
4/1/85	4/16/85	4/3/85	4/9/85	4/17/85

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

Distribution
Docket File
LPDR
NRC PDR
ORB#4 Rdg
Gray Files+4
HThompson
CMiles
OELD
BGrimes
EBlackwood
MDunenfeld
WJones
LHarmon
TBarnhart+12
HNicolaras
RIngram
ACRS-10
EButcher
JPartlow
EJordan
RDiggs
HOrnstein

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

Dear Mr. Tucker:

The Commission has issued the enclosed Amendments Nos. , , and to Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units Nos. 1, 2 and 3. These amendments consist of changes to the Station's common Technical Specifications (TSs) in response to your request dated December 19, 1984, as supplemented on March 8, 1985.

These amendments revise the TSs to support the operation of Oconee Unit 2 at full rated power during the upcoming Cycle 8. The amendments change the following areas:

1. Rod Position Limits of TS 3.5.2; and
2. Power Imbalance Limits of TS 3.5.2.

By letter dated January 22, 1985, as revised on February 20 and March 8, 1985, you modified the Oconee Startup Physics Test Program. The proposed modifications to the Program requested by your January 22, 1985, letter were reviewed by the staff, and the staff has found the proposed startup testing program modifications to be acceptable.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance of the enclosed amendments will be included in the Commission's next monthly FEDERAL REGISTER notice.

Sincerely,

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

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

cc w/enclosures:
See next page

ORB#4:DL	ORB#4:DL	ORB#4:DL	OELD	AD:OR:DL
RIngram	HNicolaras;	JStolz		GLainas
4/1/85	4/1/85	4/3/85	4/9/85	/ /85
			with deletion of all references to Test Program	

Duke Power Company

cc w/enclosure(s):

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

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

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

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

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

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

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

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

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

Dr. J. Nelson Grace, Regional
Administrator
U. S. Nuclear Regulatory Commission,
Region II
101 Marietta Street, N. W.
Suite 2900
Atlanta, Georgia 30303

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

DUKE POWER COMPANY

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 137
License No. DPR-38

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

   A. The application for amendment by Duke Power Company (the licensee) dated December 19, 1984, as supplemented on March 8, 1985, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 137 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 18, 1985

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

DUKE POWER COMPANY

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 137
License No. DPR-47

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

    A. The application for amendment by Duke Power Company (the licensee) dated December 19, 1984, as supplemented on March 8, 1985, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

    3.B Technical Specifications

    The Technical Specifications contained in Appendices A and B, as revised through Amendment No.137 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 18, 1985

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

DUKE POWER COMPANY

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 134
License No. DPR-55

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

   A. The application for amendment by Duke Power Company (the licensee) dated December 19, 1984, as supplemented on March 8, 1985, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 134 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 18, 1985

ATTACHMENTS TO LICENSE AMENDMENTS

AMENDMENT NO. 137 TO DPR-38

AMENDMENT NO. 137 TO DPR-47

AMENDMENT NO. 134 TO DPR-55

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

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

Remove Pages	Insert Pages
2.1-3a	2.1-3a
2.1-3b	2.1-3b
3.5-16 (3 pages)	3.5-16 (1 page)
3.5-19 (3 pages)	3.5-19 (1 page)
3.5-22 (3 pages)	3.5-22 (1 page)
3.5-25 (2 pages)	3.5-25 (1 page)
3.5-28 (1 page)	3.5-28 (1 page)

Bases - Unit 2

The safety limits presented for Oconee Unit 2 have been generated using the BAW-2 and BWC critical heat flux correlations[1,3] and the Reactor Coolant System flow rate of 106.5 percent of the design flow (design flow is 352,000 gpm for four-pump operation). The flow rate utilized is conservative compared to the actual measured flow rate[2].

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

The curve presented in Figure 2.1-1B represents the conditions at which a minimum allowable DNBR is predicted for the maximum possible thermal power (112 percent) when four reactor coolant pumps are operating (minimum reactor coolant flow is 374,880 gpm). This curve is based on the following nuclear power peaking factors with potential fuel densification and fuel rod bowing effects:

$$F_q^N = 2.565; \; F_{\Delta H}^N = 1.71^{(3)} F_z^N = 1.50$$

The design peaking combination results in a more conservative DNBR than any other power shape that exists during normal operation.

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

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

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limit is 20.15 kw/ft for fuel rod burnup less than or equal to 1,000 MWD/MTU and 21.2 kw/ft after 1,000 MWD/MTU.

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

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

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

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

The maximum thermal power for three-pump operation is 88.07 percent due to a power level trip produced by the flux-flow ratio 74.7 percent flow x 1.07 = 79.92 percent power plus the maximum calibration and instrument error. The maximum thermal power for other coolant pump conditions is produced in a similar manner.

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

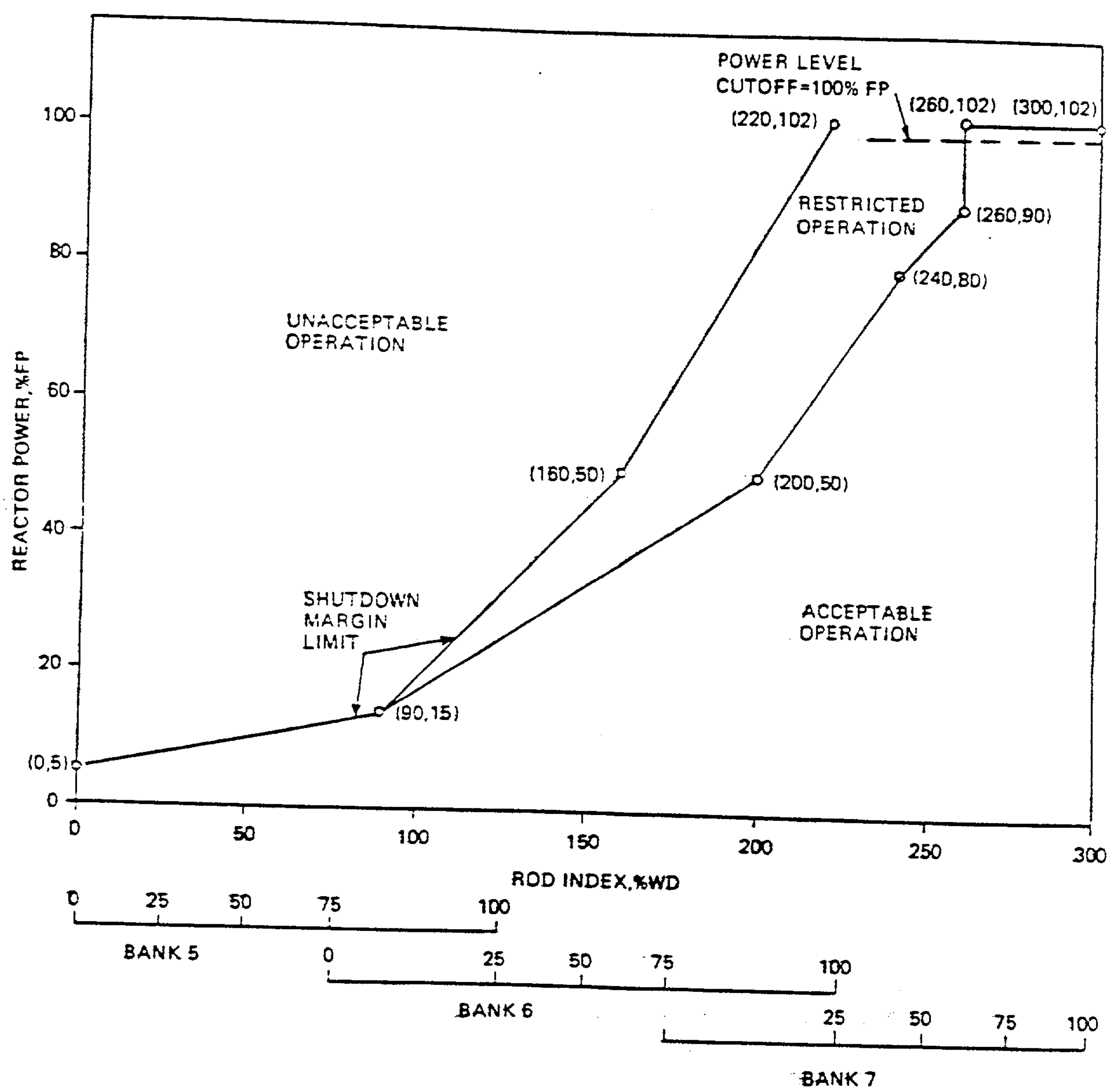
References

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

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

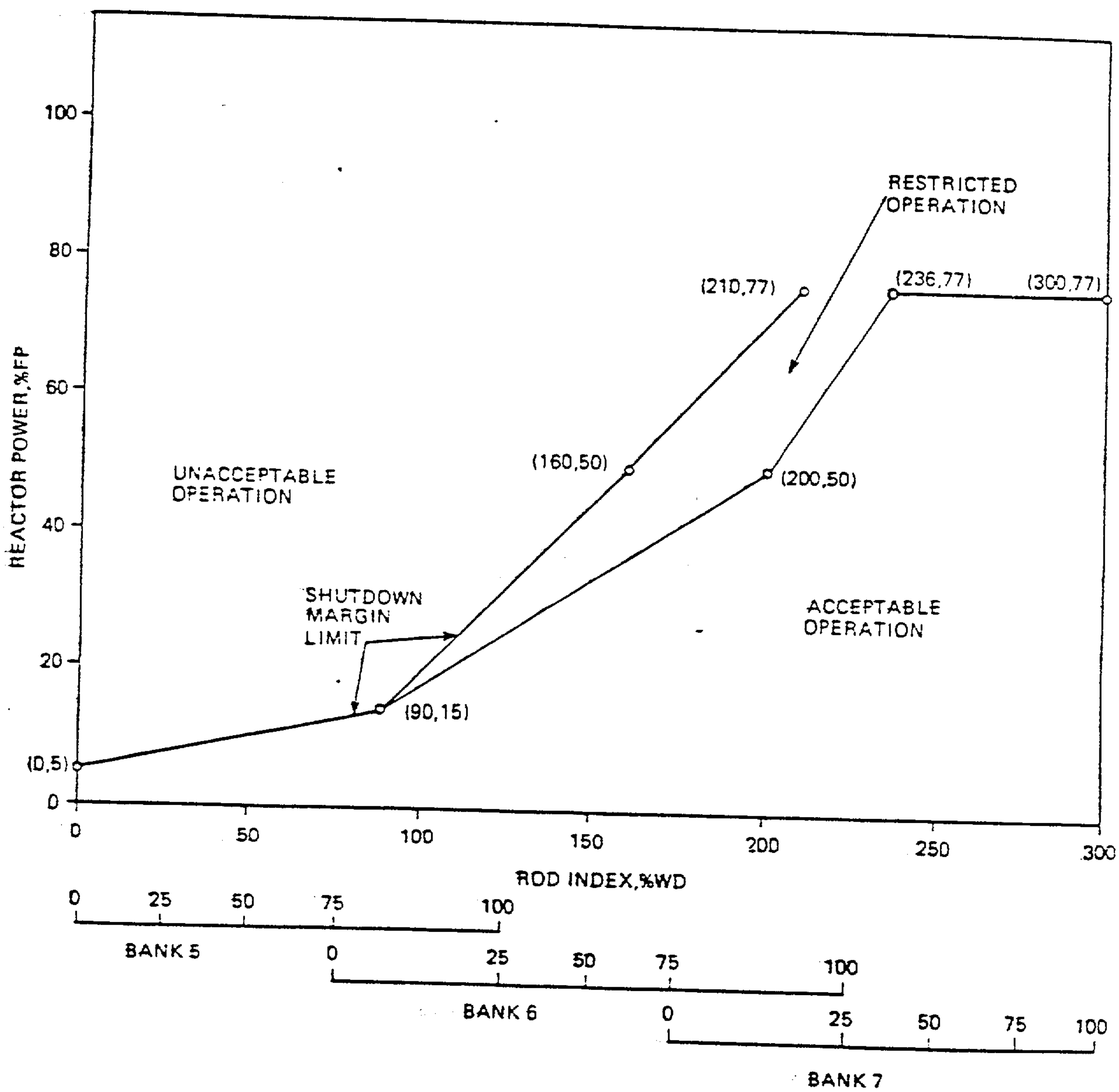
(3) Correlation of 15 x 15 Geometry Zircaloy Grid Rod Bundle CHF Data with the BWC Correlation, BAW-10143P, Part 2, Babcock & Wilcox, Lynchburg, Virginia, August 1981.

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



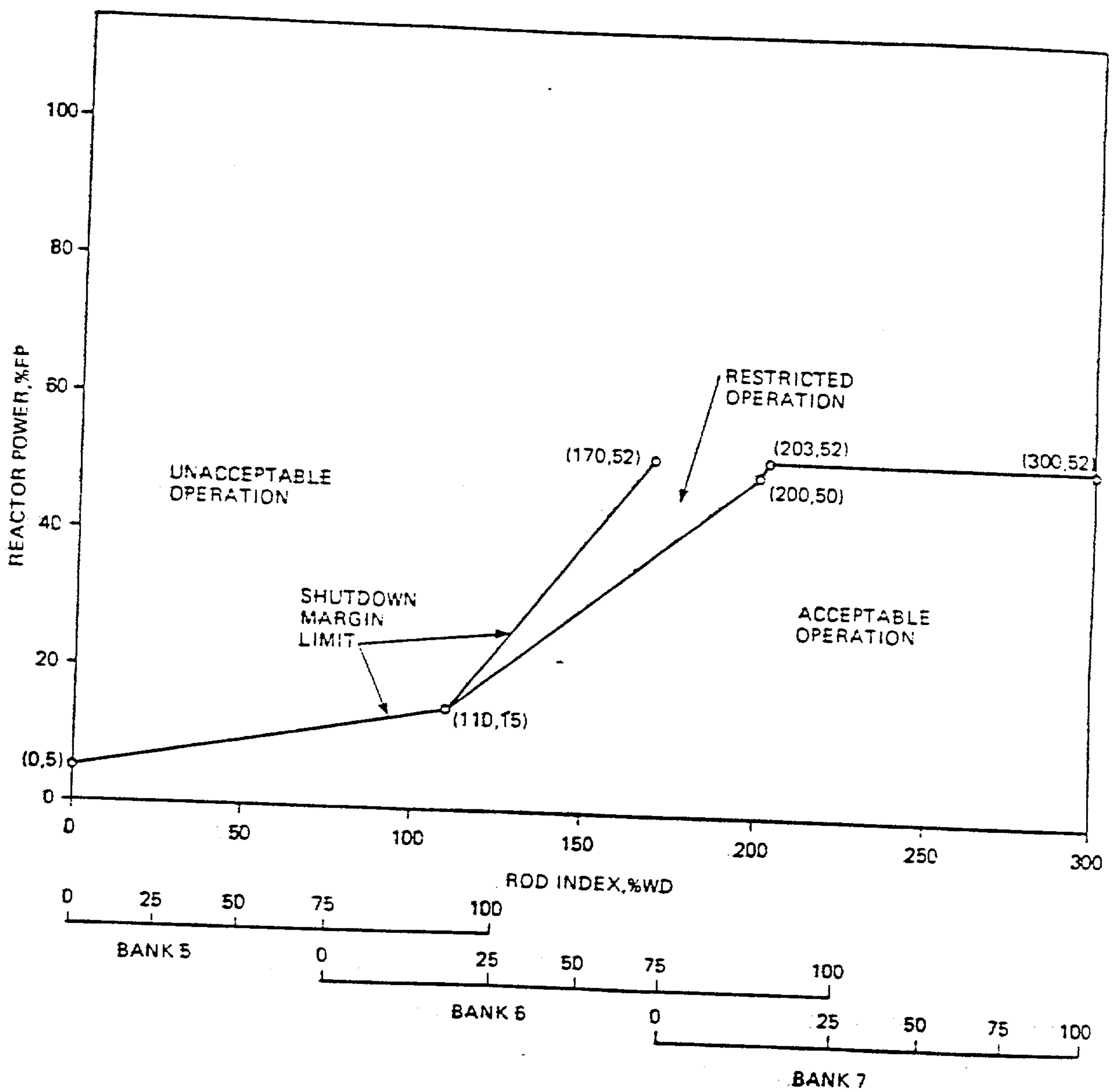
ROD POSITION LIMITS
FOR FOUR PUMP OPERATION
FROM 0 EFPD TO EOC
UNIT 2

OCONEE NUCLEAR STATION
Figure 3.5.2-2
(1 of 1)



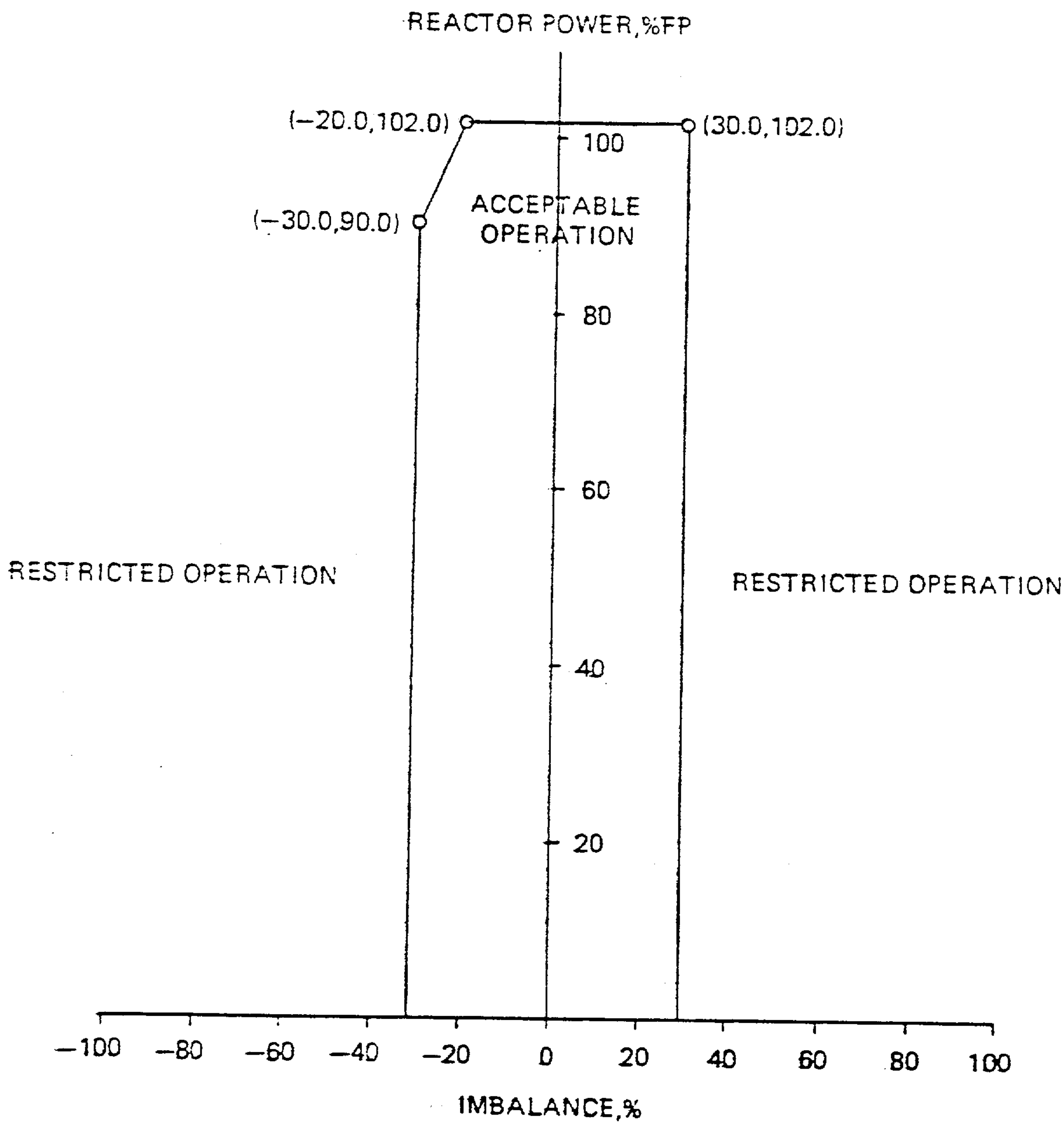
ROD POSITION LIMITS
FOR THREE PUMP OPERATION
FROM 0 EFPD TO EOC
UNIT 2

OCONEE NUCLEAR STATION
Figure 3.5.2-5
(1 of 1)



ROD POSITION LIMITS
FOR TWO PUMP OPERATION
FROM 0 EFPD TO EOC
UNIT 2

OCONEE NUCLEAR STATION
Figure 3.5.2-8
(1 of 1)



OPERATIONAL POWER
IMBALANCE ENVELOPE
FROM 0 EFPD TO EOC
UNIT 2

OCONEE NUCLEAR STATION
Figure 3.5.2-11
(1 of 1)

Figure 3.5.2-14
(Deleted)

[Note that no rod position limits exist for Unit 2 axial power shaping rods.]

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 137 TO FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 137 TO FACILITY OPERATING LICENSE NO. DPR-47

AMENDMENT NO. 134 TO FACILITY OPERATING LICENSE NO. DPR-55

DUKE POWER COMPANY

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

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

INTRODUCTION

By letter dated December 19, 1984 (Ref. 1), as supplemented with additional information on March 8, 1985 (Ref. 17), Duke Power Company (the licensee) proposed changes to the Technical Specifications (TSs) of Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units 1, 2 and 3. These amendments would consist of changes to the Station's common TSs.

These amendments would authorize proposed changes to the Oconee Nuclear Station TSs which are required to support the operation of Oconee Unit 2 at full rated power during the upcoming Cycle 8. The proposed amendments would change the following areas:

1. Rod Position Limits of TS 3.5.2, and
2. Power Imbalance Limits of TS 3.5.2.

To support the license amendment application, the licensee submitted a Duke Power Company report, DPC-RD-2004 (Ref. 2), "Oconee Unit 2, Cycle 8 Reload", as an attachment to Reference 1. A summary of the Cycle 8 operating parameters is included in the report, along with safety analyses.

The analytical methods used in the safety analysis of the proposed eighth cycle of operation at Oconee Unit 2 are described in the Duke Power Company Oconee Nuclear Station Reload Design Methodology Report (Ref. 3) which has been reviewed and approved by the NRC staff (Ref. 4). The methodology report relies on a number of analytical methods developed by the fuel vendor, Babcock and Wilcox (B&W), and some developed by the licensee. The methods used in the Cycle 8 analysis are unchanged from those described in the Methodology Report and have not received additional review for Cycle 8 operation. Also, where conditions are identical or limited by the analysis of a previous cycle of operation, the evaluation of that cycle continues to apply.

The Cycle 8 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 Cycle 8 core design differs from the design of the previous cycle in four areas: (1) the use of gray axial power shaping rods (APSRs); (2) a new set of eight full-length control rod assemblies (CRAs) to replace eight CRAs which have reached their design exposure limit; (3) a new group assignment of the control rods, and (4) the introduction of 68 Mark BZ design fuel assemblies. Cycle 8 is to have a length of approximately 400 effective full power days (EFPD) of operation.

As has been the case for Cycle 7, Cycle 8 will be operated in rods-out, feed-and-bleed mode. Specific aspects of the Oconee Unit 2 Cycle 8 reload are discussed in the following sections.

EVALUATION

1.0 Evaluation of the Fuel System Design

1.1 Fuel Assembly Mechanical Design

Although all batches in the Oconee Unit 2 Cycle 8 core will utilize the same B&W 15x15 fuel design, the Batch 8 and 9 assemblies will be of the Mark B4 fuel design and the Batch 10 assemblies will be of the Mark BZ design. Four regenerative neutron sources will be used in Mark B4 assemblies. The Mark B4 assembly design was reviewed and found acceptable for previous B&W 177 fuel assembly reloads and this conclusion also applies to Oconee Unit 2 Cycle 8.

The Mark BZ design is similar to the Mark B fuel assembly except that the six intermediate Inconel spacer grids have been replaced with Zircaloy grids. Four Mark BZ demonstration assemblies have been previously approved in Cycles 7 and 8 of Oconee Unit 1, and 68 Mark BZ assemblies have been approved in Cycle 9 of Oconee Unit 1. For Oconee Unit 2 Cycle 8 a significant portion of the core will contain Mark BZ fuel. The design (Ref. 5) of these assemblies has been reviewed and approved by the NRC staff (Ref. 6). However, in the Safety Evaluation Report on Asymmetric LOCA Loads for Oconee Units 1, 2 and 3 (Ref. 7), a measured critical load versus maximum spacer grid load and a resulting temperature rise of 12 degrees Farenheit (°F) in peak cladding temperature (PCT) due to a fully collapsed grid (41% of flow area reduction) in the core peripheral assemblies was identified for the Mark B fuel. At our request, and as required in the approval of the Mark BZ fuel (Ref. 5), the licensee has submitted a corresponding analysis for an Oconee mixed core and a pure Mark BZ fuel core considering the reduction in strength due to the use of Zircaloy grids (Ref. 8). The PCT increase for Mark BZ fuel was estimated on the basis of calculations performed for Mark B fuel. The Mark B PCT analysis assumed the maximum flow area reduction of 41% along the entire assembly. Since dynamic response analyses showed that the maximum horizontal impact forces and maximum flow area reduction occur on the two mid-height spacer grids for both the Mark B and Mark BZ assemblies and the calculated maximum flow area reduction for Mark BZ fuel was 37%, it has been concluded that the assumptions used in the Mark B analysis are conservative. It is believed that these conservatisms as well as the similarities in the grid geometry justify the estimation of the PCT increase for Mark BZ fuel on the basis of calculations performed for Mark B fuel. The PCT increase was evaluated for the racking failure mode only since this case resulted in a larger flow area reduction than the crushing failure mode. Since the maximum horizontal impact forces occur in a peripheral fuel assembly, any racking or crushing failures would also be observed there. For these reasons, the PCT analysis for the Mark B fuel is expected to bound the analysis for the Mark BZ fuel. The Mark BZ assemblies are, therefore, acceptable for use in Cycle 8 and future cycles.

Cycle 8 contains one Advanced Cladding Pathfinder (ACP) assembly from Cycle 7 containing 12 advanced cladding rods.

1.2 Fuel Rod Design

The Oconee Unit 2 Cycle 8 core contains both Mark B4 and Mark BZ fuel assemblies, and the fuel rods used in both of these assemblies are virtually identical. The results of the linear-heat-rate-to-melt analysis (Table 4.2 of Ref. 2) are the same for all batches in the Cycle 8 core.

1.2.1 Cladding Collapse

The licensee has stated that the cladding collapse time for the most limiting Cycle 8 assembly was conservatively determined to be greater than the maximum projected residence time for any Cycle 8 assembly. The creep collapse analysis used the CROV computer code (Ref. 9) using input conditions from TACO-2 (Ref.10) in a manner described in the Reload Methodology Report (Ref.3).

All of these methods have been reviewed and approved by the NRC staff. We conclude that cladding collapse has been appropriately considered for Cycle 8 operation.

1.2.2 Cladding Stress and Strain

The cladding stress and strain analyses for the Cycle 8 fuel designs, including the gray APSRs, are either bounded by conditions previously analyzed for Oconee Unit 2 or were analyzed specifically for Cycle 8 using methods and limits previously reviewed and approved by the NRC. We conclude that the analysis of cladding stress and strain has been appropriately considered for Cycle 8 operation.

1.2.3 Rod Internal Pressure

Section 4.2 of the Standard Review Plan (Ref.11) was issued as a source for acceptance criteria for the design bases and evaluation of the fuel system. Among those criteria which may affect the operation of the fuel rod is the internal pressure limit. The pressure criterion (SRP 4.2, Section II.A.1(f)) states that the fuel rod internal gas pressure should remain below normal system pressure during normal operation unless otherwise justified. Based on a TACO-2 analysis, the licensee has stated that the fuel rod internal pressure will not exceed nominal system pressure during normal operation for Cycle 8. We find this acceptable and conclude that the rod internal pressure limits have been adequately considered for Cycle 8 operation.

1.3 Fuel Thermal Design

There are no major changes in the physical characteristics of the Cycle 8 core which would result in altered thermal conditions. As pointed out in Section 1.2 of this report, the linear-heat-rate-to-melt for all batches in the Cycle 8 core is the same. The linear-heat-rate-to-melt capability was determined separately for Batches 8B, 9 and 10 using TACO-2. The centerline melt limits are generated at both low and high burnup conditions. These values have been incorporated into the proposed TSs, and we find them acceptable.

A combination of TAFY and TACO-2 analyses was used to generate the LOCA limits as described in Tables 7-2 and 7-3 of Reference 2. Three sets of bounding values for allowable LOCA peak linear heat rates are given as a function of core height. These limits apply during the periods approximately 0-25 EFPD, 25-65 EFPD and 65 EFPD to end-of-cycle. These limits have been incorporated into the TSs for Cycle 8 through the operating limits on rod index and axial power imbalance. We conclude that the initial thermal conditions for LOCA analysis have been appropriately considered for Cycle 8 operation.

1.4 Conclusions

We have reviewed the sections of the reload report for Oconee Unit 2 Cycle 8 dealing with the fuel system design and find these portions of the application acceptable.

2.0 Evaluation of the Nuclear Design

The nuclear design parameters characterizing the operation of Oconee Unit 2 Cycle 8 have been obtained with the Duke Power physics calculational methods (Ref. 3). These methods have been approved for use in reload design calculations (Ref. 4) and were used previously in deriving the Cycle 7 nuclear design parameters. The Cycle 8 core will contain 68 fresh assemblies of the Mark BZ design with a Uranium-235 (U-235) initial enrichment of 3.22%. In addition to the 68 fresh assemblies, there are two batches of exposed assemblies: a batch of 37 assemblies having an initial U-235 enrichment of 3.17% and a batch of 72 assemblies having an initial enrichment of 3.24%. Four fresh assemblies are located in the central core region with the remaining fresh assemblies distributed in a checkerboard pattern in the surrounding annular region. The excess reactivity is controlled by soluble boron which is supplemented by 61 full-length silver-indium-cadmium (Ag-In-Cd) control rods and 60 burnable poison rod assemblies (BRPAs). In addition to the full length control rods, eight Inconel axial power shaping rods (APSRs) are provided for additional control of the axial power distribution. These less absorbing (gray) APSRs are longer than the highly absorbing (black) Ag-In-Cd APSRs they are replacing. Due to the presence of gray APSRs and to reduce the axial offset response to group 7 rod movement, there are now eight control rods in group 7 and twelve in group 5. This control rod group rearrangement differs from previous cycles in which group 5 and group 7 included 8 and 12 control rods, respectively.

Verification that the gray APSRs provide adequate axial power distribution control was made with the approved three-dimensional model. All safety criteria are satisfied. Shutdown margin values at beginning and end of cycle are 4.09% and 2.91% k/k, respectively, compared to the minimum required value of 1.0% k/k. Beginning of cycle radial power distributions show acceptable margins to limits.

2.1 Conclusions

Based on our review of the nuclear design, we conclude that approved methods have been used, that the nuclear design parameters meet the applicable criteria and that the nuclear design of Cycle 8 is acceptable.

3.0 <u>Evaluation of the Thermal-Hydraulic Design</u>

The objective of this review is to confirm that the thermal-hydraulic design of the reload core has been accomplished using acceptable methods and provides an acceptable margin of safety from conditions which could lead to fuel damage during normal and anticipated operational transients. The reload design methodology is described in Reference 3 and has been approved (Ref. 4). Discussion of the main features of the thermal-hydraulic design affected by the Cycle 8 reload follows.

3.1 Decrease in Core Bypass Flow

The incoming Batch 10 fuel consists of 68 fresh Mark BZ fuel assemblies. For Cycle 8 operation, 60 BPRAs will be inserted and two assemblies will contain regenerative neutron sources, leaving 46 open assemblies with open guide tubes. This will result in a core bypass flow of 7.9% compared to 7.8% for Cycle 7. The bypass flow of 7.9% is less than the 8.0% assumed in the generic thermal-hydraulic design analysis. The smaller core flow assumed in the generic analysis establishes it as conservative for Cycle 8 operation.

3.2 DNBR Performance in the Cycle 8 Transition Core

The Mark BZ fuel has higher hydraulic resistance compared to Mark B fuel as a result of the differences between the two spacer grid designs. In fuel bundle flow distribution tests, the pressure drop across a Mark BZ assembly was found to be less than 3% greater than across a Mark B assembly (Ref. 6). The presence of 68 Mark BZ assemblies in a transition core containing 109 Mark B assemblies results in a decrease in coolant flow in the Mark BZ fuel compared to that in an all Mark BZ core. The Mark B hot channel, however, receives more coolant flow and yields better departure from nucleate boiling (DNB) performance in the transition core than in a full Mark B core. Thus, for the Mark B fuel, the generic Mark B analyses, based on the B&W-2 critical heat flux (CHF) correlation (Ref. 12), are bounding and conservative for the transition core.

The thermal margin in the Mark BZ fuel was calculated using the BWC (Ref. 13) correlation for Mark BZ fuel with a minimum acceptable DNBR of 1.18. Using a flux-to-flow setpoint of 1.07 and a total radial peaking factor of 1.71, the licensee has determined a minimum DNBR greater than 1.18 for the Mark BZ fuel in the Cycle 8 flux-to-flow setpoint analysis. We conclude, therfore, that sufficient margin to DNB has been demonstrated for both Mark B and Mark BZ fuel in the Cycle 8 transition core.

A B&W Topical Report (Ref. 14) discussing the mechanisms for bowing in B&W fuel and its consequences has been reviewed and approved by the NRC staff (Ref. 15). The report concludes that a DNBR rod bow penalty need not be imposed since the power production capability of the fuel decreases sufficiently with irradiation to offset the effects of bowing. We conclude, therefore, that no rod bow penalty need be considered for Cycle 8 operation.

3.3 Conclusions

The pertinent thermal-hydraulic parameters summarized in Table 6-1 of the reload submittal (Ref.2) for Cycle 7 and Cycle 8 operation are identical except for (1) the 7.9% bypass flow in Cycle 8 compared to 7.8% in Cycle 7 (and 8.0% in the generic analysis), (2) a 0.97 hot channel flow area factor assumed in the design analysis of the Mark BZ fuel compared to 0.98 for the Mark B fuel in Cycle 7 and Cycle 8, (3) the use of the BWC CHF correlation in the Mark BZ fuel thermal-hydraulic analysis in contrast to the use of the BAW-2 CHF correlation with the Mark B fuel, and (4) a design minimum DNBR value in excess of 1.74 for the Mark BZ fuel, compared to a value of 2.05 for the Mark B fuel. As discussed in Section 3.1, the Cycle 8 value for bypass flow (item-1) implies that the Cycle 8 analysis was performed using a value of core coolant flow rate that is conservative compared to the generic value. The Mark BZ hot channel flow area factor (item-2) leads to additional conservatism in the thermal-hydraulic analysis of the Mark BZ fuel. The use of the BWC CHF correlation for the Mark BZ fuel (item-3) is appropriate, and the corresponding calculated minimum DNBR with densification penalty (item-4) is in excess of the minimum allowable DNBR.

We conclude from the examination of the Cycle 8 core thermal-hydraulic design that the core reload will not adversely affect the capability to operate Oconee Unit 2 safely during Cycle 8.

4.0 <u>Accident Analyses</u>

The important kinetics parameters for Cycle 8 are compared to the values used in the Final Safety Analysis Report (FSAR) in Table 7.1 of the reload submittal (Ref. 2). For the parameters included, the Cycle 8 values are bounded by those used previously. The licensee has also determined that the initial conditions of the transients in Cycle 8 are bounded by the FSAR and/or the fuel densification report (Ref. 16). Since the Batch 10 reload fuel contains rods with a theoretical density higher than those considered in the densification report, the conclusions in Reference 16 are still valid. These analyses have been previously accepted by the NRC staff.

The licensee's Reload Methodology Technical Report (Ref. 3), which has been accepted by the NRC staff (Ref. 4), was examined vis-a-vis the Accident Analysis Review process. Virtually all of the items contained in the Key

Safety Parameter Checklist (Table 8-1, Ref. 3) are addressed in Table 7.1 and other tables in the submittal. The Minimum Tripped Rod Worth available in case of a steamline break is given in Reference 17. The quoted Cycle 8 values are all bounded by the values assumed in previous accident analyses, except for the delayed neutron fractions at BOC and EOC (Table 5-1) which are lower than the nominal values assumed in the FSAR analysis of the rod ejection accident (REA). While this would tend to increase the maximum fuel enthalpy associated with a postulated REA event, the maximum ejected rod worth at HFP for Cycle 8 is so much lower than that assumed in the FSAR analysis that it offsets this nonconservatism.

Three sets of bounding values for allowable LOCA peak linear heat rates are given as a function of core height. These limits apply during the periods 0 to 1000 MWD/MTU, 1000 to 2600 MWD/MTU, and for the balance of the cycle. These results are based upon a bounding analytical assessment of NUREG-0630 on LOCA and operating kw/ft limits performed by Babcock and Wilcox (Ref. 18). The B&W analyses have been approved by the NRC staff and the three sets of limits were accepted in conjunction with the review of the Oconee Unit 2 Cycle 7 reload submittal (Refs. 19 and 20).

The LOCA limit at the 6 foot elevation is reduced in the present reload, relative to the previous value, for the 68 fresh Mark BZ assemblies. The lower limit is due to the material and geometrical differences of the new fuel relative to the standard Mark-B assemblies, and applies only for the first 1000 MWD/MTU (approximately 25 EFPD).

New dose calculations were not performed for Oconee Unit 2 Cycle 8. The licensee has determined that the dose considerations for Oconee Unit 1 Cycle 9 (Ref. 21) are applicable to Oconee Unit 2 Cycle 8, based on comparisons of key parameters which determine radionuclide inventories.

5.0 Technical Specification Modifications

Oconee Unit 2 Cycle 8 TSs have been modified to account for changes in power peaking and control rod worths, the replacement of black (Ag-In-Cd) APSRs with gray (Inconel) APSRs, the use of the BWC CHF correlation, and the elimination of the DNBR rod bow penalty. We have reviewed the proposed revisions to the TSs for Cycle 8. These changes concern the Rod Position Limits and Operational Power Imbalance Envelope of TS 3.5.2. On the basis that approved methods were used to obtain these limits, we find these TS modifications acceptable. Rod Position Limits and an Operational Power Imbalance envelope have been determined that apply to the entire Cycle 8 operation. These were obtained by performing the requisite analyses at selected exposures spanning the cycle and determining the most restrictive Rod Position Limits and Operational Power Imbalance Envelope. Since the analyses span the entire cycle of operation, we find the corresponding TSs applicable to the entire cycle.

6.0 Evaluation Findings

We have reviewed the fuels, physics, thermal-hydraulic and accident analysis information presented in the Oconee Unit 2 Cycle 8 reload report as stated above. We find the proposed reload and the associated modified TSs acceptable.

ENVIRONMENTAL CONSIDERATION

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

CONCLUSION

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

REFERENCES

1. Letter from H. B. Tucker (DPC) to H. R. Denton (NRC) "Oconee Nuclear Station Unit 2, Docket No. 50-270", December 19, 1984.

2. "Oconee Unit 2 Cycle 8 Reload Report, "DPC-RD-2004, Duke Power Company, December 1984.

3. "Oconee Nuclear Station Reload Design Methodology Technical Report," NFS-1001, Revision 4, Duke Power Company, Charlotte, North Carolina, April 1979.

4. Letter from P. C. Wagner (NRC) to W. B. Parker, Jr. (DPC), "Safety Evaluation by the Office of Nuclear Reactor Regulation of the Reload Design Methodology Technical Report NFS-1001," July 29, 1981.

5. "Rancho Seco Cycle 7 Reload Report, Volume-1, Mark BZ Fuel Assembly Design Report," BAW-1781P, Babcock & Wilcox, April 1983.

6. Letter from H. Nicolaras (NRC) to H. B. Tucker (DPC), "Safety Evaluation by the Office of Nuclear Reactor Regulation of Mark BZ Fuel Assembly Design and Oconee Unit 1 Cycle 9 Reload," USNRC, November 23, 1984.

7. Letter from J. F. Stolz (NRC) to H. B. Tucker (DPC), "Safety Evaluation Report on Asymmetric LOCA Loads - Oconee Units 1, 2 and 3," May 20, 1983.

8. Letter from H. B. Tucker (DPC) to J. Stolz (NRC), "Mark BZ Fuel Assembly Supplemental Information," October 22, 1984.

9. A. F. J. Eckert, H. W. Wilson, K. E. Yoon, "Program to Determine In-Reactor Performance of B&W Fuels: Cladding Creep Collapse," Babcock and Wilcox Company Report, BAW-10084P-A, Revision 2, October 1978.

10. "TACO2 - Fuel Pin Performance Analysis, " BAW-10141P-A, Rev. 1, Babcock & Wilcox, Lynchburg, Virginia, June 1983.

11. U. S. Nuclear Regulatory Commission Standard Review Plan, Section 4.2, "Fuel System Design," U. S. Nuclear Regulatory Commission Report NUREG-0800 (Formerly NUREG-75/087), Revision 2, July 1981.

12. "Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water," BAW-10000, Babcock & Wilcox, March 1970.

13. "Correlation of 15 x 15 Geometry Zircaloy Grid Rod Bundle CHF Data with the BWC Correlation," BAW 10143P, Part 2, Babcock & Wilcox, August 1981.

14. "Fuel Rod Bowing in Babcock & Wilcox Fuel Designs," BAW-10147P-A, Rev. 1, Babcock & Wilcox, May 1983.

15. Letter from Cecil O. Thomas (NRC) to James H. Taylor (B&W), "Acceptance for the Referencing of Licensing Topical Report BAW-10147(P)," February 15, 1983.

16. "Oconee 2 Fuel Densification Report," BAW-1395, Babcock & Wilcox, June 1973.

17. Letter from H. B. Tucker (DPC) to H. R. Denton (NRC), "Supplemental Information in Support of the Oconee-2 Cycle 8 Reload Report," dated March 8, 1985.

18. "Bounding Analytical Assessment of NUREG-0630 on LOCA and Operating kw/ft Limits," B&W Document No. 77-1141256-00, Babcock & Wilcox.

19. "Oconee Unit 2, Cycle 7 Reload Report," DPC-RD-2002, Duke Power Company, September 1983.

20. Letter from J. F. Stolz (NRC) to H. B. Tucker (DPC), "Safety Evaluation by the Office of Nuclear Reactor Regulation of the Oconee Unit 2 Reload for Cycle 7" November 23, 1983.

21. "Oconee Unit 1, Cycle 9 Reload Report," BAW-1841, Babcock & Wilcox, August 1984.

Dated: April 18, 1985

Principal Contributor: M. Dunenfeld