aDCS-
MS-016

JUL 21 1981

Docket No. 50-368

Mr. William Cavanaugh, III
Senior Vice President, Energy
Supply Department
Arkansas Power & Light Company
P. O. Box 551
Little Rock, Arkansas 72203

Dear Mr. Cavanaugh:

SUBJECT: OPERATION OF ANO-2 DURING CYCLE 2

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 26 to Facility Operating License No. NPF-6 for the Arkansas Power and Light Company for the Arkansas Nuclear One, Unit 2 plant. The amendment consists of changes to the license in accordance with the satisfactory completion of those issues which previously required limiting the authorized level to 70% of the full power rating of 2815 MWt. Operation up to 70% of 2815 MWt was authorized in Amendment No. 24 issued June 19, 1981. It also consists of changes to the Technical Specifications (TS) consistent with the resolution of these issues and other TS changes made in accordance with the ANO-2 Cycle 2 Reload Report and request dated February 20 and March 5, 1981 as supplemented.

With the issuance of Amendment No. 24 the Commission authorized operation up to 70% of the full power level of 2815 MWt pending resolution of the remaining details of the staff's review of the changes proposed for the Core Protection Calculator System software for Cycle 2 operation. These matters were discussed in Section 2.3 of the Safety Evaluation accompanying Amendment No. 24. The staff's review of these matters has now been completed. The staff's evaluation of these issues, supporting the authorization to operate at 100% of 2815 MWt, is included in Section 2.3 of the enclosed Safety Evaluation.

CP1

OFFICE ▶							
SURNAME ▶							
DATE ▶							

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

Sincerely,

Original signed by
Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No.?6 to NPF-6
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
See next page

DISTRIBUTION:

Docket File	OELD
NRC PDR	I&E (4)
L PDR	GDeegan (4)
TERA	BScharf (10)
NSIC	JWetmore
ORB#3 Rdg	ACRS (10)
DEisenhut	CMiles
RDiggs	JHeltemes
PMKreutzer (3)	Gray File (+4)
REMartin	
RAClark	

Concur in license amendment (correct typo noted by paper clip) and Fed Reg. notice.

OFFICE	ORB#3;DL	ORB#3;DL	ORB#3;DL	AD:OR:DL	OELD		
SURNAME	PMKreutzer	REMartin	RAClark	TMNovak	M. Rothschild		
DATE	7/13/81	7/1/81 7/21/81	7/21/81	7/16/81	7/16/81		

NRC FORM 318 (10-80) NRCM 0240 **OFFICIAL RECORD COPY** USGPO: 1981—335-960

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

July 21, 1981

Docket No. 50-368

Mr. William Cavanaugh, III
Senior Vice President, Energy
Supply Department
Arkansas Power & Light Company
P. O. Box 551
Little Rock, Arkansas 72203

Dear Mr. Cavanaugh:

SUBJECT: OPERATION OF ANO-2 DURING CYCLE 2

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 26 to Facility Operating License No. NPF-6 for the Arkansas Power and Light Company for the Arkansas Nuclear One, Unit 2 plant. The amendment consists of changes to the license in accordance with the satisfactory completion of those issues which previously required limiting the authorized level to 70% of the full power rating of 2815 MWt. Operation up to 70% of 2815 MWt was authorized in Amendment No. 24 issued June 19, 1981. It also consists of changes to the Technical Specifications (TS) consistent with the resolution of these issues and other TS changes made in accordance with the ANO-2 Cycle 2 Reload Report and request dated February 20 and March 5, 1981 as supplemented.

With the issuance of Amendment No. 24 the Commission authorized operation up to 70% of the full power level of 2815 MWt pending resolution of the remaining details of the staff's review of the changes proposed for the Core Protection Calculator System software for Cycle 2 operation. These matters were discussed in Section 2.3 of the Safety Evaluation accompanying Amendment No. 24. The staff's review of these matters has now been completed. The staff's evaluation of these issues, supporting the authorization to operate at 100% of 2815 MWt, is included in Section 2.3 of the enclosed Safety Evaluation.

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

Sincerely,

Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 26 to NPF-6
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
See next page

Arkansas Power & Light Company

cc:

Mr. David C. Trimble
Manager, Licensing
Arkansas Power & Light Company
P. O. Box 551
Little Rock, Arkansas 72203

Mr. James P. O'Hanlon
General Manager
Arkansas Nuclear One
P. O. Box 608
Russellville, Arkansas 72801

Mr. Robert B. Borsum
Babcock & Wilcox
Nuclear Power Generation Division
Suite 420
7735 Old Georgetown Road
Bethesda, Maryland 20014

Nick Reynolds
c/o DeBevoise & Liberman
1200 Seventeenth Street, N.W.
Washington, D. C. 20036

Arkansas Polytechnic College
Russellville, Arkansas 72801

Mr. Charles B. Brinkman
Manager - Washington Nuclear
Operations
C-E Power Systems
4853 Cordell Avenue, Suite A-1
Bethesda, Maryland 20014

Honorable Ermil Grant
Acting County Judge of Pope County
Pope County Courthouse
Russellville, Arkansas 72801

U.S. Environmental Protection Agency
Region VI Office
ATTN: EIS COORDINATOR
1201 Elm Street
First International Building
Dallas, Texas 75270

cc w/enclosure(s) and incoming
dated: 2/20/81, 3/5/81

Director, Bureau of Environmental
Health Services
4815 West Markham Street
Little Rock, Arkansas 72201

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

ARKANSAS POWER AND LIGHT COMPANY

DOCKET NO. 50-368

ARKANSAS NUCLEAR ONE, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 26
License No. NPF-6

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

   A. The application for amendment by Arkansas Power and Light Company (the licensee) dated February 20 and March 5, 1981, as supplemented, 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, Facility Operating License No. NPF-6 is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and by revising Paragraphs 2.C.(1) and 2.C.(2) to read as follows:

   (1) Maximum Power Level

   The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2815 megawatts thermal. Prior to attaining this power level the licensee shall comply with the conditions in Paragraph 2.C.(3).

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 26, 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

Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the
Technical Specifications

Date of Issuance: July 21, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 26

FACILITY OPERATING LICENSE NO. NPF-6

DOCKET NO. 50-368

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Pages

2-6
3/4 2-8
B3/4 2-3
6-13

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Linear Power Level - High		
a. Four Reactor Coolant Pumps Operating	≤ 110% of RATED THERMAL POWER	≤ 110.712% of RATED THERMAL POWER
b. Three Reactor Coolant Pumps Operating	*	*
c. Two Reactor Coolant Pumps Operating - Same Loop	*	*
d. Two Reactor Coolant Pumps Operating - Opposite Loops	*	*
3. Logarithmic Power Level - High (1)	≤ 0.75% of RATED THERMAL POWER	≤ 0.819% of RATED THERMAL POWER
4. Pressurizer Pressure - High	≤ 2362 psia	≤ 2370.887 psia
5. Pressurizer Pressure - Low	≥ 1766 psia (2)	≥ 1712.757 psia (2)
6. Containment Pressure - High	≤ 18.4 psia	≤ 19.024 psia
7. Steam Generator Pressure - Low	≥ 751 psia (3)	≥ 729.613 psia (3)
8. Steam Generator Level - Low	≥ 46.7% (4)	≥ 45.811% (4)

* These values left blank pending NRC approval of safety analyses for operation with less than four reactor coolant pumps operating.

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
9. Local Power Density - High	$\leq$ 20.3 kw/ft (5)	$\leq$ 20.3 kw/ft (5)
10. DNBR - Low	$\geq$ 1.24 (5)(6)(7)	$\geq$ 1.24 (5)(6)(7)
11. Steam Generator Level - High	$\leq$ 93.7% (4)	$\leq$ 94.589% (4)

TABLE NOTATION

(1) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\leq 10^{-4}$ of RATED THERMAL POWER.

(2) Value may be decreased manually, to a minimum value of 100 psia, during a planned reduction in pressurizer pressure, provided the margin between the pressurizer pressure and this value is maintained at $\leq$ 200 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is $\geq$ 500 psia.

(3) Value may be decreased manually during a planned reduction in steam generator pressure provided the margin between the steam generator pressure and this value is maintained at $\leq$ 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and lower level instrument nozzles.

(5) As stored within the Core Protection Calculator (CPC). Calculation of the trip setpoint includes measurement, calculational and processor uncertainties, and dynamic allowances. Trip may be manually bypassed below 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq 10^{-4}$% of RATED THERMAL POWER.

(6) The minimum allowable value of the addressable constant BERR1 in each OPERABLE channel is 1.086.

(7) The approved SCU equivalent DNBR limit is 1.26 which includes a two percent rod bow compensation. A DNBR trip setpoint of 1.24 is allowed provided that the difference is compensated by an increase of the addressable constant BERR1 to a minimum allowable value of 1.065.

DNBR MARGIN

LIMITING CONDITION FOR OPERATION

3.2.4 The DNBR margin shall be maintained by operating within the region of acceptable operation of Figure 3.2-3 or 3.2-4, as applicable.

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

With operation outside of the region of acceptable operation, as indicated by either (1) the COLSS calculated core power exceeding the COLSS calculated core power operating limit based on DNBR; or (2) when the COLSS is not being used, any OPERABLE Low DNBR channel exceeding the DNBR limit, within 15 minutes initiate corrective action to reduce the DNBR to within the limits and either:

a. Restore the DNBR to within its limits within one hour, or

b. Be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The DNBR shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the DNBR, as indicated on all OPERABLE DNBR channels, is within the limit shown on Figure 3.2-3.

4.2.4.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on DNBR.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.4.4 The following DNBR penalty factors shall be verified to be included in the COLSS and CPC DNBR calculations at least once per 31 days:

Burnup $(\frac{GWD}{MTU})$	DNBR Penalty (%)
0-3.1	0
3.1-5	2.0
5-10	5.9
10-15	8.8
15-20	11.4
20-25	13.6
25-30	15.6
30-35	17.4

The penalty for each batch will be determined from the batch's maximum burnup assembly and applied to the batch's maximum radial power peak assembly. A single net penalty for COLSS and CPC will be determined from the penalties associated with each batch, accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches. An alternate method is to determine the penalty for each individual assembly in the core based on that assembly's burnup and apply that penalty to that assembly's radial power peak.

P_{tilt}/P_{untilt} is the ratio of the power at a core location in the presence of a tilt to the power at that location with no tilt.

3/4.2.4 DNBR MARGIN

The limitation on DNBR as a function of AXIAL SHAPE INDEX represents a conservative envelope of operating conditions consistent with the safety analysis assumptions and which have been analytically demonstrated adequate to maintain an acceptable minimum DNBR throughout all anticipated operational occurrences, of which the loss of flow transient is the most limiting. Operation of the core with a DNBR at or above this limit provides assurance that an acceptable minimum DNBR will be maintained in the event of a loss of flow transient.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the DNBR channels in the Core Protection Calculators (CPCs), provide adequate monitoring of the core power distribution and are capable of verifying that the DNBR does not violate its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core operating limit corresponding to the allowable minimum DNBR. Reactor operation at or below this calculated power level assures that the limits of Figure 3.2-3 are not violated. The COLSS calculation of core power operating limit based on DNBR includes appropriate uncertainty and penalty factors necessary to provide a 95/95 confidence level that the core power at which a DNBR of less than 1.24 could occur, as calculated by COLSS, is less than or euqal to that which would actually be required in the core. To ensure that the design margin to safety is maintained, the COLSS computer program includes an F_{xy} measurement uncertainty factor of 1.053, an engineering uncertainty factor of 1.03, a THERMAL POWER measurement uncertainty factor of 1.02 and appropriate uncertainty and penalty factors for flux peaking augmentation and rod bow.

Parameters required to maintain the margin to DNB and total core power are also monitored by the CPCs. Therefore, in the event that the COLSS is not being used, operation within the limits of Figure 3.2-4 can be maintained by utilizing a predetermined DNBR as a function of AXIAL SHAPE INDEX and by monitoring the CPC trip channels. The above listed uncertainty and penalty factors are also included in the CPC.

The DNBR penalty factors listed in section 4.2.4.4 are penalties used to accommodate the effects of rod bow. The amount of rod bow in each individual fuel assembly is dependent upon the burnup experienced by that assembly. Higher burnup assemblies will experience a higher degree of rod bow and should be assigned a higher penalty factor. Conversely, low burnup assemblies will experience a lesser degree of rod bow and should be assigned a lower penalty factor.

3/4.2.5 RCS FLOW RATE

This specification is provided to ensure that the actual RCS total flow rate is maintained at or above the minimum value used in the LOCA safety analyses.

3/4.2.6 REACTOR COOLANT COLD LEG TEMPERATURE

This specification is provided to ensure that the actual value of reactor coolant cold leg temperature is maintained within the range of values used in the safety analyses.

3/4.2.7 AXIAL SHAPE INDEX

This specification is provided to ensure that the actual value of AXIAL SHAPE INDEX is maintained within the range of values used in the safety analyses.

3/4.2.8 PRESSURIZER PRESSURE

This specification is provided to ensure that the actual value of pressurizer pressure is maintained within the range of values used in the safety analyses.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The unit shall be placed in at least HOT STANDBY within one hour.

b. The Safety Limit violation shall be reported to the Commission, the Assistant Vice-President, Nuclear Operations, and to the SRC within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PSC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the SRC and the Assistant Vice-President, Nuclear Operations, within 14 days of the violation.

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of of Regulatory Guide 1.33, Revision 2, February 1978.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

g. Modification of Core Protection Calculator (CPC) Addressable Constants
NOTE: Modification to the CPC addressable constants based on information obtained through the Plant Computer - CPC data link shall not be made without prior approval of the Plant Safety Committee.

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be reviewed by the PSC and approved by the General Manager prior to implementation and reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the PSC and approved by the General Manager within 14 days of implementation.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS AND REPORTABLE OCCURRENCES

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Director of the Regional Office of Inspection and Enforcement unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO.24 TO FACILITY OPERATING LICENSE NO. NPF-6

ARKANSAS POWER AND LIGHT COMPANY

ARKANSAS NUCLEAR ONE, UNIT NO. 2

DOCKET NO. 50-368

Table of Contents

1.0 Introduction

In Amendment No. 24, issued on June 19, 1981, we reported the results of our review of the licensee's Cycle 2 Reload Report. We concluded that the licensee's proposed bases for operation at full power during Cycle 2 were acceptable with the exception of some aspects of the Core Protection Calculator System (CPCS) review.

With respect to these subjects we determined in Amendment No. 24 that insufficient time was available to complete all details of the review prior to the scheduled attainment of core criticality and startup operations for Cycle 2 operation. The incomplete review subjects were: (1) the CE-1 DNBR correlation, (2) the CETOP-D code, (3) the CETOP-2 code, and (4) the statistical combination of uncertainties (SCU) methodology.

Our concerns relating to the four subjects listed above were with respect to whether or not sufficient margins had been represented in the changes to the CPCS to account for the uncertainties associated with these subjects. The staff determined that the effect of all of the changes made to the CPCS for Cycle 2 operation provided a total overpower thermal margin gain on the order of about fifteen percent. Therefore, pending the completion and reporting of the results of our review we imposed in Amendment No. 24 a thirty percent power margin by temporarily limiting operation of the plant to seventy percent of the licensed full power level of 2815 MWt. We considered that the thirty percent power margin was sufficient to account for uncertainties while we completed the remaining details of our review. Neither this SE and license amendment nor Amendment No. 24 involves a change in the 100% power level of 2815 MWt which was authorized by Amendment No. 1 dated September 1, 1978.

We have now completed our review of the four subjects listed above and have applied the results to the Technical Specifications governing operation of the plant up to and including 100 percent of 2815 MWt. Therefore, the thirty percent power margin imposed by restricting operations to seventy percent of 2815 in Amendment No. 24 is no longer needed and is removed by the issuance of this amendment.

The information in Sections 2.3.1 through 2.3.5 of the safety evaluation accompanying Amendemnt No. 24 is superseded by this safety evaluation.

2.0 Discussion and Evaluation

2.1 Thermal-Hydraulic Design

By letters dated February 20 and March 5, 1981 (Refs. 1 and 2) Arkansas Power and Light Company (AP&L), the licensee, has provided the reload reports and proposed modifications to Technical Specifications for Arkansas Nuclear One, Unit 2 (ANO-2), Cycle 2 reload review. These reports include the safety analyses for those transients which required reanalysis, a comparison of the Cycle 2 thermal hydraulic parameters at full power with those of Cycle 1, and the proposed modifications to the Technical Specifications due to changes in methodology. In addition, AP&L submitted the following reports describing the methodology changes for ANO-2 Cycle 2 reload review:

(a) The CETOP-D Core Thermal Margin Design Code (Ref. 3)

This code replaces the COSMO code used in ANO-2 Cycle 1 analysis.

(b) CE-1 Critical Heat Flux (CHF) Correlation (Refs. 4 and 5), Generic DNBR Limit.

This correlation replaces the W-3 correlation used in ANO-2 Cycle 1 DNBR analysis.

(c) Effects of Fuel Rod Bow on DNBR Margin (Ref. 6)

Proposed modifications on the effects of fuel rod bow on DNBR to the ANO-2 Cycle 1 are described in this report. This report is under review by the staff and is scheduled for completion in November 1981.

(d) Statistical Combination of Uncertainties (Ref. 7)

CE's thermal margin methodology for ANO-2 Cycle 2 has been modified by the application of statistical methods instead of the application of deterministic methods applied in ANO-2 Cycle 1.

(e) CPC/CEAC Software Modifications (Ref. 8)

The Core Protection Calculators (CPC) and Control Element Assembly Calculators (CEAC) software for ANO-2 Cycle 2 has been modified as compared to the software for ANO-2 Cycle 1.

(f) The CETOP-2 Algorithm for CPC Thermal Margin Calculations (Ref. 9)

CETOP-2 algorithm for ANO-2 Cycle 2 replaces the CPCTH algorithm used in Cycle 1 CPC software.

(g) CPC/CEAC System Phase II Test Report (Ref. 10)

The implementation of the CETOP-2, as well as other CPC/CEAC software modifications into the CPC system has been examined through testing of the integrated system.

The ANO-2 Cycle 2 core contains 177 fuel assemblies of the 16 x 16 geometry. These assemblies consist of presently operating Batch A, B, and C assemblies, along with fresh Batch D assemblies. The Cycle 1 termination burnup has been assumed to be approximately 12.5 GWD/t. After the reload, the BOC-2 exposure will be 7.9 GWd/t, and the EOC-2 exposure is predicted to be 19.0 GWd/t. The maximum EOC-2 exposure of any individual assembly will be 25.2 GWd/t.

The objective of the review is to confirm that the thermohydraulic design of the reload core has been accomplished using acceptable methods, and provides acceptable margin of safety from conditions which would lead to fuel damage during normal operation and anticipated operational transients.

2.2 Design Methodology Review

The ANO-2 Cycle 2 design methodology involves several changes over Cycle 1. The COSMO/W-3 thermal margin design code has been replaced by the TORC/CE-1 (Ref. 11) and CETOP-D/CE-1 (Ref. 3) codes. The treatment of plant system parameter uncertainties has been changed from the deterministic approach to a statistical combination of uncertainties (SCU) and incorporates the system parameter uncertainties directly in the DNBR limit (Ref. 7). The rod bow compensation for the proposed DNBR limit is also calculated using a method (Ref. 6) which is under review but not yet approved. In addition, the DNBR calculational method in the CPC software has been changed from CPCTH to CETOP-2 (Ref. 8). Therefore, the Cycle 2 thermal design is a major change from the original Cycle 1 design methodology.

2.3 CETOP-D Thermal Margin Design Analysis Code:

The CETOP-D computer code is used as a core thermal margin design analysis tool for the ANO-2 Cycle 2 reload. CETOP-D is an open-lattice thermal hydraulic code which solves the same conservation equations and uses the same constitutive equations as in the TORC code (Ref. 11). TORC, derived from COBRA-III C (Ref. 12), is a multi-stage thermal margin code. The determination of hot channel coolant conditions and minimum DNBR are performed through three sequential steps, i.e., core-wide, hot fuel assembly and hot subchannel DNBR calculations. A simplified TORC design modeling method was developed and described in CENPD-206P (Ref. 13). In simplified TORC, two sequential calculations are made for thermal margin analysis, i.e., a core-wide analysis determining lateral boundary conditions for hot assembly; and a hot assembly analysis determining hot subchannel coolant conditions and minimum DNBR. The CETOP-D design code simplifies one step further by simply using a one step calculation for the core thermal margin analysis. The modeling uses a four-channel core representation with a lumped-channel technique. It uses "transport coefficients" serving as weighting factors for the treatment of diversion crossflow and turbulent mixing between adjoining channels. Furthermore, a "prediction-correction" method is used to solve the conservation equations, replacing the iterative method used in the TORC code. The magnitude of the changes, therefore, requires that the CETOP-D code be totally reviewed for acceptability as a thermal design tool.

The staff has reviewed the CETOP-D topical report. The review includes the conservation equations, constitutive equations, transport coefficients, method of solutions, and the benchmark result compared to TORC. Highlights of the review are described as follows:

(a) The derivation of the governing conservation equations has been examined. The staff has discovered two errors in the axial momentum equation (equation 1.7) and a vector direction error in the axial momentum control volume representation (Figure 1.3 of the CETOP-D topical). However, these errors have been identified as just typographical errors. The final axial momentum equation has been verified to be correct.

(b) Several errors in the constitutive equations have been discovered. These errors include the Dittus-Boelter forced convection correlation, the Jens-Lottes nucleate boiling correlation, the Martinelli-Nelson void fraction correlation, and two-phase friction factor multiplier. The errors have been identified as typographical errors and are programmed correctly and, therefore, non-consequential.

(c) Two errors have been found in the Tong-F factor used for the critical heat flux correction for non-uniform axial heat flux distribution. The errors are (i) using the node inlet location, rather than a varying axial location in the integrand and (ii) using the critical heat flux instead of the local heat flux in the denominator of the F-factor. These errors also exist in the TORC topical. The staff has required the licensee to provide the derivation, using the correct F-factor, leading to the final numerical formula to be used in the FORTRAN programming. The result shows that the correct F-factor has been used in the program. The errors are, therefore, non-consequential.

(d) The staff has reviewed the finite difference method used in solving the conservation equations. The finite difference equations are the same as used in the COBRA-III C code except that transport coefficients are used in the energy equation and axial and transverse momentum equations. Typographical errors exist in the momentum equations but are non-consequential.

Since a prediction-correction method is used in solving the conservation equations, the staff has raised the concern about numerical instability where an error might be propagated and amplified without bound throughout the subsequent calculation. However, the complexity of the diversion crossflow solution, involving a simultaneous solution of the mass and axial momentum as well as transverse momentum equations, makes an analytical stability analysis a formidable task. The licensee has run thousands of cases covering the entire range of operating conditions comparing CETOP-D to TORC without encountering any instability. Therefore, the numerical stability should be of no concern.

(e) The accuracy of the prediction-correction solution method has been examined. For each axial segment, the solution calculates a "predicted" diversion crossflow based on the assumption of zero lateral pressure difference at the node exit. The predicted crossflow is then used to calculate the lateral pressure difference with the adjacent channel which, in turn, is used to calculate the "corrected" crossflows. The error in the predicted crossflow depends on the relative importance of the lateral pressure difference in the crossflow equation and the local conditions at each node. The licensee has cited a fictitious example (response to Question 492.55, Ref. 14) to demonstrate the relatively small overall error of the prediction-correction method. Assuming that the exclusion of the node exit lateral pressure difference term accounts for a 30 percent error in the predicted crossflow, the error will result in nine percent error in the corrected crossflow. Since the diversion crossflow is small (less than five percent) compared to the axial flow, the error in the mass flow will be even smaller and the prediction-correction method is, therefore, acceptable.

(f) The sensitivity study performed by the licensee has shown minimal effects of the pressure and velocity transport coefficients on DNBR. These coefficients are calculated from TORC subchannel results and the values used for the CETOP-D code for ANO-2 Cycle 2 are provided in the response to the NRC question 492.3 (Ref. 14).

The enthalpy transport coefficient plays an important role in the accuracy of the lumped subchannel model. The staff has reviewed the lumped subchannel modeling and the assumptions concerning the mass flux, diversion crossflow and turbulent exchange in the lumped subchannel. Based on the assumptions, the enthalpy transport coefficient is derived from the energy equation for each axial segment. Except for a typographical error in the equation 4.2 of the topical, the staff has concluded that the CETOP-D equation for calculation of enthalpy transport coefficient is correct and, therefore, acceptable.

(g) The lumped channel model of four-channel core representation is a simplification of the detailed model used in TORC. One channel represents the core-wide average coolant conditions; the second channel represents the hottest assembly. The other two channels are the hot channels and the lumped channel representing peripheral subchannels. These two channels are then lumped within the hot assembly channel. The hot assembly and hot channel selections are the same as that described in the TORC topical. However, the CETOP-D model is only approximate in describing the true physical phenomena. The actual locations of the hot assembly and hot channel are deemed unimportant. An inlet flow factor obtained from reactor model experiment data is used for the hot assembly in the same manner as the simplified TORC modeling (Ref. 13). For ANO-2 Cycle 2, the hot assembly inlet flow factor with the value described in response to the NRC question 492.14 has been used to ensure that the CETOP-D result always calculates a lower DNBR than the detailed TORC over all operating conditions.

(h) In response to the NRC questions 492.7 and 492.68 (Ref. 14), the licensee has provided comparison between the CETOP-D and TORC results over the whole spectrum of operating conditions for ANO-2, Calvert Cliff Units 1 and 2, and San Onofre 2 and 3. In all cases, the CETOP-D calculates minimum DNBR lower than the TORC calculations. Since the TORC code has been approved for use in CE thermal margin design, the staff concludes, based on the conservatism of CETOP-D relative to TORC, that the CETOP-D code is acceptable for ANO-2 thermal margin calculations. Based on our review, the acceptance of CETOP-D carries the condition that the conservative hot assembly inlet flow factor described in response to question 492.14 (Ref. 14), or a smaller value be used for ANO-2 Cycle 2.

2.4 CE-1 Correlation (Generic Limit)

For ANO-2 Cycle 2, the CHF calculation has been changed from the W-3 correlation to the CE-1 correlation (Refs. 4 and 5). The CE-1 correlation has previously been approved for interim plant specific applications with a minimum DNBR limit of 1.19. However, our generic evaluation has now been completed and concludes that the 1.19 limit is consistent with the submitted data base. Our findings will be discussed in detail in the Safety Evaluation Report of CENPD-207-P (Ref. 5).

2.5 Fuel Rod Bow

The licensee has proposed a rod bow compensation of 2 percent on DNBR using the method described in Supplement 3P to CENPD-225-P (Ref. 6), which is not an approved document. Accordingly, it is the staff position that the rod bow compensation currently specified in the modified (Ref. 15) Technical Specification 4.2.4.4 shall be applicable for initial Cycle 2 operation. The modified technical specification requires that the row bow compensation for each batch be determined from the batch's maximum burnup assembly and applied to the batch's maximum radial power peaking assembly. This is acceptable to the staff. We estimate that the peak bundle average burnup for the most limiting batch will be approximately 5.0 GWD/t by the end of November 1981, when the rod bow compensation review is expected to be complete. The rod bow compensation required for that burnup is 2.0 percent of the DNBR limit value, the same as proposed by the licensee. If the rod bow compensation methodology is not approved by then, the licensee is required to re-evaluate the rod bow compensation every 31 days in accordance with the modified Technical Specification.

The staff agrees to issue the proposed Technical Specification change further reducing the rod bow compensation in Technical Specification 4.2.4.4 if the CENPD-225-P Supplement 3P is approved.

2.6 Statistical Combination of Uncertainties (SCU)

Data required for a detailed thermal-hydraulic analysis are divided into system parameters, which describe the physical system and are not monitored during reactor operation, and state parameters, which describe the operational state of the reactor and are monitored during operation. There is a degree of uncertainty in the value used for each of the parameters. This uncertainty has been handled in the past by assuming that each variable is at its extreme most adverse limit of its uncertainty range. The assumption that all factors affecting DNB are simultaneously at their most adverse values is very unlikely and leads to conservative restrictions in reactor operation. The licensee has proposed in CEN-139(A)-P (Ref. 7) a new methodology to statistically combine uncertainties of the system parameters and incorporate their effects on DNBR to derive a new equivalent DNBR limit. This new DNBR limit, while using the nominal values of system parameters in design analysis, will ensure with at least 95 percent probability and 95 percent confidence level that DNB will not occur.

The licensee's approach for SCU is to adopt a single set of "most adverse state parameters" and generate a MDNBR response surface of the system parameters, which is, in turn, applied in Monte Carlo methods to combine numerically the system parameter probability distribution functions with the CHF correlation uncertainty. Our review of the SCU methodology includes the selection of the most adverse state parameters, the elimination of some system parameters from the response surface, the uncertainties of system parameters in the response surface and the statistical method used in calculating the final equivalent MDNBR limit.

(a) Most Adverse State Parameters

Generation of the actual response surface simultaneously relating MDNBR to both system and state variables would require an inordinate number of detailed TORC analyses. The licensee's solution to this problem is to select one single set of state parameters for use in developing the system variable response surface. The problem then becomes one of selecting a single set of state parameters, termed the most adverse state parameter set, that leads to conservatism in the system parameter response surface; i.e., the resultant MDNBR uncertainty is maximized. Calculations are performed with the detailed TORC code to determine the sensitivity of the system parameters at several sets of operating conditions (state parameters). By tabulating the results of the sensitivity studies and through an examination of tables and exercise of engineering judgment, the "most adverse is listed in Section 3.1.5 of the CEN-139(A)-P report.

Our review has found that the values of these parameters, such as system pressure, inlet coolant temperature and primary flow rate, are very likely at their most adverse values. However, the conclusion is not valid for the axial shape index (ASI).

It is stated that the MDNBR is a smoothly varying function of the state parameters. This is not the case for the ASI. The ASI enters the calculation of MDNBR by the selection of a value of ASI from a finite collection of axial shapes and corresponding ASI's. Because the correspondence between ASI and axial shape is a multi-valued relationship, MDNBR cannot be a continuous function of ASI. Thus, a relatively small perturbation in ASI could lead to a large change in MDNBR. The data presented in CEN-139(A)-P indicate the possibility of an ASI that is considerably more adverse than the ASI selected as most adverse. In response (Ref. 17) to our question (Ref. 18) the licensee provided additional evaluations of the sensitivity of MDNBR near the most adverse ASI. With this additional information, the ASI selected as most adverse can be accepted as leading to conservative estimates of the sensitivity of MDNBR to system parameter variation. We, therefore, conclude that the licensee has achieved the goal of finding the most adverse set of state parameters.

(b) System Parameter Uncertainties

The CEN-139(A)-P report lists each of the system variables and then either provides the rationale for eliminating the variable from the statistical combination or provides the appropriate uncertainty value. Our review of these variables follows:

(i) Radial Power Distribution

Conservatism in the thermal margin modeling is listed as a reason that uncertainty in the radial power distribution need not be considered. A subsequent response to questions (Ref. 17) outlined the proprietary calculational technique currently being used to maintain the conservatism. The technique was reviewed and found to be satisfactory. The elimination of the radial power distribution uncertainty is justified.

(ii) Inlet Flow Distribution

The sensitivity studies in CEN-124(B)-P (Ref. 19) has shown that MDNBR in the limiting hot assembly is unaffected by changes in the inlet flow of assemblies which are diagonally adjacent to the hot assembly. Therefore, only the inlet flow to the hot assembly and its contiguous neighbors are included in the analysis. We find this approach acceptable.

We have also reviewed the flow test data report provided in response to NRC Question 492.63 and concluded that the means and standard deviations of inlet flow factors listed on Table 5.1 of CEN-139(A)-P are correct.

(iii) Exit Pressure Distribution

The sensitivity study provided in CEN-124(B)-P (Ref. 19) has shown the insensitivity of MDNBR with respect to the variation in exit pressure distribution. Therefore, we conclude the elimination of the exit pressure distribution uncertainty from the MDNBR response surface acceptable.

(iv) Enthalpy Rise Factor

Enthalpy rise factor is used to account for the effect on hot channel enthalpy rise of the fuel manufacturing deviation from nominal values of fuel dimension, density, enrichment, etc. The enthalpy rise factor is determined in accordance with an approved quality assurance procedure (Ref. 20). This involves a 100 percent recording of the relevant data which are then collected into a histogram. The mean and standard deviation are determined with 95 percent confidence. We find this procedure and the uncertainty listed on Table 5.1 (Ref. 7) acceptable.

(v) Heat Flux Factors

Manufacturing tolerance limits and fuel specifications are used which conservatively define the probability distribution function of the heat flux factor. We find the mean and the standard deviation of heat flux factor used in the analysis are conservative and, therefore, acceptable.

(vi) Clad O.D.

Clad diameter mean values and standard deviations are given based on as-built data. The minimum systematic clad O.D. and its standard deviation are used in the development of the heat flux factor since this gives the most adverse effect on DNB. The maximum clad O.D. and its standard deviation are used in wetted perimeter calculations which penalizes the MDNBR. This accounting of the clad O.D. uncertainty introduces conservatism in the analysis and is acceptable.

(vii) Systematic Pitch Reduction

As-built data are used to determine the mean and standard deviation of the gap width. The minimum mean and its standard deviation are chosen for combination with maximum clad O.D. to give the minimum pitch. The use of the minimum gap width is a conservative approach and is acceptable.

(viii) Fuel Rod Bow

The methodology for calculating rod bow compensation is discussed in Section 2.3. The rod bow compensation is applied directly as a multiplier to the MDNBR limit and the approach is acceptable.

(ix) CHF Correlation

The DNBR limit associated with the CE-1 correlation as discussed in Section 2.2 is imposed to account for the uncertainty of the correlation itself only. Other uncertainties associated with plant system parameters and measurements of operating state parameters are accounted for, separately, through accompanying uncertainty factors.

In our review of the correlation prediction uncertainty, we also applied a cross-validation technique, where the test data are divided into two equal portions. The parameters of the correlation are estimated separately on each half. The estimated correlation from one half is then used to predict the data from the other half. Based on results of the cross validation technique, we conclude that the standard deviation of the measured to predicted CHF ratio should be increased by 5 percent. This increase in correlation uncertainty should be included in the derivation of the DNBR limit.

(x) Code Uncertainty

Uncertainty exists in all subchannel codes. Our evaluation result of the CE-1 DNBR limit using the COBRA IV code differs slightly from the licensee's analysis using the TORC code. This is, to a great extent, a result of the inherent calculational uncertainties in the two codes. The licensee contends that since the same TORC code is used for both CHF test data analysis and CHF calculations in the reactor, the code uncertainty is implicitly included in the minimum DNBR limit that is used for reactor application. However, we find the argument not valid since the CHF test section, being a small number of representative pins, differs from the reactor fuel assemblies in the large reactor core. Even though the heated shrouds are used in the test assembly, the two-phase frictional pressure drop and diversion cross flow phenomena, etc., result in uncertainties in thermal hydraulic conditions predicted in the test assembly and reactor core. Information to quantify these uncertainties is not easily obtained and has not been provided. Therefore, consistent with past practice, we have imposed a 4 percent uncertainty for the subchannel codes and 1 percent uncertainty for transient codes which predict conservatively against data. These code uncertainties are imposed only when SCU is used for design analysis. The code uncertainties are included in our evaluation of the applications of SCU to account for the effect of the uncertainties on DNBR limit.

(c) Response Surface of System Parameters

The use of a response surface to represent a complicated, multi-variate function is an established statistical method. A response surface relating MDNBR to system parameters is created. Conservatism is achieved by selecting the "most adverse set" of state parameters that maximizes the sensitivity of MDNBR to system parameter variations. The response surface includes linear, cross-product, and quadratic terms in the system parameters. Data to estimate the coefficients of the response surface are generated in an orthogonal central composite design using the TORC code with the CE-1 CHF correlation. The resulting MDNBR response surface is described in Table 4-2 of CEN-139(A)-P.

The licensee has calculated the coefficient of determination associated with the response surface to be 0.9988 and the standard error of 0.002826. We conclude that the response surface prediction of MDNBR is acceptable.

(d) Derivation of Equivalent MDNBR Limit

The probability distribution function (pdf) of MDNBR is estimated using the response surface in a Monte Carlo simulation. The simulation also accounts for uncertainty in the CHF correlation. The estimated MDNBR pdf is approximately normal, and a 95/95 probability/confidence limit is assigned using normal theory.

The SIGMA code is used in a simulation to estimate the distribution of MDNBR. SIGMA is described in the statistical evaluation of Part 1 of CENPD-124(B)-P (Ref. 21). The results of the simulation were compared to results obtained using an analytical propagation of variance. The two methods are in close agreement. Therefore, we conclude the use of Monte Carlo simulation and SIGMA code is acceptable.

In our review of the statistical methodology used in deriving the final equivalent MDNBR limit, we discovered that an incorrect number of degrees of freedom is used in calculating the error associated with the response surface at 95 percent confidence level. However, since the error associated with the response surface is very small, the error results in minimal effect on DNBR limit.

The derivation of the SCU - equivalent MDNBR limit is generally acceptable except for the omissions of the CE-1 correlation cross-validation uncertainty and code uncertainty. As described in Section 2.4.b.ix, the standard deviation of the measured/predicted CHF ratio should be increased by 5 percent resulting from cross-validation of the test data. This increased uncertainty results in an increase of MDNBR by 0.005. Secondly, a 5 percent code uncertainty should be included in the response surface. Assuming this uncertainty equal to two standard deviations, and combining the standard deviation with the standard deviation of the response surface by root sum square method, the MDNBR limit will increase by a factor of 1.008, i.e., an increase of 0.01 in MDNBR limit. With the generic MDNBR limit of 1.19 for the CE-1 correlation, the SCU-equivalent MDNBR becomes 1.231. With 2 percent rod bow compensation, as proposed by the licensee, the final MDNBR limit should be 1.26 (1.256) compared to the proposed 1.24. This increase from the proposed value of 1.24 to 1.256 has been accounted for by increasing the value of BERR 1 as discussed in Section 2.10.

(e) SCU Review Conclusion

The SCU methodology presented in CEN-139(A)-P has been found acceptable with the following exceptions:

1. code uncertainties of 5 percent should be included in SCU analysis;
2. pending approval of CENPD-225-P, the current Technical Specification should be used for rod bow compensation calculation;
3. the new equivalent DNBR limit is 1.26 (1.256) including SCU for system parameters and an interim rod bow compensation of 2 percent on DNBR;
4. any changes in codes or correlations used in the analysis will require a re-evaluation of the SCU.

2.7 CPC/CEAC Software Modifications

The Core Protection Calculators (CPC) and Control Element Assembly Calculators (CEAC) of the ANO-2 Cycle 2 are basically identical hardware with a modified version of the software from that of Cycle 1. The software modifications are described in CEN-143(A)-P (Ref. 8).

Since the Cycle 1 CPC/CEAC was reviewed extensively and approved, the staff's review efforts of the Cycle 2 CPC/CEAC have been concentrated on the software modifications. The following is a list of software modifications and the staff evaluations:

(a) Addressable constants have been added for CEA shadowing factor adjustments, planar radial peaking factor adjustments, and boundary point power correlation coefficients. These addressable constants have been added to adjust CPC power distributions based on startup measurement tests. Cycle 1 operating experience has shown that this improves the accuracy in calculations of core power and power distribution. We find these changes acceptable.

(b) Some fixed numbers in the power distribution calculation (POWER) have been changed to data base constants. The original fixed numbers were based on Cycle 1 design conditions. Making them data base constants provides flexibility to change plant-specific or cycle dependent values without changing the CPC Functional Specifications. We have reviewed the new data base constants involved and their previous values as well as their data base values for the Cycle 2, where available. We find these modifications acceptable.

(c) Planar radial peaking factors are now adjusted by a correction factor based on the CE proprietary value of a reactor parameter. We have reviewed this modification and agree that it provides a more accurate calculation of power distribution for various core conditions. We, therefore, find it acceptable.

(d) The boundary point power correlation has been simplified and the constant coefficients in the correlation have been made addressable. Cycle 1 startup experience has shown that the previous dependencies in the algorithm are not necessary. We have reviewed the new boundary point power correlation calculation including the additional data base constants which have been added and find the modifications acceptable.

(e) A pre-selected axial power distribution is now used during low power operation. This provides a conservatively independent axial power distribution at low power levels. We find the use of this pre-selected shape acceptable.

(f) The slope of the coolant temperature shadowing factor has been made an addressable constant. The slope was previously a non-addressable data base value. However, since the shadowing factor is verified during startup testing, the slope can be adjusted based on test measurements. We agree that this should result in more accurate CPC calculations of neutron flux and power distribution and find this modification to be acceptable.

(g) The pump-dependent uncertainty on local power density (LPD) is revised to be applied to the DNBR and LPD update program, UPDATE, instead of the trip sequence program. This change results in including the uncertainty in the LPD margin to the CPC operator's module. The staff has reviewed the software algorithm and found the modification acceptable.

(h) The DNBR and LPD pre-trip set points have been made addressable constants. This change adds flexibility in setting pre-trip alarm set points and allows for adjustment of the set point without a revision to the data base. The change does not require a change to the DNBR pre-trip logic. However, the LPD pre-trip set point has to be converted from the unit of kW/ft to percent of core average power density. The staff has reviewed the software algorithm and found it acceptable.

(i) Two new curve fits are used for the core coolant enthalpy/temperature ratio and the normalized specific volume as functions of pressure and temperature. The enthalpy/temperature ratio curve fit is good for the temperature range from 455^{o}F to 5^{o}F below saturation temperature. If the hot leg temperature is within 5^{o}F of the saturation temperature, the CPC will initiate the hot leg saturation trip. The staff has done audit calculations of coolant enthalpy using the enthalpy/temperature ratio curve fit. The resulting enthalpy compares within 0.1 percent of the enthalpy value obtained from ASME steam tables. Therefore, the new curve fit for the enthalpy/temperature curve is acceptable. The staff has also done audit calculations of the normalized specific volume.

It was found that using the curve fit described in CEN-143(A)-P results in the specific volume of water rather than the normalized value as described. However, during the staff's audit of CE internal files, we found that the actual curve fit in the CPC software has been normalized with the specific volume of water at the condition of 2250 psia and 553^{o}F. Therefore, the staff concludes that the error in CEN-143 is non-consequential and the new curve fit for the normalized specific volume is acceptable.

(j) The Cycle 2 CPC now uses CETOP2/CE-1 for minimum DNBR calculations compared to CPCTH/W-3 used in ANO-2 Cycle 1. This modification creates the most impact on core operating thermal margin. CETOP2 is the fourth generation of the steady state thermal margin analysis code, TORC. The first-generation TORC requires three-stage core modeling to determine the hot channel minimum DNBR. The second-generation simplified-TORC (CENPD-206-P) requires two-stage core modeling. The third-generation CETOP-D uses one-stage lumped channel modeling and transport coefficients for the treatment of crossflow and turbulent mixing between adjoining channels. CETOP-D also uses a prediction-correction method instead of the iteration method used in TORC to solve the conservation equations. The difference in modeling and solution technique results in a very large difference between the TORC and CETOP-D codes even though the same conservation equations and constitutive equations are used. CETOP2 used in the CPC is the offspring of CETOP-D. It uses constant transport coefficients rather than calculating them as is done in the CETOP-D. Any error resulting from this simplification is accommodated by an algorithm uncertainty factor applied to the CETOP2 core power to ensure that the CETOP2 calculated DNBR is conservative with respect to the CETOP-D with 95/95 probability/ confidence level.

Since CETOP-D is a new code, the evaluation of CETOP2 is dependent upon the acceptability of the CETOP-D code. The staff has reviewed the CETOP-D code, which has shown conservative results compared to the TORC code over a wide range of operating conditions, and found it acceptable as described in Section 2. .

2.8 CETOP2 Algorithm Review

The staff has reviewed the CETOP2 functional specification and has performed an audit (Ref. 22) of the functional tests of the integrated system to assure that CETOP2 with the algorithm uncertainty factor is programmed properly and predicts minimum DNBR conservatively.

The CETOP2 functional description is provided in the Appendix B of CEN-143. The following is a summary of the results of our review:

(a) Errors have been discovered in the Martinelli-Nelson void fraction correlation and the two-phase friction factor multiplier. However, the errors have been identified as just typographical errors and are programmed properly. Therefore, these errors are nonconsequential.

(b) The single-phase friction factor calculation using the Blasius correlation, where the friction factor is a function of Reynolds number, has been studied. Since ANO-2 fuel cladding surface roughness ranges from 14 to 21 micro inches RMS, the calculated friction factor agrees with the Moody friction factor within three percent in the normal operating condition range where the Reynolds number is around 5×10^5. Therefore, the friction factor calculation using the Blasius correlation is acceptable.

(c) In order to reduce the CPC execution time, many friction factor and two-phase multiplier calculation algorithms have been converted from exponential functions to polynomial fits. The staff has examined the accuracy of these conversions and found them acceptable.

(d) CETOP2 uses lumped channel modeling where the core is divided into four modeling channels, i.e., core region channel, hot assembly channel, buffer channel, and hot channel. The hot channel is a pseudo-hot channel which models a corner guide tube subchannel. The staff has raised questions (Ref. 14) as to how the hot channel is selected; whether the selected hot channel always predicts the lowest DNBR; whether minimum DNBR always occurs in a guide tube channel; and whether it is legitimate to use a guide tube channel to represent other channels where the minimum DNBR might occur. To answer these questions, the licensee has addressed the fact that the modeling is independent of the actual location of the hot assembly and hot channel within the core. An inlet flow split factor for the hot assembly is used to yield conservative DNBR predictions relative to the detailed TORC code. The inlet flow split factor is obtained from the reactor model flow test experiment. During operating transients, the flow split may change significantly. However, the most adverse of the flow splits has been used in the CETOP2. The inlet flow split factor is described in Table B-2 of CEN-143, plant-specific constants for ANO-2. As for the legitimacy of using a guide tube subchannel, the licensee has stated that the present fuel management schemes result in power distributions which produce the largest pin peaks near guide tube water holes throughout the core life. The cold wall correction factor in the CE-1 CHF correlation is also used to reduce the predicted DNBR in the quide tube channels. As a result, the minimum DNBR will always be predicted to occur in a corner guide tube channel. The staff concludes that the pseudo-hot channel modeling is acceptable provided that the fuel management scheme ensures that the calculated minimum DNBR always occurs in a guide tube subchannel.

(e) In the lumped channel modeling, transport coefficients are used to account for the fact that the coolant properties associated with turbulent mixing and diversion crossflow between adjacent channels are not the lumped channel average values. Constant values of the transport coefficients are used in the CETOP-2. In response to the staff question 492.3, the licensee has provided a sensitivity study of the DNBR with respect to the transport coefficients. The DNBR has been shown to be insensitive to the pressure transport coefficient. However, the enthalpy transport coefficient has been shown to have a significant effect on the hot channel enthalpy. In CETOP-D, an enthalpy transport coefficient is calculated for each axial level. The value chosen for the CETOP-2 is such that the CETOP-2 results match the CETOP-D results for a typical axial power distribution and nominal operating conditions. Any errors resulting from this simplification are covered by an algorithm penalty factor on core power.

(f) The algorithm uncertainty factor represents compensation applied to the core power in CPCs to ensure that the DNBR results from CETOP2 are conservative relative to CETOP-D. In response to our question 492.15 (Ref. 14), the licensee has run 6400 cases of comparison between CETOP2 and CETOP-D; and a compensation factor has been derived so that application of the compensation factor to the core power results in a 95/95 probability/confidence level that CETOP2 is more conservative than CETOP-D. (These cases are run using the value of BERR1 equal to 1.0). However, the licensee has subsequently submitted a supplement (Ref. 10) indicating that errors exist in the original analysis resulting from the use of hot pin peaking factor as hot channel peaking factor in the CETOP-2 input. The reanalysis of the 6400 cases results in a decrease of the algorithm uncertainty factor by 2 percent. Since the use of the algorithm uncertainty power compensation factor or a larger value as a core power multiplier is to ensure a conservative DNBR prediction from CETOP2, our acceptance of the CETOP2 code as applied to ANO-2 depends upon the acceptability of the algorithm uncertainty factor. Until we complete our review of the new uncertainty factor, we require that the original uncertainty factor be used.

Since the new algorithm uncertainty factor is built into the CPC software, the adjustment should be made through the addressable constant BERR1. Also, the original algorithm uncertainty factor is listed in Table B-2 of the CEN-143(A)-P as a plant specific constant.

The licensee is required to submit a revision to the CEN-143(A)-P for consistency. After the data base constant has been corrected and our review of the revised uncertainty factor is complete, we will issue a supplement to this SER providing the conclusion of our review for this open issue.

2.9 CPC Phase II Test Review:

The implementation of the CETOP-2, as well as other modifications, into the CPC system has been examined through the utilization of Phase II testing. The primary objective of the Phase II testing is to verify that the CPC and CEAC software modifications have been properly integrated with the CPC and CEAC software and the system hardware. The testing also provides confirmation that the static and dynamic operation of the integrated system as modified is consistent with that predicted by design analysis. The objectives are achieved by comparing the response of the integrated system to the response predicted by the CPC FORTRAN simulation code. The licensee has submitted the CPC Phase II test report (Ref. 10). In the Dynamic Software Verification Test (DSVT), 40 transient cases, ranging from four-pump loss of flow to CEA withdrawal and primary system depressurization transients, have been run on both the FORTRAN Simulation and the CPC software in the single channel test facility.

The resulting initial DNBR, initial LPD and the trip times from the single channel test fall well within the acceptance criteria for each case established from the FORTRAN simulation runs. For the six cases where the trip times fail to stay within the acceptance criteria (the single channel trip time is 0.1 second outside the FORTRAN acceptance criteria), the cause has been identified to be the difference between the single channel and FORTRAN simulation in the interpolation of time dependent parameters input tables provided to the UPDATE program.

Because UPDATE is run every 0.1 second, this interpolation difference can result in a 0.1 second difference in trip time for the cases with faster transients. The plant parameters before and after trip are compared between the FORTRAN simulation and the single channel to verify that the discrepancy is indeed due to slight difference in the input to the UPDATE program. We, therefore, conclude that this result does not indicate the existence of software error.

The staff has performed an audit (Ref. 22) of the Phase II test and confirmed the accuracy of the report. During the staff's audit of the DSVT test, a comparison between Cycles 1 and 2 was made for the loss of flow test Case No. 1 using the same initial conditions. With about the same neutron flux power, the static component of the thermal power differs about 2 percent between the two cycles. Further investigation of the reason for the difference has identified the cause as due to the new curve fit used in the Cycle 2 for the coolant enthalpy/temperature ratio calculation. Since the new curve fit has been found accurate compared with the ASME steam tables, the issue is closed.

Since the CPC software and the FORTRAN Simulation are developed independently by two different divisions within the CE organization, the agreement of the Phase II testing has shown the adequacy of the implementation of the functional specification. Therefore, the staff concludes that the software modification implementation is acceptable.

2.10 Thermal Margin Limits:

The DNBR limit of 1.19 associated with the CE-1 CHF correlation is imposed to account for only the uncertainty of the correlation itself. Other uncertainties associated with plant system parameters and measurements of operating state parameters are accounted for as follows:

(a) A SCU as described in Section 2.4 is used for the treatment of uncertainties of the system parameters, such as enthalpy rise factor and systematic cladding diameter, etc. This results in the incorporation of the system parameter uncertainties directly into the DNBR limit.

(b) An addressable constant, BERR1, is used to account for the operating state parameter measurement uncertainties and other uncertainties not included in the SCU. The Cycle 2 calculations of BERR1 were similar to Cycle 1, except that the simulation was expanded to include the stochastic simulation of uncertainty on the state variables, i.e., pressure, temperature and mass flow.

The licensee has determined a BERR1 value of 1.055 (Ref. 15) to account for the overall uncertainties of power distribution synthesis, radial peaking factor, DNBR algorithm modeling and constants, CPC processing, and static and dynamic allowances. The CETOP2 algorithm uncertainty is included in the DNBR algorithm modeling uncertainty and is, therefore, treated statistically as opposed to the direct multiplication described in CEN-143(A)-P. We have reviewed the stochastic methodology used for the treatment of the state parameter uncertainties (as described in response to question 492.65 Ref. 14) and found it acceptable. We have also reviewed the detailed justification of the sources and magnitude of plant measurement uncertainties (response to Question 492.64, Ref. 14) and found it acceptable. We have further reviewed the BERR1 uncertainty components and calculation (Refs. 15 and 10) and conclude that the BERR1 value of 1.055 correctly accounts for the uncertainties discussed above in this paragraph. However, this BERR1 value is based on a late revision to the CETOP2 algorithm factor (see Section 3.2, Item F) which is still under review. We require that the original uncertainty factor be used.

The licensee also proposed a minimum DNBR limit of 1.24 resulting from the SCU including a rod bow compensation of 2 percent on DNBR. However, our review has concluded that the equivalent DNBR limit of 1.26 should be used for Cycle 2 including 2 percent row bow compensation (see Section 2.4, Item e).

Since the DNBR limit of 1.24 has been built into the CPC software as a trip set point, a temporary remedy is to incorporate the difference in DNBR into the addressable constant, BERR1. The licensee has performed a sensitivity study (response to Question 492.66, Ref. 14) of the derivative of percent Power Operating Limit (POL) with respect to percent DNBR for various ASI. The results show that the derivatives vary as a function of ASI. Based on the most conservative derivative value, we conclude that the DNBR limit of 1.24 for CPC trip set point is acceptable provided that the addressable constant, BERR1 is set to not less than 1.065.

In addition, any further adjustment in rod bow compensation (see Section 2.3) and other additional compensation should be provided by adjusting the BERR1 value accoding to the following formula:

$$BERR1 = 1.065 \times \{1 + (RB + C - 2) \times D/100\} \cdot B$$

where RB is the rod bow compensation (percent of DNBR) corresponding to the maximum fuel burnup of the limiting fuel batch; C (percent of DNBR) is any additional compensation to the DNBR limit; B is the uncertainty compensation directly affecting BERR1; D is the absolute value of the most negative derivative from the response to 492.66.

Since the new CETOP2 algorithm uncertainty factor is not yet approved, we require the increase of the BERR1 value by 2 percent. Therefore, pending the approval of the CETOP2 algorithm uncertainty, the BERR1 value should be no less than 1.086.

2.11 Comparison of Thermal Hydraulic Design Conditions:

Comparison of the thermal hydraulic design conditions for ANO-2 Cycle 1 and Cycle 2 is provided in Table 4.1 of the SER for Amendment No. 24. Significant differences in the design parameters between the two cycles are in calculational factors. However, because of the design methodology changes, the thermal margin gained for Cycle 2 is very large. In response to the NRC question 492.67 (Ref. 14), the licensee has provided a breakdown of the estimated margin gained due to various methodology changes, such as TORC/CE-1 vs COSMO/W-3; CETOP2 vs CPCTH; SCU vs deterministic treatment of system parameter uncertainties; and stochastic treatment vs. combination of deterministic and statistical treatment of state parameters. Overall, these methodology changes result in a total overpower margin gain on the order of 15 percent or more.

In response to NRC questions 492.22, 492.27, and 492.62 (Ref. 14), the licensee provided comparisons of the minimum DNBR's calculated by TORC/CE-1, CETOP-D/CE-1, CETOP-2/CE-1 and COSMO/W-3 (for Cycle-2 loss of coolant flow and full power CEA withdrawal transients). These comparisons do show the conservatism of the CETOP2 and CETOP-D codes relative to the TORC code. However, for the COSMO/W-3 calculation, which is used in Cycle 1 thermal margin analysis, the estimated minimum DNBRs are 1.115 and 1.121, respectively, for Cycle 2 loss of flow and CEA withdrawal transients. These values are well below the allowable W-3 DNBR limit of 1.3 and infer that Cycle 2 and later power distributions were not fully considered in the FSAR analyses of these events. In response to the NRC question 492.76 (Ref. 14), the licensee has indicated that primarily the first-core parameters were used in the FSAR analyses. As for the Cycle 2 result using COSMO/W-3, the licensee explains that the comparisons were made at the point of minimum DNBR during the LOF and CEA withdrawal transients. These transients were terminated by CPC trip using CETOP/CE-1 thermal margin methodology. Had the analyses been done with the COSMO/W-3 methodology, CPCs would have been designed to trip at W-3 DNBR below 1.3 and more restrictive operating limits may have been required. We agree with the explanation.

2.12 Evaluation Summary:

We have reviewed the ANO-2 Cycle 2 thermal design methodology, CPC/CEAC software and safety analyses as summarized below:

(a) The CETOP-D code is acceptable for use in ANO-2 safety analyses as a substitute for TORC. However, in using CETOP-D, the hot assembly inlet flow factor with the value described in the response to NRC question 492.14 or a smaller value must be used to ensure conservative DNBR predictions relative to TORC.

(b) The CE-1 DNBR limit for ANO-2 has been evaluated. The CHF test data support a limit of 1.19 for the CE-1 correlation.

(c) Our review of SCU is complete. We have found the SCU methodology acceptable. However, a correlation cross-validation uncertainty and a 5 percent code uncertainty must be included resulting in an increase of DNBR limit. The approved DNBR limit is 1.26 including a 2 percent rod bow compensation. The increase in DNBR limit from the proposed value of 1.24 may be accommodated in the addressable constant, BERR1 (Item e).

(d) The proposed rod bow compensation calculation is under review and, pending its approval, we require that the modified Technical Specification be used for rod bow compensation evaluation. We estimate that the peak bundle average burnup for the limiting batch will be 5 GWD/t by the end of November 1981. The rod bow compensation for that burnup is 2 percent of the DNBR limit value, the same as proposed. After November 1981, if the CENPD-225 methodology is not approved, the rod bow compensation should be recalculated every 31 days in accordance with the Technical Specifications.

(e) The CPC software modifications and implementation have been found acceptable for ANO-2 Cycle 2 except that the CETOP2 algorithm uncertainty factor is still under review. Pending the completion of our review, we require that the uncertainty factor (E_1) listed in the plant specific constant for ANO-2 (Table B-2, Ref. 8) be used. With a built-in DNBR trip set point of 1.24, we require that the addressable constant BERR1 be at least 1.086. This value should be adjusted upward using the formula described in Section 3.4, if the rod bow and other additional compensation is changed.

(f) The thermal power margin gained througn the Cycle 2 methodology changes has been estimated to be on the order of 15 percent or more. It appears that the original design analyses of ANO-2 were not sufficient to assure that adequate operating thermal margin could be maintained for the core lifetime in accordance with Regulatory Guide 1.70, the standard format. In response to a staff question, the licensee has indicated that the licensing basis for ANO-2 Cycle 2 is applicable to later cycles and that at present no additional methodology changes for the purpose of thermal margin gain are anticipated.

3.0 Technical Specification Changes

RPS Instrument Trip Setpoints, Table 2.2-1

The value of BERR1 in Note (6) is changed to reflect the completion of the staff's review of the Cycle 2 CPCS changes as discussed in detail in Section 2.10 of this report. Upon resolution of the appropriate value of the CETOP-2 algorithm uncertainty factor the required value of BERR1 will be adjusted downward to as low as 1.065. This adjustment will be included in further changes to the TS.

4.0 ENVIRONMENTAL CONSIDERATION

We have determined that this amendment does not authorized a change in effluent types or total amounts nor an increase in the licensed power level of 2815 MWt and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR 51.5(d)(4) that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

5.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: July 21, 1981

Principle Contributors:
L. Phillips, CPB
G. Hsii, CPB
S. Gupta, CPB
H. Balukjian, CPB
G. Hesson, PNL
R. Martin, ORB#3

6.0 References

1. Letter from D. C. Trimble (AP&L) to Director, NRR, with ANO-2, Cycle 2 Reload Report submittal, February 20, 1981.

2. Letter from W. Cavanaugh III (AP&L) to Director, NRR, with ANO-2, Cycle 2 Reload Report submittal, March 5, 1981.

3. Letter from D. C. Trimble (AP&L) to Director, NRR, dated July 15, 1981, "CETOP-D Code Structure and Modeling Methods, Response to First Round Questions on the Statistical Combination of Uncertainties Program," (CEN-139(A)-P, March 1981).

4. CENPD-162-P-A, "Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spacer Grids Part 1, Uniform Axial Power Distribution," September 1976.

5. CENPD-207-P, June 1976, "C-E critical heat flux: Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spacer Grids Part 2, Nonuniform Axial Power Distribution."

6. Supplement 3-P (Proprietary) to CENPD-225P, "Fuel and Poison Rod Bowing," June 1979.

7. Letter from D. C. Trimble (AP&LCo) to Director, NRR dated December 1, 1980 submitting CEN-139 "Statistical Combination of Uncertainties" (SCU).

8. Letter from D. C. Trimble (AP&LCo) to Director, NRR dated January 9, 1981 submitting CEN-143 "CPC/CEAC Software Modifications for Arkansas Nuclear One - Unit No. 2, December 1980".

9. Letter from D. C. Trimble (AP&LCo) to Director, NRR, dated January 9, 1981 submitting Appendix B, CEN-143(A)-P, "CPC/CEAC Software Modifications for Arkansas Nuclear One Unit 2, December 1980".

10. Letter from D. C. Trimble (AP&LCo) to Director, NRR, dated July 20, 1981 submitting CEN-162-A-P, "CPC/CEAC System Phase II Software Verification Test Report, Document 50-361, May 1981" and a supplement to the response to 492.15 and a description of the calculational methodology for BERR1 with SCU.

11. CENPD-161-P, "TORC Code, A Computer Code for Determining the Thermal Margin of A Reactor Core," July 1975.

12. D. S. Rowe, "COBRA III C: A Digital Computer Program for Steady State and Transient Thermal-Hydraulic Analysis of Rod Bundle Nuclear Fuel Elements," BNWL-1695, March 1973.

13. CENPD-206-P, "TORC Code, Verification and Simplification Methods," January 1977.

14. Letter from D. C. Trimble to Director NRR dated June 19, 1981 submitting CEN-157(A)-P; Amendment 1-P, "Response to Questions (492.1 to 492.22, 492.25 to 492.29, and 492.48 to 492.77, June 1981) and CEN-167(A)-P "Reactor Vessel Open Core Flow Model Test Report, " June, 1981.

15. Letter from W. Cavanaugh, III (AP&L) to R. Clark (NRC), dated June 10, 1981 submitting TS changes on BERR 1 using SCU and non-SCU methods.

16. Letter from D. C. Trimble (AP&L) to H. Denton (NRR) with responses to NRC Questions (Question 492.31), April 21, 1981.

17. Letter from D. C. Trimble (AP&LCo) to Director, NRR dated May 6, 1981 submitting responses to staff's letter of April 23, 1981 on SCU questions 492.38 - 492.47.

18. Letter from G. M. Hesson (PNL) to H. Balukjian (NRR), dated March 27, 1981.

19. Combustion Engineering, "Statistical Combination of Uncertainties Methodology, Part 2: Combination of System Parameter Uncertainties in Thermal Margin Analyses for Calvert Cliffs Units 1 and 2," CEN-124(B)-P, January 1980.

20. Combustion Engineering, "Quality Assurance Program," CENPD-210-A, Rev. 3, November 1977.

21. Combustion Engineering, "Statistical Combination of Uncertainties Methodology, Part 1: C-E Calculated Local Power Density and Thermal Margin/Low Pressure LSSS For Calvert Cliffs Units 1 and 2," CEN-124(B)-P, December 1979.

22. Memorandum from Y. Hsii to L. Phillips, "Trip Report - ANO-2 Cycle 2 Core Protection Calculator Phase II Test Audit," June 5, 1981.

23. CENPD-107, "CESEC Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," April 1974.

24. CENPD-199-P, "CE Set Point Methodology," April 1976.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-368

ARKANSAS POWER AND LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 26 to Facility Operating License No. NPF-6 issued to the Arkansas Power and Light Company, which revised the license and the Technical Specifications for operation of Arkansas Nuclear One, Unit No. 2, (the facility) at steady state reactor core power levels not in excess of 2815 megawatts thermal, in accordance with the provisions of the license and the Technical Specifications. The facility is located at the licensee's site in Pope County, Arkansas. The amendment is effective as of its date of issuance.

The ANO-2 plant has been authorized by Amendment No. 24 to operate during Cycle 2 subject to a license condition which limited the plant to seventy percent of the licensed power level of 2815 MWt pending completion of all details of the staff's review of the thermal hydraulic aspects of the Cycle 2 Core Protection Calculator System software changes. This review has been satisfactorily completed and accordingly this amendment relieves the license condition restricting power to seventy percent and authorizes Cycle 2 operation at the licensed power level of 2815 MWt.

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

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

For further details with respect to this action, see (1) the applications for amendment dated February 20 and March 5, 1981, as supplemented by references identified in the related Safety Evaluation, (2) Amendment No. 26 to Facility Operating License No. NPF-6, and (3) the Commission's related Safety Evaluation. These items are available for public inspection at the Commission's Public Document Room at 1717 H Street, N.W., Washington, D.C. 20555 and at the Arkansas Tech University, Russellville, Arkansas 72801. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Divison of Licensing.

Dated at Bethesda, Maryland, this 21st day of July, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing