May 7, 1985

Docket No. 50-368

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Mr. John M. Griffin
Senior Vice President
Energy Supply
Arkansas Power & Light Company
P. O. Box 551
Little Rock, Arkansas 72203

Dear Mr. Griffin:

The Commission has issued the enclosed Amendment No. 66 to Facility Operating License No. NPF-6 for Arkansas Nuclear One, Unit No. 2. The amendment consists of changes to the Technical Specifications (TS) in response to your applica-application dated January 28, 1985.

The amendment revises the Departure from Nucleate Boiling Ratio (DNBR) limit used by the Core Protection Calculators (CPC), modifies Table 2.2-2, "CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS" and removes the rod bow penalty factor surveillance requirement. As discussed with certain members of your staff, the allowable ranges of the azimuthal tilt allowance cannot be approved at this time and will be addressed in a future correspondence.

A copy of the Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's next monthly Federal Register Notice.

Sincerely,

/S/

Robert Lee, Project Manager
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 66 to NPF-6
2. Safety Evaluation

cc w/enclosures:
See next page

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Signed 7 May '85 pet

Mr. John M. Griffin
Arkansas Power & Light Company

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

ARKANSAS POWER & LIGHT COMPANY

DOCKET NO. 50-368

ARKANSAS NUCLEAR ONE, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 66
License No. NPF-6

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

   A. The application for amendment by Arkansas Power & Light Company (the licensee) dated January 25, 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.

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2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-6 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 66 , 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

for James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 7, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 66

FACILTIY 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 areas of change. The corresponding overlead pages are also provided to maintain document completeness.

Remove Pages	Insert Pages
2-1	2-1
2-6	2-6
2-7	2-7
2-9	2-9
B 2-1	B 2-1
B 2-2	B 2-2
B 2-3	B 2-3
B 2-6	B 2-6
3/4 2-8	3/4 2-8
B 3/4 2-3	B 3/4 2-3
B 3/4 2-4	B 3/4 2-4 (repositioned)

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

2.1.1 REACTOR CORE

DNBR

2.1.1.1 The DNBR of the reactor core shall be maintained $\geq$ 1.25.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the DNBR of the reactor core has decreased to less than 1.25, be in HOT STANDBY within 1 hour.

PEAK LINEAR HEAT RATE

2.1.1.2 The peak linear heat rate (adjusted for fuel rod dynamics) of the fuel shall be maintained $\leq$ 21.0 kw/ft.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the peak linear heat rate (adjusted for fuel rod dynamics) of the fuel has exceeded 21.0 kw/ft, be in HOT STANDBY within 1 hour.

SAFETY LIMITS

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2750 psia.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

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	≥ 23% (4)	≥ 22.111 (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	$\geq$ 1.25 (5)	$\geq$ 1.25 (5)
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.

TABLE 2.2-2

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE I ADDRESSABLE CONSTANTS

POINT ID NUMBER	PROGRAM LABEL	DESCRIPTION	ALLOWABLE VALUE
60	FC1	Core coolant mass flow rate calibration constant	≤1.15
61	FC2	Core coolant mass flow rate calibration constant	0.0
62	CEANOP	CEAC/RSPT inoperable flag	0, 1, 2 or 3
63	TR	Azimuthal tilt allowance	≥1.02
64	TPC	Thermal power calibration constant	≥0.80
65	KCAL	Neutron flux power calibration constant	≥0.60
66	DNBRPT	DNBR pretrip setpoint	Unrestricted
67	LPDPT	Local power density pretrip setpoint	Unrestricted
98	TCREF	Reference cold leg temperature	525°F TCREF ≤555°F
104	PCALIB	Secondary calorimetric power	≤102.0%

TABLE 2.2-2 (Continued)

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

II. TYPE II ADDRESSABLE CONSTANTS

POINT ID NUMBER	PROGRAM LABEL	DESCRIPTION
68	BERR0	Thermal power uncertainty bias
69	BERR1	Power uncertainty factor used in DNBR calculation
70	BERR2	Power uncertainty bias used in DNBR calculation
71	BERR3	Power uncertainty factor used in local power density calculation
72	BERR4	Power uncertainty bias used in local power density calculation
73	EOL	End of life flag
74	ARM1	Multiplier for planar radial peaking factor
75	ARM2	Multiplier for planar radial peaking factor
76	ARM3	Multiplier for planar radial peaking factor
77	ARM4	Multiplier for planar radial peaking factor
78	ARM5	Multiplier for planar radial peaking factor
79	ARM6	Multiplier for planar radial peaking factor
80	ARM7	Multiplier for planar radial peaking factor
81	SC11	Shape annealing correction factor
82	SC12	Shape annealing correction factor
83	SC13	Shape annealing correction factor
84	SC21	Shape annealing correction factor
85	SC22	Shape annealing correction factor
86	SC23	Shape annealing correction factor
87	SC31	Shape annealing correction factor
88	SC32	Shape annealing correction factor

TABLE 2.2-2 (Continued)

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

II. TYPE II ADDRESSABLE CONSTANTS (Continued)

POINT ID NUMBER	PROGRAM LABEL	DESCRIPTION
89	SC33	Shape annealing correction factor
90	PFMLTD	DNBR penalty factor correction multiplier
91	PFMLTL	LPD penalty factor correction multiplier
92	ASM2	Multiplier for CEA shadowing factor
93	ASM3	Multiplier for CEA shadowing factor
94	ASM4	Multiplier for CEA shadowing factor
95	ASM5	Multiplier for CEA shadowing factor
96	ASM6	Multiplier for CEA shadowing factor
97	ASM7	Multiplier for CEA shadowing factor
99	BPPCC1	Boundary point power correlation coefficient
100	BPPCC2	Boundary point power correlation coefficient
101	BPPCC3	Boundary point power correlation coefficient
102	BPPCC4	Boundary point power correlation coefficient
103	RPCLIM	Reactor power cutback time limit

2.1 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

2.1.1 REACTOR CORE

The restrictions of these safety limits prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by (1) restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature, and (2) maintaining the dynamically adjusted peak linear heat rate of the fuel at or less than 21 kw/ft which will not cause fuel centerline melting in any fuel rod.

First, by operating within the nucleate boiling regime of heat transfer, the heat transfer coefficient is large enough so that the maximum clad surface temperature is only slightly greater than the coolant saturation 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 higher cladding temperatures and the possibility of cladding failure.

Correlations 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 predicted DNB heat flux at a particular core location to the actual heat flux at that location, is indicative of the margin to DNB. The minimum value of DNBR during normal operational occurrences is limited to 1.25 for the CE-1 correlation and is established as a Safety Limit.

Second, operation with a peak linear heat rate below that which would cause fuel centerline melting maintains fuel rod and cladding integrity. Above this peak linear heat rate level (i.e., with some melting in the center), fuel rod integrity would be maintained only if the design and operating conditions are appropriate throughout the life of the fuel rods. Volume changes which accompany the solid to liquid phase change are significant and require accomodation. Another consideration involves the redistribution of the fuel which depends on the extent of the melting and the physical state of the fuel rod at the time of melting. Because of the above factors, the steady state value of the peak linear heat rate which would not cause fuel centerline melting is established as a Safety Limit. To account for fuel rod dynamics (lags), the directly indicated linear heat rate is dynamically adjusted.

Limiting safety system settings for the Low DNBR, High Local Power Density, High Logarithmic Power Level, Low Pressurizer Pressure and High

Linear Power Level trips, and limiting conditions for operation on DNBR and kw/ft margin are specified such that there is a high degree of confidence that the specified acceptable fuel design limits are not exceeded during normal operation and design basis anticipated operational occurrences.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The Reactor Coolant System components are designed to Section III of the ASME Code for Nuclear Power Plant Components. (The reactor vessel, steam generators and pressurizer are designed to the 1968 Edition, Summer 1970 Addenda; piping to the 1971 Edition, original issue; and the valves to the 1968 Edition, Winter 1970 Addenda. Section III of this Code permits a maximum transient pressure of 110% (2750 psia) of design pressure. The Safety Limit of 2750 psia is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3125 psia to demonstrate integrity prior to initial operation.

2.2.1 REACTOR TRIP SETPOINTS

The Reactor Trip Setpoints specified in Table 2.2-1 are the values at which the Reactor Trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

The DNBR - Low and Local Power Density - High are digitally generated trip setpoints based on Limiting Safety System Settings of 1.25 and 20.3 kw/ft, respectively. Since these trips are digitally generated by the Core Protection Calculators, the trip values are not subject to drifts common to trips generated by analog type equipment. The Allowable Values for these trips are therefore the same as the Trip Setpoints.

To maintain the margins of safety assumed in the safety analyses, the calculations of the trip variables for the DNBR - Low and Local Power Density - High trips include the measurement, calculational and processor uncertainties and dynamic allowances as defined in CEN-147(S)-P, "Functional Design Specification for a Core Protection Calculator," January 1981; CEN-148(S)-P, "Functional Design Specification for a Control Element Assembly Calculator," January 1981; CEN-296(A)-P, "ANO-2 CPC and CEAC Data Base Listing,"; and CEN-288(A), "CPC Methodology Changes for Arkansas Nuclear One Unit 2 Cycle 5," October 1984, which references CEN-281(S)-P, "CPC/CEAC Software Modifications for San Onofre Nuclear Generating Station Units 2 and 3," June 1984, and Enclosure 1-P to LD-82-039, "CPC/CEAC Software Modification for System 80," March 1982.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Linear Power Level-High

The Linear Power Level-High trip provides reactor core protection against rapid reactivity excursions which might occur as the result of an ejected CEA. This trip initiates a reactor trip at a linear power level of $\leq$ 110.712% of RATED THERMAL POWER.

Logarithmic Power Level-High

The Logarithmic Power Level - High trip is provided to protect the integrity of fuel cladding and the Reactor Coolant System pressure boundary in the event of an unplanned criticality from a shutdown condition. A reactor trip is initiated by the Logarithmic Power Level - High trip at a THERMAL POWER level of $\leq$ 0.819% of RATED THERMAL POWER unless this trip is manually bypassed by the operator. The operator may manually bypass this trip when the THERMAL POWER level is above 10^{-4}% of RATED THERMAL POWER; this bypass is automatically removed when the THERMAL POWER level decreases to 10^{-4}% of RATED THERMAL POWER.

Pressurizer Pressure-High

The Pressurizer Pressure-High trip, in conjunction with the pressurizer safety valves and main steam safety valves, provides reactor coolant system protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is at ≤ 2370.887 psia which is below the nominal lift setting (2500 psia) of the pressurizer safety valves and its operation avoids the undesirable operation of the pressurizer safety valves.

Pressurizer Pressure-Low

The Pressurizer Pressure-Low trip is provided to trip the reactor and to assist the Engineered Safety Features System in the event of a Loss of Coolant Accident. During normal operation, this trip's setpoint is set at ≥1712.757 psia. This trip's setpoint may be manually decreased, to a minimum value of 100 psia, as pressurizer pressure is reduced during plant shutdowns, provided the margin between the pressurizer pressure and this trip's setpoint is maintained at ≤ 200 psi; this setpoint increases automatically as pressurizer pressure increases until the trip setpoint is reached.

Containment Pressure-High

The Containment Pressure-High trip provides assurance that a reactor trip is initiated concurrently with a safety injection. The setpoint for this trip is identical to the safety injection setpoint.

Steam Generator Pressure-Low

The Steam Generator Pressure-Low trip provides protection against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The setpoint is sufficiently below the full load operating point of approximately 900 psia so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This trip's setpoint may be manually decreased as steam generator pressure is reduced during plant shutdowns, provided the margin between the steam generator pressure and this trip's setpoint is maintained at ≤ 200 psi; this setpoint increases automatically as steam generator pressure increases until the trip setpoint is reached.

Steam Generator Level-Low

The Steam Generator Level-Low trip provides protection against a loss of feedwater flow incident and assures that the design pressure of the Reactor Coolant System will not be exceeded due to loss of the steam generator heat sink. This specified setpoint provides allowance that there will be sufficient water inventory in the steam generator at the time of the trip to provide sufficient margin before emergency feedwater is required.

Local Power Density-High

The Local Power Density-High trip is provided to prevent the linear heat rate (kw/ft) in the limiting fuel rod in the core from exceeding the fuel design limit in the event of any anticipated operational occurrence. The local power density is calculated in the reactor protective system utilizing the following information:

a. Nuclear flux power and axial power distribution from the excore flux monitoring system;

b. Radial peaking factors from the position measurement for the CEAs;

c. ΔT power from reactor coolant temperatures and coolant flow measurements.

The local power density (LPD), the trip variable, calculated by the CPC incorporates uncertainties and dynamic compensation routines. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core peak LPD is sufficiently less than the fuel design limit such that the increase in actual core peak LPD after the trip will not result in a violation of the peak LPD Safety Limit. CPC uncertainties related to peak LPD are the same types used for DNBR calculation. Dynamic compensation for peak LPD is provided for the effects of core fuel centerline temperature delays (relative to changes in power density), sensor time delays, and protection system equipment time delays.

DNBR-Low

The DNBR - Low trip is provided to prevent the DNBR in the limiting coolant channel in the core from exceeding the fuel design limit in the event of anticipated operational occurrences. The DNBR - Low trip incorporates a low pressurizer pressure floor of 1750 psia. At this pressure a DNBR - Low trip will automatically occur. The DNBR is calculated in the CPC utilizing the following information:

a. Nuclear flux power and axial power distribution from the excore neutron flux monitoring system;

b. Reactor Coolant System pressure from pressurizer pressure measurement;

c. Differential temperature (ΔT) power from reactor coolant temperature and coolant flow measurements;

d. Radial peaking factors from the position measurement for the CEAs;

e. Reactor coolant mass flow rate from reactor coolant pump speed;

f. Core inlet temperature from reactor coolant cold leg temperature measurements.

The DNBR, the trip variable, calculated by the CPC incorporates various uncertainties and dynamic compensation routines to assure a trip is initiated prior to violation of fuel design limits. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core DNBR is sufficiently greater than 1.25 such that the decrease in actual core DNBR after the trip will not result in a violation of the DNBR Safety Limit. CPC uncertainties related to DNBR cover CPC input measurement uncertainties, algorithm modelling uncertainties, and computer equipment processing uncertainties. Dynamic compensation is provided in the CPC calculations for the effects of coolant transport delays, core heat flux delays (relative to changes in core power), sensor time delays, and protection system equipment time delays.

The DNBR algorithm used in the CPC is valid only within the limits indicated below and operation outside of these limits will result in a CPC initiated trip.

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 3.2-4.

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

DELETED

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 1.25 could occur, as calculated by COLSS, is less than or equal 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.

A DNBR penalty factor has been included in the COLSS and CPC DNBR calculations to accommodate the effects of rod bow. The amount of rod bow in each assembly is dependent upon the average burnup experienced by that assembly. Fuel assemblies that incur higher average burnup will experience a greater magnitude of rod bow. Conversely, lower burnup assemblies will experience less rod bow. In design calculations, the penalty for each batch required to compensate for rod bow is determined from a batch's maximum average assembly burnup applied to the batch's maximum integrated planar-radial power peak. A single net penalty for COLSS and CPC is then 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.

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 66 TO FACILITY OPERATING LICENSE NO. NPF-6

ARKANSAS POWER & LIGHT COMPANY

ARKANSAS NUCLEAR ONE, UNIT 2

DOCKET NO. 50-368

1.0 INTRODUCTION

By letters dated November 9, 1984 (Ref. 1) and January 28, 1985 (Ref. 2), respectively, Arkansas Power and Light Company (the licensee) proposed changes to the ANO-2 Core Protection Calculator (CPC) methodology and Technical Specifications. In addition to the proposed Technical Specification changes, the licensee also submitted CEN-289(A)-P (Ref. 3) for a revised fuel rod bow penalty calculation. In connection with the CPC methodology changes, the licensee by letter dated March 29, 1985 (Ref. 4) also submitted CPC/CEAC Data Base Listing, Phase I and Phase II Software Verification Test reports. Our evaluation of these submittals follows.

2.1 CPC Methodology Changes

The changes to the ANO-2 CPC methodology as described in CEN-288(A), "CPC Methodology Changes for Arkansas One Unit 2 Cycle 5" (Ref. 5) consist of ten (10) items. They are evaluated below:

(1) Implementation of LPD PF with Failed CEACs:

An error was discovered previously in the ANO-2 CPC software where a penalty factor (PF) associated with failure of both Control Element Assembly Calculators (CEACs) was not applied to the Local Power Density (LPD) calculation required by the CPC functional specification. Although the licensee had found no safety implication of this software error, they had committed by a letter dated October 18, 1983 (Ref. 6) to correct the CPC software error prior to initial reactor criticality for Cycle 5. The proposed modification is to fulfill this commitment to bring the CPC software consistent with the functional specification and is acceptable.

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(2) Conversion of Hot Pin to Hot Channel Peaking Factors:

A discrepancy exists in the ANO-2 CPC software where the hot pin radial peaking factors were used as the hot channel peaking factors. This is inconsistent with the off-line CETOP-D thermal-hydrualic calculation. The proposed modification is to apply the correct factors of hot channel to hot pin peaking factor ratios to the hot channel power distribution calculation. This modification makes the hot channel linear heat rate distribution consistent with the hot pin input values. The modification has been reviewed and approved by NRC for CESSAR-80 plants (Ref. 7) such as Palo Verde Unit 1 and is therefore acceptable.

(3) Transient Adjustment of Non-Uniform Heating Correction Factors:

This modification is related to transient adjustment in the UPDATE program of the non-uniform axial power shape F-factor for DNBR calculation. A new adjustment formula is used for updating the STATIC calculated F-factor based on the updated quality resulting from change in reactor conditions. The new formula incorporates a coefficient whose value depends on the direction of the change in quality margin, i.e., a constant coefficient is used for the updated quality below the static critical quality whereas another constant coefficient is used for the updated quality greater than the other critical quality. This change has been reviewed and approved previously for CESSAR-80. The licensee has also indicated that the same method is used in the determination of the constant coefficients in the adjustment equation. Therefore, the modification for ANO-2 is acceptable.

(4) PFMLTD and PFMLTL Range Limits

The Type II addressable constants PF_{MLTD} and PF_{MLTL} are multipliers to the CEAC penalty factors for DNBR and LPD calculations. When positive numbers are input for those multipliers, they are applied to the fractional parts (the part greater than 1.0) of the penalty factors. These multipliers permit adjustment of the slopes of the curves used to derive the penalty factors in the CEACS. The positive allowable range limits of PF_{MLTD} and PF_{MLTL} are proposed to be shifted to smaller values. The changes in the allowable ranges provides flexibility to set the values of PF_{MLTD} and PF_{MLTL}. This change has been approved previously for CESSAR-80 and is therefore acceptable.

(5) RC Pump Pressure Rise Calculation:

This modification is related to the rapid pump flow coastdown events, such as a locked rotor event, where the pump affinity law does not apply for pumps at near zero speed. The pump pressure rise calculation is modified to include an algorithm in the FLOW program to account for a very rapid occurrence of negative head across a RC pump. This additional algorithm, which applies to the region of the forward flow through the pump at or near zero RPM, makes use of the data provided by the RC pump vendor. A new constant is added to the curve fit coefficients for the pump characteristics for the forward flow-low pump speed region. This modification represents an improvement to the pump pressure rise calculation and is consistent with CESSAR-80 and San Onofre 2/3 CPC. Therefore this modification is acceptable.

(6) Reactor Power Cutback System:

An algorithm for Reactor Power Cutback (RPC) system is added to the ANO-2 CPC software. The RPC system is designed to eliminate power imbalance without a reactor trip for large turbine load rejection and loss of one main feedwater pump events. During the RPC mode, pre-determined smaller penalty factors for CEA deviation are used by the CPC. After the RPC mode, the CPC returns to the normal mode where larger CEA deviation PFs are used by CPC. Since ANO-2 does not have a RPC system, the effect of numbers are input for those multipliers, they are applied to the fractional parts (the part greater than 1.0) of the penalty factors. These multipliers permit adjustment of the slopes of the curves used to derive the penalty factors in the CEACS. The positive allowable range limits of PF_{MLTD} and PF_{MLTL} are proposed to be shifted to smaller values. The changes in the allowable ranges provides flexibility to set the values of PF_{MLTD} and PF_{MLTL}. This change has been approved previously for CESSAR-80 and is therefore acceptble.

this algorithm will be nullified through appropriate data base constants. The licensee also proposes to add an addressable constant RPCLIM defining the maximum RPC mode duration. As shown in the ANO-2 CPC/CEAC Data Base Listing, RPCLIM (which is designated as TCBSP in the functional design specification) is set to 0 and therefore the reactor power cutback system is nullified. Since the CE CPC Owners Group is planning to standardize the CPC software for all CPC plants, addition of the RPC algorithm, which has been approved for other CPC plants, to the ANO-2 CPC is acceptable. However, use of any value of RPCLIM other than zero will require NRC approval with the RPCLIM determined through appropriate safety analysis to ensure no violation of the specified acceptable fuel design limit of DNBR.

(7) Modification of Heat Flux Distribution Extrapolation:

The hot pin axial power distribution is extrapolated in the STATIC algorithm by an extrapolation from a 20-node representation to a 21-node representation. For certain CEA configurations, this extrapolation could result in negative values of heat flux in the top node. The heat flux extrapolation algorithm is modified by adding a flux value check to prevent a non-physical zero or negative nodal heat flux. This modification has been approved for San Onofre Units 2 and 3 (Ref. 8) and is therefore acceptable for ANO-2.

(8) Improvement to UPDATE Algorithm:

The CPC detailed DNBR calculation is performed in the STATIC algorithm routine. Since the STATIC calculation is performed only once every two seconds, the DNBR is updated every 0.1 second within the two seconds interval in the UPDATE routine using the STATIC-calculated DNBR and the change in the state parameters, such as hot channel mass flow, heat flux and quality at the minimum DNBR node. Presently, the CPC UPDATE algorithm applies a penalty to the updated DNBR at all times. This algorithm is modified so that the penalty factor is applied only when the updated-DNBR

differs significantly from the STATIC-DNBR. This modification thus eliminates an unnecessary penalty for steady state operation. The penalty factors applied to the updated-DNBR depend on the magnitude of the deviation between the updated and STATIC DNBRs. This modification has been approved for San Onofre Units 2 and 3 CPCs and the licensee also indicates (Ref. 4) that the methodology used for generating penalty factors is the same as that used in San Onofre. Therefore, this modification is acceptable.

(9) Power Measurement Uncertainty:

Power measurement uncertainties are conservatively added to the thermal and neutron powers in the CPC core power calculation. These uncertainties include the CPC neutron flux synthesis error, the secondary calorimetric power measurement error, the secondary calorimetric power to the CPC power calibration allowance, and the thermal power transient offset. These uncertainties are treated as bias terms and are added together to be applied to CPC through the addressable constants, BERR0 and BERR2, which are the overall uncertainty biases for thermal and neutron power, respectively. Since the secondary calorimetric power measurement has a higher measurement uncertainty at lower power conditions due to higher instrumentation error and noise level, the revised CPC software will incorporate an algorithm to calculate the power-dependent measurement uncertainty for the secondary calorimetric power. This approach has been approved for San Onofre Units 2 and 3. The licensee also indicated that the same method used in San Onofre is used in ANO-2 to determine the power dependent uncertainty values. Therefore, this modification is acceptable. Since the secondary calorimetric power measurement error is dependent on the power level at which the calorimetric measurement is performed, an addressable constant PCALIB is added to the CPC for the determination of uncertainty. PCALIB is defined as the calorimetric power at the time the latest calibration was performed.

(10) Temperature Shadowing Factor Algorithm Modification:

The temperature shadowing factor (TSF) is a moderator temperature-dependent multiplier applied to the neutron power calculation in the CPC to correct the excore detector response to the decalibration effects due to changes in inlet coolant density. The current CPC calculates TSF as a simple function of the change in moderator temperature slope which is determined in start up testing. The modification will be made in the calculation of TSF so that TSF is a function of inlet moderator temperature consistent with the off-line TSF data. The slope of the correction for temperature is chosen to bound all expected TSF data. Therefore, the TSF uncertainty is included directly in the TSF itself. There is no need to further incorporate the TSF uncertainty in the overall uncertainty factors used in the CPC DNBR and LHR calculations. In addition, since the TSF algorithm is changed, an addressable constant, TCREF, defined as the reference cold leg temperature, is added to and the TSF correction multiplier, CORR1, is removed from the CPC addressable constants. This modification has been approved for San Onofre Units 2 and 3 CPCs and the licensee has indicated that the same method is used in ANO-2 for the determination of the TSF function. Therefore this modification is acceptable.

2.2 CPC/CEAC Data Base Constants

By letter dated March 29, 1985 (Ref. 4), the licensee submitted CEN-296(A), "ANO-2 CPC and CEAC Data Base Listing" (Ref. 9) and Enclosure 1-P to A-CE-R-92, "Typical Data Base Constants for Arkansas Nuclear One Unit 2". The latter document provides those data base constants relevant to the proposed CPC software changes and the values are extracted from CEN-296(A). The licensee also indicated that the methodology used for generating these constants is the same as used for San Onofre Units 2 and 3 which has been approved previously. We have reviewed the values of these constants and find them acceptable.

2.3 Verification of CPC/CEAC Software Modification Implementation:

The implementation of the CPC/CEAC software modifications translates the system functional requirements into modules of machine executable coding and integrates these modules into a real time software system. The overall CPC/CEAC software implementation is verified through the Phase I and Phase II software verification tests. By letter dated March 29, 1985, the licensee submitted ANO-2 CPC/CEAC Phase I and Phase II test reports, CEN-298(A) (Ref. 10) and CEN-162(A) (Ref. 11). The Phase I test was performed at the CEAC Single Channel Unit on relatively small, single-entry/single-exit segments of modules. The objective was to verify the implementation of CPC/CEAC software. Sufficient test cases were chosen to exercise each functional branch in the application program and executive software system. Expected results for the application program test cases were generated by either the CPC FORTRAN Simulation Code or by hand calculations by the test engineer based on the system functional requirements. When test case input had been selected and expected results had been generated, a test tape was prepared to be read by the automated Phase I test program. Whenever the actual value differed from the expected value by more than 0.1 percent, an analysis of the error was performed to assure that the deviation was not caused by a coding error. There were several branches not exercised because of the fact that the assigned constant values made it impossible to branch on certain conditions. In these cases the module was verified by inspection to assure correct implementation. The executive software was tested through the debug program, CLUB, which was used to insert test execution and to examine results. The report indicated that the overall Phase I test was performed in accordance with the approved Phase I test procedures and that the test results show no coding error in the application program and executive software. Therefore, the implementation of CPC/CEAC software into machine executable modules has been verified correctly.

The objectives of the Phase II tests are to verify that the CPC/CEAC software modifications have been properly integrated with the CPC/CEAC software and system hardware, and that the static and dynamic operation of the integrated system is consistent with the predictions of design analyses. These objectives were achieved by comparing the response to that projected by the CPC/CEAC FORTRAN Simulation Code. The test was performed in the Single Channel CPC and the test cases were selected in accordance with the approved procedure described in CEN-39(A)-P, Revision 02 (Ref 12).

The Phase II testing consisted of Input Sweep Tests (ISTs), Dynamic Software Verification Tests (DSVTs) and Live Input Single Parameter Tests (LISPs). ISTs were utilized to determine the processing uncertainties inherent in the CPC/CEAC designs. Thousands of cases were run in the ISTs and the resulting uncertainties were factored into the acceptance criteria for the DSVT and LISP. The DSVT is a real time exercise of the CPC software to verify the dynamic response of the integrated CPC software with design analyses by determining whether the initial DNBR and LPD calculations and the trip time of each transient are within the acceptance criteria predicted by the FORTRAN Simulation Code. In contrast to the DSVT where the transient CPC input values are read from a storage device, the LISP test is a real-time exercise with transient input values generated from an external source and read through the CEAC/CPC input hardware. The LISP test is to verify that the dynamic response of the trip time of the integrated CPC/CEAC software/hardware system is consistent with the design analysis prediction during operational modes approximating plant conditions. These tests have shown that in all cases the CPC/CEAC calculated results are within the acceptance criteria. Therefore, we conclude that the CPC/CEAC software has been implemented correctly.

3.0 Rod Bow Penalty:

The licensee proposed in its letter of January 28, 1985 (Ref. 2) to revise the DNBR limit used by the CPC. In support of the this DNBR limit revision, the licensee submitted a topical report CEN-289(A) entitled "Revised Rod Bow Penalties for Arkansas Nuclear One Unit 2" (Ref. 3).

Fuel rod bowing results in closure of the channel gap between fuel rods and therefore affects thermal-hydraulic behavior. The effects of fuel rod bowing on the critical heat flux (CHF) and DNBR are accounted for through a rod bow penalty on DNBR. The method of calculation of rod bow penalty is described in CENPD-225 and its supplements (Ref. 13). This method applies a statistical convolution of the CHF data (both bowed and unbowed) and channel gap closure data to derive a rod bow penalty on DNBR. The method has been approved by NRC except for a scaling factor for channel gap closure calculation. The scaling factor is used for extrapolation of channel gap closure data from one fuel assembly to another fuel assembly of different geometry. A scaling factor of L^2/I was derived in CENPD-225, where L is the span length between two adjacent grids and I is the moment of inertia of fuel rod cladding. This scaling factor was based on the fact that the deflection at the mid-span of a simple column subject to a compressive load is proportional to L^2/I. However, since the mechanism governing rod bowing is complicated and not well established, the derivation of the scaling factor assuming a simple column is questionable at best. The staff therefore rejected the scaling factor of L^2/I proposed in CENPD-225 and imposed a scaling factor of L/I, which results in more severe rod bowing when extrapolating from the 14x14 to 16x16 fuel assemblies. CEN-289(A) provides ANO-2 irradiation fuel data to support the scaling factor of L^2/I. The data obtained as part of a CE/EPRI Fuel Performance Characterization program consist of five sets of channel width measurements from three fuel bundles at different fuel exposures. Figure 1 of CEN-289(A) provides a comparison between the measured channel closure data from the 16x16 fuel bundles and the calculated results from the channel closure equation using scaling factors of L/I and L^2/I, respectively, for extrapolating from the 14x14 to 16x16 bundles. The values of L's and I's for both the 14x14 and 16x16 fuel-assemblies and the scaling factors L/I and

L^2/I (where $L/I = (L/I)16x16/(L/I)14x14$ and $L^2/I = (L^2/I)16x16/(L^2/I)14x14$) are provided in Table 3 of CEN-289(A). Since L/I is larger than L^2/I use of the scaling factor L/I in the channel closure equation will calculate more severe channel closure than using L^2/I. Figure 1 of CEN-289(A) shows that the calculated channel closure results using both L/I and L^2/I are much higher than the measured channel closure. Therefore, the licensee concluded that use of L^2/I as a scaling factor is justified. However, the channel closure equation includes a batch-to-batch variation correction factor of greater than 1.0 which distorts the calculated channel gap closure to a more severe value. For the purpose of evaluating the ability of a scaling factor to fit the data, we removed the batch-to-batch correction factor from the equation. The staff calculations with removal of the batch-to-batch correction factor from the channel closure equation show that use of the scaling factor L^2/I provides a better prediction than L/I and still results in a more conservative (more severe) prediction of channel closure than the measured data. We therefore conclude that the scaling factor of L^2/I is acceptable for ANO-2 fuel. This scaling factor of L^2/I rather than L/I will be used in ANO-2 for the calculation of fuel rod bowing augmentation factor and poison rod bowing augmentation factor.

The rod bow penalty or DNBR is re-calculated using the approved method described in Supplement 3 to CENPD-225 with the exception of the scaling factor being L^2/I instead of L/I. The input to the calculation is also changed as follows:

1. The hot rod average heat flux is changed from $0.387x10^6$ to $0.374x10^6$ Btu/hr-ft^2.

2. The reactor pressure is reduced from 2475 to 2422 psia

The licensee indicates that these new values are expected to bound future cycles. Therefore use of these values in the rod bow penalty calculators is acceptable. Figure 4 provides a rod bow DNBR penalty as a function of fuel burnup. Specifically, at fuel exposure of 30,000 MWD/MTU, the rod bow penalty is 0.5 percent DNBR. This figure is the result of calculations using the L^2/I correction and the new values of hot rod average heat flux and reactor pressure. Since we now find these to be acceptable, this rod bow penalty is acceptable.

4.0 Technical Specification Changes:

The staff has reviewed the proposed modifications to the ANO-2 Technical Specifications as presented in the letter dated January 28, 1985. Our evaluations follows.

4.1 DNBR:

The DNBR limit in Technical Specifications 2.1.1.1, Safety Limits Reactor Core DNBR, Table 2.2-1 - Limiting Safety System Settings Reactor Trip Setpoints, Bases 2.1.1, Reactor Core, and Bases 2.2.1, Reactor Trip Setpoints, is to be changed from 1.24 to 1.25. The Surveillance Requirement 4.2.4.4 on rod bow penalty is to be deleted. As result of application of statistical combination of uncertainties described in CEN-139(A) (Ref.14), the DNBR limit of 1.26 was approved by the staff. The DNBR limit of 1.26 includes a fuel rod bow penalty of 2.0% on DNBR. Starting in Cycle 3, the HID-1 spacer grids

are used in ANO-2 fuel assemblies to replace the standard grids which were used in the previous fuel assemblies and the test assemblies in the development of the CE-1 critical heat flux correlation. The staff had imposed a penalty of 0.01 on DNBR to account for CHF effect due to the slight difference between the HID-1 and standard grids. This penalty of 0.01 should be added to the DNBR limit. The DNBR limit trip setpoint of 1.24 was acceptable because the difference between the approved DNBR limit and 1.24 is compensated by an increase in the value of the CPC addressable constant BERR1, which is a penalty factor applied to the core power in the CPC DNBR calculation. As described in Section 3 of this Safety Evaluation, the rod bow penalty is 0.5 percent at fuel exposure of 30,000 MWD/MTU. Using 0.5 percent rather than 2 percent rod bow penalty, the DNBR limit is 1.24. With the addition of 0.01 DNBR to account for HID-1 spacer grids, the final DNBR limit is 1.25. Therefore, the proposed Technical Specification change of DNBR limit to 1.25 is acceptable. Since the rod bow penalty and grid effect penalty are already incorporated in the DNBR limit, there is no need to increase the value of BERR1.

Surveillance Requirements 4.2.4.4 requires that the rod bow penalty on DNBR as a function of fuel exposure should be verified to be included in the COLSS and CPC DNBR calculations at least once per 31 days. As discussed in Section 3, the rod bow penalty has been determined to be 0.5 percent at a fuel exposure of 30,000 MWD/MTU. Because of the physical burndown effect, a fuel assembly with burnup exceeding 30,000 MWD/MTU will not be subject to a limiting DNBR condition, therefore 30,000 MWD/MTU is a cutoff point for rod bow penalty calculation. Since the rod bow penalty of 0.5 percent at 30,000 MWD/MTU has been incorporated in the minimum DNBR limit, the Surveillance Requirement 4.2.4.4 can be deleted.

4.2 CPC Addressable Constants:

Changes are made to the CPC addressable constants defined in Table 2.2-2, "Core Protection Calculator Addressable Constants", of the ANO-2 Technical Specifications. These changes are evaluated as follows.

(a) Azimuthal Tilt Allowance (TR):

The addressable constant TR is an azimuthal power tilt allowance used in CPC as a power multiplier to increase the hot pin radial peaking. TR is frequently changed during restarts following reactor trips with transient core xenon conditions. Technical Specification 3.2.3, Limiting Condition for Operation, requires that the azimuthal power tilt (Tq) be less than or equal to the azimuthal power tilt allowance (TR) used in CPC. Surveillance Requirement 4.2.3 specifies that above 20% rated thermal power, the Tq shall be determined to be within the limit (i.e., $\leq$ TR) by (a) continuously monitoring the tilt with COLSS when the COLSS is operable, (b) verifying at least once per 31 days that the COLSS azimuthal tilt alarm is actuated when Tq > TR; (c) using the incore detectors at least once per 31 days to independently confirm the validity of the COLSS calculated Tq; and (d) calculating the tilt at least once per 12 hours when the COLSS is inoperable. This LCO limit is to ensure that the design safety margins are maintained. If the LCO is not met, specific actions are required per Technical Specification 3.2.3. Therefore, LCO 3.2.3 does prohibit operation with a non-conservative value of TR used in CPC. Previously, COLSS used an "arithmetic average" technique to calculate a core average azimuthal tilt value. Using this method, single noise input is enhanced by accumulating the magnitude component without considering the directional effect. The calculation in COLSS has been modified to use a "planar vector average" technique

which performs a vector sum of the individual tilt estimates at each axial plane for the calculation of the average tilt value of each plane. This planar average technique reduces the noise effects by allowing possible cancellation of some of the random components of noise. Therefore, an appropriately low tilt value is calculated when there is no tilt in the core. This calculation has also been demonstrated to agree well with the arithmetic average method when there is a true tilt in the core. The planar vector average technique for the estimation of azimuthal power tilt is also used in the approved CECOR code. Therefore, this modification to COLSS is acceptable. The purpose of the Technical Specification change to lower the minimum allowable value of TR is to reflect the reduced COLSS tilt estimate in the situation where there is no appreciable azimuthal power tilt in the core. However, since the minimum value of azimuthal tilt is 1.0, use of 1.0 for the TR in the CPC would result in frequent occurrences of the azimuthal tilt exceeding the TR, and therefore violating Technical Specification 3/4.2.3. This would increase the burden of the plant operators for compliance with the Action requirements specified in the Technical Specification. Therefore, the proposed Technical Specification change to reduce the minimum allowable value of TR from 1.02 to 1.0 is not acceptable. We recommend that the licensee propose a minimum allowable TR greater than 1.0 which provide a trade-off in power penalty versus excessive operator actions.

(b) Power Calibration Constants TPC and KCAL:

TPC and KCAL are power calibration constants used in the CPC calculations of the thermal (Δ T) and neutron powers, respectively, to reflect calibration against the secondary calorimetric power. These calibration constants require frequent change to the CPC during power operation. Technical Specification 3.3.1, Limiting Condition for Operation, specifies that as a minimum, the reactor protective instrumentation channels (e.g., CPC, DNBR and LPD functions) and bypasses shall be operable. Since thermal and neutron powers are important parameters in the DNBR and LPD trip functions, Surveillance Requirement 4.3-1 specifies periodic calibration of the CPC calculated thermal and neutron flux powers. Specifically, Note #2 of Technical Specification Table 4.3-1 requires that above 15% of rated thermal power, (at least once per 24 hours) the linear power level signals and the CPC addressable constant multipliers (TPC and KCAL) should be adjusted to make the CPC delta T power and CPC nuclear power calculations agree with the calorimetric calculations if the absolute difference is greater than 2%.

Since the thermal and neutron powers have a direct impact on the CPC DNBR and LPD calculations and respective trip functions, this surveillance requirement provides adequate assurance that the CPC calculated reactor powers are either accurate or conservative. The addressable constants TPC and KCAL provide adequate means to ensure that reactor power meets the surveillance requirement. The licensee is concerned that there may be instances where the calibrations of thermal and neutron powers require the calibration constants to be entered into the CPC outside their allowable ranges specified in Table 2.2-2. Since the thermal and neutron powers are the primary safety-related parameters in the DNBR and LPD calculations, it is important for these parameters to be within the required calibration limits. Therefore, the proposed changes to lower the allowable values of TPC and KCAL from 0.9 and 0.85 to 0.8 and 0.65, respectively, are acceptable.

(c) Change from CORR1 to TCREF:

The Type II addressable constant, point ID No. 98, CORRI is replaced by TCREF, defined as the reference cold leg temperature. This change is necessary due to the change in the CPC TSF calculational algorithm. The change is acceptable as described in Section 2.1, Item 10, "Temperature Shadowing Factor Algorithm Modification", of this SER.

(d) Addition of addressable constant RPCLIM:

A Type II addressable constant RPCLIM (Note: RPCLM in the proposed Table 2.2-2, Reference 2, is a typographic error) defined as the reactor power cutback time limit is added as point ID #103. This change is necessary due to the addition of the reactor power cutback algorithm to the ANO-2 CPC, and is acceptable as described in Section 2.1, Item 6, "Reactor Power Cutback System", of this SER.

(e) Addition of Addressable Constant PCALIB:

The addition of PCALIB as a Type I addressable constant is related to a CPC algorithm modification where a power-dependent measurement uncertainty for the secondary calorimetric power is used in the core power calculation. PCALIB is defined as the calorimetric power at the time of the latest calibration. The addition of PCALIB as an addressable constant is acceptable as discussed in Section 2.1, Item 9, "Power Measurement Uncertainty", of this SER.

5.0 Evaluation Findings:

The staff has reviewed the changes to the ANO-2 CPC methodolgy, the CPC/CEAC data base document, the Phases I and II CPC verification tests, the revision to the ANO-2 rod bow penalty, and the proposed changes to ANO-2 Technical Specifications. Based on the evaluation given in the preceding sections, the staff has found all of these items acceptable with the exception of the proposed change to lower the allowable value of the Type I addressable constant TR, azimuthal tilt allowance, to 1.0. The staff recommends that the licensee propose a new minimum allowable value of TR greater than 1.0 which provides a trade-off between power penalty and excessive operator actions.

6.0 Environmental Consideration:

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves 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 published a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets 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 the amendment.

7.0 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 the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: May 7, 1985

Principal Contributor:
Y. Hsii

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2. Letter from T. G. Campbell (AP&L) to J. R. Miller (NRC), "Arkansas Nuclear One - Unit 2, Docket No. 50-368, License No. NPF-6, Proposed Technical Specifications Change Request ANO-2 CPC Update", January 28, 1985.

3. CEN-289(A)-P, "Revised Rod Bow Penalties for Arkansas Nuclear One Unit 2", Docket No. 50-368, December 1984.

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7. Enclosure 1-P to LD-82-039, " CPC/CEAC Software Modifications for System 80", Dockets STN-50-470F, March 1982.

8. CEN-281(S)-P, "CPC/CEAC Software Modifications for San Onofre Nuclear Generating Station Units No. 2 and 3", July 1984.

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10. CEN-298(A)-P, Rev. 00, "Arkansas Nuclear One Unit 2, Core Protection Calculator System Phase I Design Qualification Test Report", Docket No. 50-368, March 1985.

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13. CENPD-225-P, "Fuel & Poison Rod Bowing" Combustion Engineering, Inc. October 1976, Supplement 3-P, June 1979.

14. CEN-139(A)-P, "Statistical Combination of Uncertainties: Combination of System Parameter Uncertainties in Thermal Margin Analyses for Arkansas Nuclear One Unit 2", November 1980.