PDR

July 22, 1986

Docket No.: 50-368

Mr. T. Gene Campbell
Vice President
Nuclear Operations
Arkansas Power & Light Company
Post Office Box 551
Little Rock, Arkansas 72203

Dear Mr. Campbell:

Subject: Issuance of Amendment No. 79 to Facility Operating License NPF-6 - Arkansas Nuclear One, Unit No. 2

The Commission has issued the enclosed Amendment No. 79 to Facility Operating License No. NPF-6 for Arkansas Nuclear One, Unit No. 2. The amendment consists of changes to the Technical Specifications in response to your application dated February 27, 1986.

The amendment revises the Technical Specifications pertaining to the Core Protection Calculators (CPC) as a part of the CPC Improvement Program.

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

Sincerely,

/s/

Robert S. Lee, Project Manager
PWR Project Directorate No. 7
Division of PWR Licensing-B

Enclosures:
1. Amendment No. 79 to NPF-6
2. Safety Evaluation

cc: See next page

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No legal objection

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7/8/86	7/9/86	7/11/86	7/21/86

Mr. T. Gene Campbell
Arkansas Power & Light Company

Arkansas Nuclear One
Unit No. 2

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Pope County Courthouse
Russellville, Arkansas 72801

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. 79
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 February 27, 1986, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's 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 regulations of the Commission;

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 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. 79 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert S. Lee

Robert S. Lee, Project Manager
PWR Project Directorate No. 7
Division of PWR Licensing-B

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 22, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 79

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

Remove Pages	Insert Pages
2-6	2-6
B 2-2	B 2-2
B 2-3	B 2-3
B 2-6	B 2-6
B 2-7	B 2-7
3/4 2-1	3/4 2-1
3/4 2-2	3/4 2-2
3/4 2-3	3/4 2-3
3/4 2-7	3/4 2-7
3/4 2-7a	3/4 2-7a
3/4 2-9	3/4 2-9
3/4 2-10	3/4 2-10
3/4 2-10a	3/4 2-10a
3/4 3-5	3/4 3-5
3/4 3-5a	3/4 3-5a
3/4 3-6a	3/4 3-6a
3/4 3-6b	3/4 3-6b
3/4 3-6c	3/4 3-6c
B 3/4 2-1	B 3/4 2-1
B 3/4 2-3	B 3/4 2-3
B 3/4 3-1	B 3/4 3-1

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	$\leq$ 110% of RATED THERMAL POWER	$\leq$ 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)	$\leq$ 0.75% of RATED THERMAL POWER	$\leq$ 0.819% of RATED THERMAL POWER
4. Pressurizer Pressure - High	$\leq$ 2362 psia	$\leq$ 2370.887 psia
5. Pressurizer Pressure - Low	$\geq$ 1766 psia (2)	$\geq$ 1712.757 psia (2)
6. Containment Pressure - High	$\leq$ 18.4 psia	$\leq$ 19.024 psia
7. Steam Generator Pressure - Low	$\geq$ 751 psia (3)	$\geq$ 729.613 psia (3)
8. Steam Generator Level - Low	$\geq$ 23% (4)	$\geq$ 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$21.0 kw/ft (5)	$\leq$21.0 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.

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 21.0 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.

BASES

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 305-P, "Functional Design Requirement for a Core Protection Calculator," July 1985; CEN-304-P, "Functional Design Requirements for a Control Element Assembly Calculator," July 1985; CEN-310-P, "CPC and Methodology Changes for the CPC Improvement Program," October 1985 and CEN-308-P, "CPC/CEAC Software Modifications for the CPC Improvement Program," August 1985.

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 $\leq$ 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 $\geq$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 $\leq$ 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 $\leq$ 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. 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.

BASES

a.	RCS Cold Leg Temperature-Low	$\geq$ 490°F
b.	RCS Cold Leg Temperature-High	$\leq$ 585°F
c.	Axial Shape Index-Positive	Not more positive than +0.6
d.	Axial Shape Index-Negative	Not more negative than -0.6
e.	Pressurizer Pressure-Low	$\geq$ 1785 psia
f.	Pressurizer Pressure-High	$\leq$ 2415 psia
g.	Integrated Radial Peaking Factor-Low	$\geq$ 1.28
h.	Integrated Radial Peaking Factor-High	$\leq$ 4.28
i.	Quality Margin-Low	$\geq$ 0

Steam Generator Level-High

The Steam Generator Level-High trip is provided to protect the turbine from excessive moisture carryover. Since the turbine is automatically tripped when the reactor is tripped, this trip provides a reliable means for providing protection to the turbine from excessive moisture carry over. This trip's setpoint does not correspond to a Safety Limit and no credit was taken in the accident analyses for operation of this trip. Its functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protection System.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 LINEAR HEAT RATE

LIMITING CONDITION FOR OPERATION

3.2.1 The linear heat rate limit shall be maintained by either:

a. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on linear heat rate (when COLSS is in service); or

b. Operating within the region of acceptable operation of Figure 3.2-1 using any operable CPC Channel (when COLSS is out of service).

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER

ACTION:

With the linear heat rate limit not being maintained as indicated by either:

1. COLSS calculated core power exceeding COLSS calculated core power operating limit based on linear heat rate; or

2. Operation outside the region of acceptable operation in Figure 3.2-1, when COLSS is out of service,

within 15 minutes initiate corrective action to reduce the linear heat rate to within the limits and either:

a. Restore the linear heat rate to within its limits within one hour, or

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

SURVEILLANCE REQUIREMENTS

4.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.1.2 The linear heat rate 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 linear heat rate, as indicated on any OPERABLE CPC channel, is within the limit shown on Figure 3.2-1.

4.2.1.3 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 linear heat rate.

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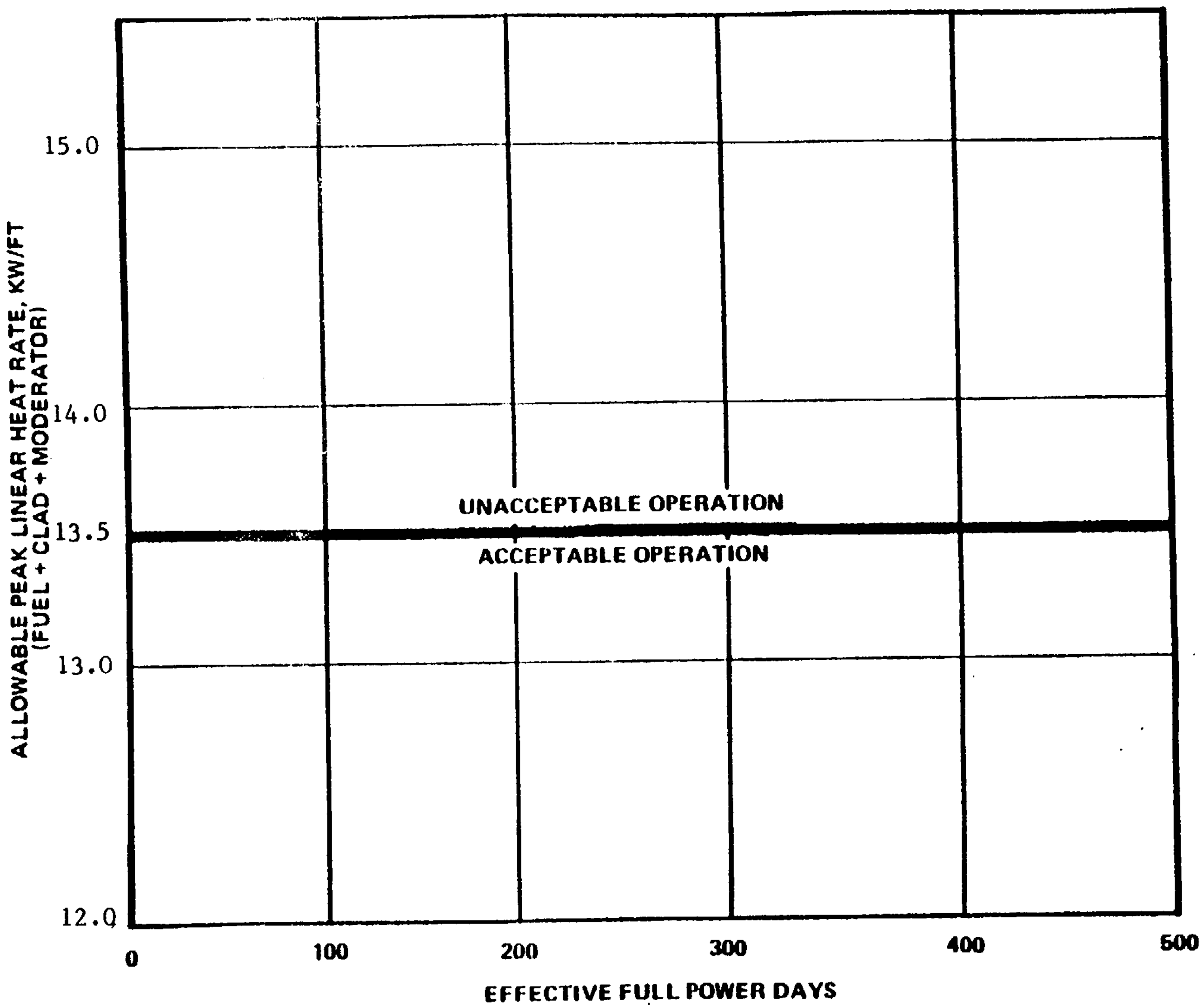

Figure 3.2-1 Allowable Peak Linear Heat Rate vs Burnup
(COLSS out of service)

POWER DISTRIBUTION LIMITS

RADIAL PEAKING FACTORS

LIMITING CONDITION FOR OPERATION

3.2.2 The measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}) shall be less than or equal to the PLANAR RADIAL PEAKING FACTORS (F^c_{xy}) used in the Core Operating Limit Supervisory System (COLSS) and in the Core Protection Calculators (CPC).

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

ACTION:

With a F^m_{xy} exceeding a corresponding F^c_{xy} within 6 hours either:

a. Adjust the CPC addressable constants to increase the multiplier applied to PLANAR RADIAL PEAKING FACTOR by a factor equivalent to $\geq F^m_{xy}/F^c_{xy}$ and restrict subsequent operation so that a margin to the COLSS operating limits of at least $[(F^m_{xy}/F^c_{xy}) - 1.0] \times 100\%$ is maintained; or

b. Adjust the affected PLANAR RADIAL PEAKING FACTORS (F^c_{xy}) used in the COLSS and CPC to a value greater than or equal to the measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}); or

c. Be in at least HOT STANDBY.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 The measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}), obtained by using the incore detection system, shall be determined to be less than or equal to the PLANAR RADIAL PEAKING FACTORS (F^c_{xy}) used in the COLSS and CPC at the following intervals:

a. After each fuel loading with THERMAL POWER greater than 40% but prior to operation above 70% of RATED THERMAL POWER, and

b. At least once per 31 days of accumulated operation in MODE 1.

---

*See Special Test Exception 3.10.2.

3/4.2.4 DNBR MARGIN

LIMITING CONDITIONS FOR OPERATION

3.2.4 The DNBR limit shall be maintained by one of the following methods:

a. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on DNBR (when COLSS is in service, and at least one CEAC is operable); or

b. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on DNBR decreased by 13.0% (when COLSS is in service and neither CEAC is operable); or

c. Operating within the region of acceptable operation of Figure 3.2-2 using any operable CPC channel (when COLSS is out of service and at least one CEAC is operable); or

d. Operating within the region of acceptable operation of Figure 3.2-3 using any operable CPC channel (when COLSS is out of service and neither CEAC is operable).

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

ACTION:

With the DNBR limit not being maintained as indicated by either:

1. COLSS calculated core power exceeding COLSS calculated core power operating limit based on DNBR; or

2. Operation outside the region of acceptable operation of Figure 3.2-2 or 3.2-3 as applicable, when COLSS is out of service,

within 15 minutes initiate corrective action to restore 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.

SURVEILLANCE REQUIREMENTS (continued)

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 any OPERABLE CPC channel, is within the limit shown on Figures 3.2-2 or 3.2-3, as applicable.

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

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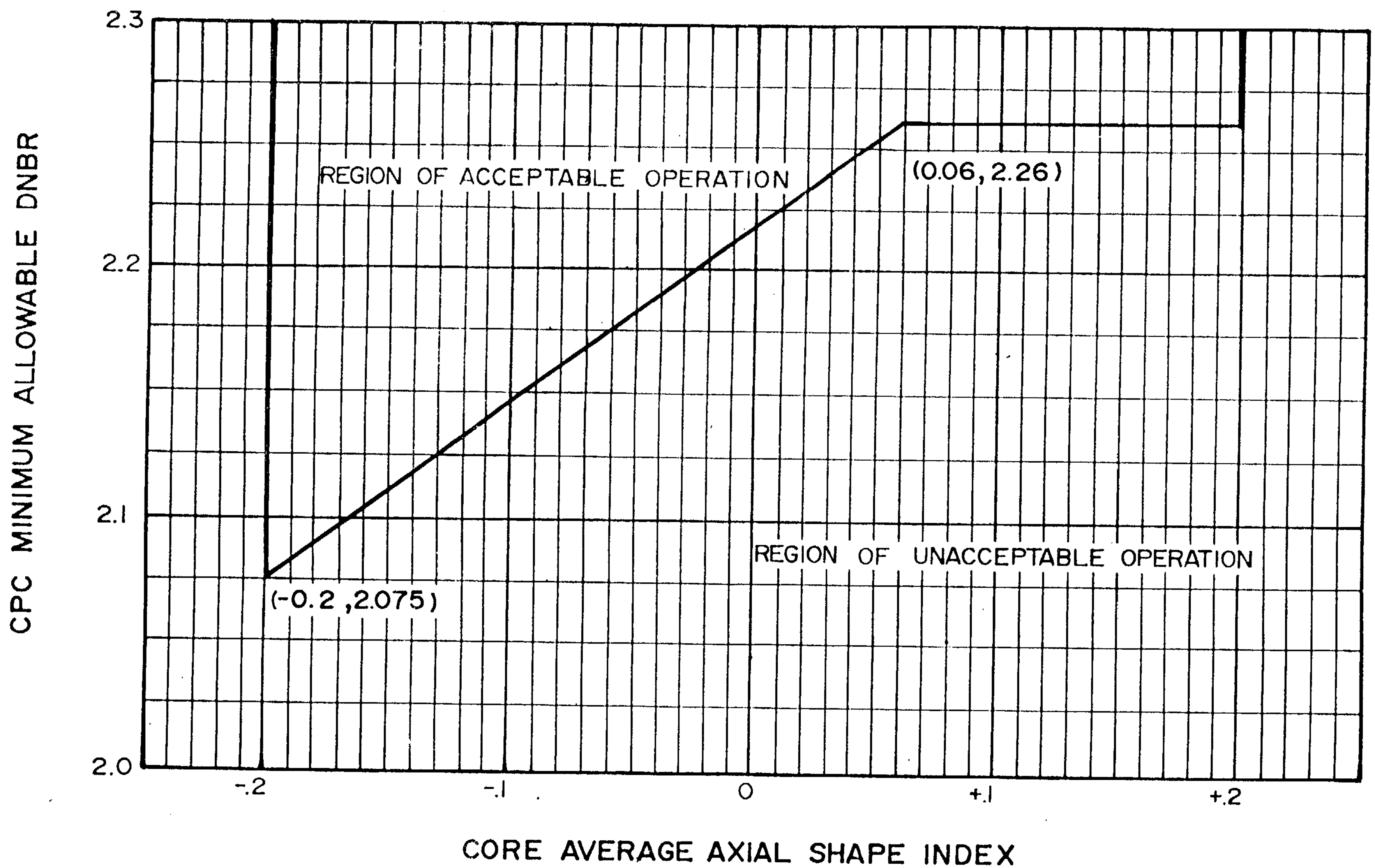

FIGURE 3.2-2
DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS
(COLSS OUT OF SERVICE, CEAC OPERABLE)

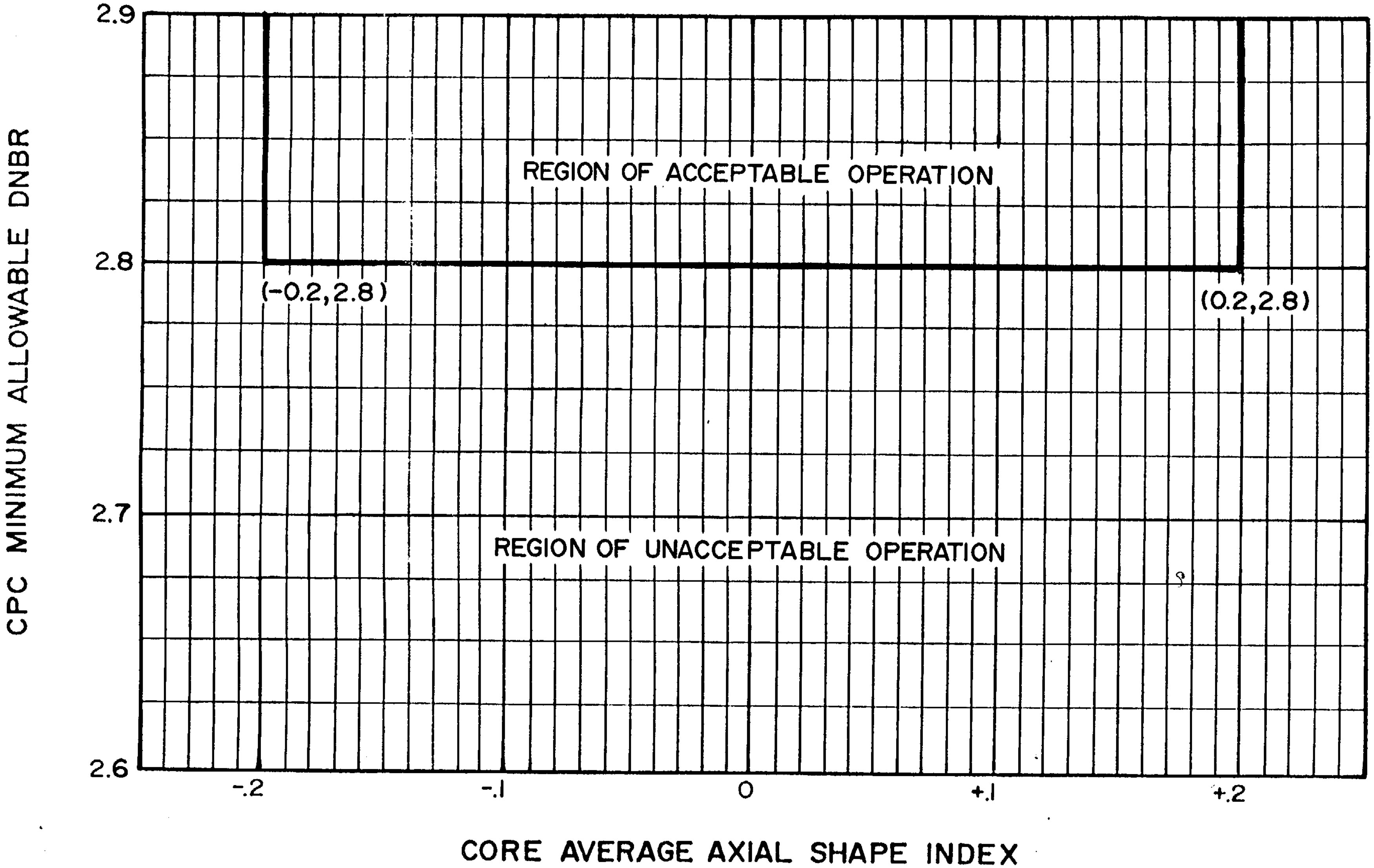

FIGURE 3.2-3
DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS
(COLSS OUT OF SERVICE, BOTH CEACS INOPERABLE)

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. For the purposes of testing and maintenance, the inoperable channel may be bypassed for up to 48 hours from time of initial loss of OPERABILITY; however, the inoperable channel shall then be either restored to OPERABLE status or placed in the tripped condition.

b. Within one hour, all functional logic units receiving an input from the inoperable channel are also placed in the same condition (either bypassed or tripped, as applicable) as that required by a. above for the inoperable channel.

c. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 48 hours while performing tests and maintenance on that channel provided the other inoperable channel is placed in the tripped condition.

ACTION 3 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, place the inoperable channel in the tripped condition within 1 hour or be in HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing per Specification 4.3.1.1.1

ACTION 5 - a. With one CEAC inoperable, operation may continue for up to 7 days provided that at least once per 4 hours, each CEA is verified to be within 7 inches (indicated position) of all other CEAs in its group. After 7 days, operation may continue provided that ACTION 5.b is met.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

b. With both CEACs inoperable, operation may continue provided that:

   1. Within 1 hour the margin required by Specification 3.2.4.b (COLSS in service) or Specification 3.2.4.d (COLSS out of service) is satisfied.

   2. Within 4 hours:

      a) All full length and part length CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 6 may be inserted no further than 127.5 inches withdrawn.

      b) The "RSPT/CEAC Inoperable" addressable constant in the CPCs is set to both CEACs inoperable.

      c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Off" mode except during CEA motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

   3. At least once per 4 hours, all full length and part length CEAs are verified fully withdrawn, except as permitted by 2. a) above, then verify at least once per 4 hours that the inserted CEAs are aligned within 7 inches (indicated position) of all other CEAs in their group.

ACTION 6 - With three or more auto restarts of one non-bypassed calculator during a 12-hour interval, demonstrate calculator OPERABILITY by performing a CHANNEL FUNCTIONAL TEST within the next 24 hours.

TABLE 3.3-2

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	Not Applicable
2. Linear Power Level - High	≤ 0.40 seconds*
3. Logarithmic Power Level - High	≤ 0.40 seconds*
4. Pressurizer Pressure - High	≤ 0.90 seconds
5. Pressurizer Pressure - Low	≤ 0.90 seconds
6. Containment Pressure - High	≤ 1.59 seconds
7. Steam Generator Pressure - Low	≤ 0.90 seconds
8. Steam Generator Level - Low	≤ 0.90 seconds
9. Local Power Density - High	
a. Neutron Flux Power from Excore Neutron Detectors	≤ 2.58 seconds*
b. CEA Positions	≤ 1.58 seconds**

TABLE 3.3-2 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
10. DNBR - Low	
a. Neutron Flux Power from Excore Neutron Detectors	$\leq$ 0.39 seconds*
b. CEA Positions	$\leq$ 1.09 seconds**
c. Cold Leg Temperature	$\leq$ 3.79 seconds##
d. Hot Leg Temperature	$\leq$ 1.54 seconds###
e. Primary Coolant Pump Shaft Speed	$\leq$ 0.80 seconds#
f. Reactor Coolant Pressure from Pressurizer	$\leq$ 3.19 seconds
11. Steam Generator Level - High	Not Applicable

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

**Response time shall be measured from the onset of a single CEA drop.

#Response time shall be measured from the onset of a 2 out of 4 Reactor Coolant Pump coastdown.

##Based on an effective resistance temperature detector (RTD) response time of $\leq$ 8.0 seconds. If the effective RTD time constant for a CPC channel exceeds 8.0 seconds, the DNBR and LPD penalties for the affected channel(s) shall be increased by the amount indicated on Figure 3.3-1.

###Based on an effective RTD response time of $\leq$ 13.0 seconds.

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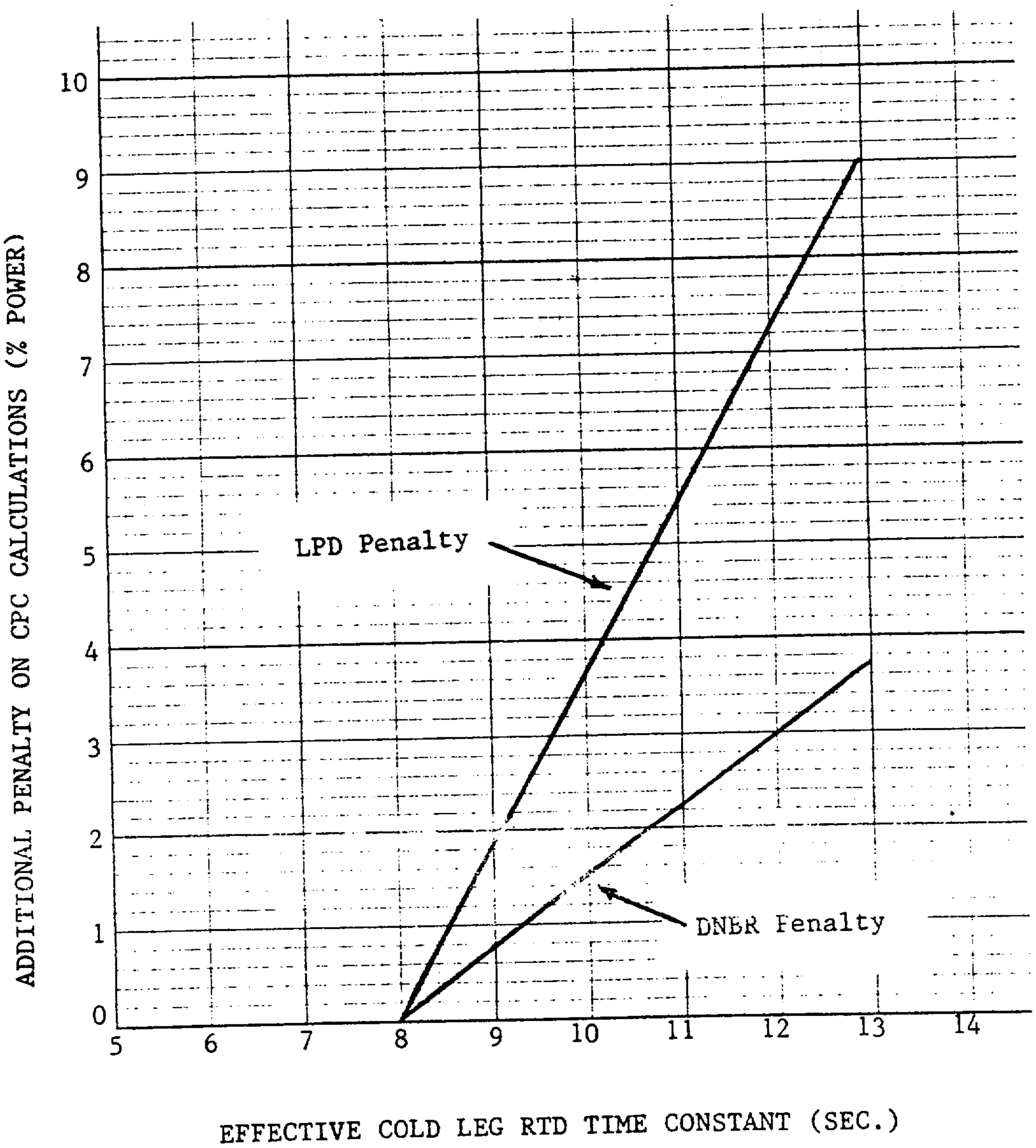
FIGURE 3.3-1

CPC PENALTY VS. EFFECTIVE RTD TIME CONSTANT



3/4.2.1 LINEAR HEAT RATE

The limitation on linear heat rate ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200°F.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the Local Power Density channels in the Core Protection Calculators (CPCs), provide adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core power operating limit corresponding to the allowable peak linear heat rate.

The COLSS calculated core power and the COLSS calculated core power operating limits based on linear heat rate are continuously monitored and displayed to the operator. A COLSS alarm is annunciated in the event that the core power exceeds the core power operating limit. This provides adequate margin to the linear heat rate operating limit for normal steady state operation. Normal reactor power transients or equipment failures which do not require a reactor trip may result in this core power operating limit being exceeded. In the event this occurs, COLSS alarms will be annunciated. If the event which causes the COLSS limit to be exceeded results in conditions which approach the core safety limits, a reactor trip will be initiated by the Reactor Protective Instrumentation. The COLSS calculation of the linear heat rate limit includes appropriate uncertainty and penalty factors necessary to provide a 95/95 confidence level that the maximum linear heat rate calculated by COLSS is greater than or equal to that existing 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 rod bow.

Parameters required to maintain the operating limit power level based on linear heat rate, 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 3.2-1 can be maintained by utilizing a predetermined local power density margin and a total core power limit in the CPC trip channels. The above listed uncertainty and penalty factors are also included in the CPCs.

3/4.2.2 RADIAL PEAKING FACTORS

Limiting the values of the PLANAR RADIAL PEAKING FACTORS (F^c_{xy}) used in the COLSS and CPCs to values equal to or greater than the measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}) provides assurance that the limits calculated by COLSS and the CPCs remain valid. Data from the incore detectors are used for determining the measured PLANAR RADIAL PEAKING FACTORS. The periodic surveillance requirements for determining the measured PLANAR RADIAL PEAKING FACTORS provides assurance that the PLANAR RADIAL PEAKING FACTORS used in COLSS and the CPCs remain valid throughout the fuel cycle. Determining the measured PLANAR RADIAL PEAKING FACTORS after each fuel loading prior to exceeding 70% of RATED THERMAL POWER provides additional assurance that the core was properly loaded.

3/4.2.3 AZIMUTHAL POWER TILT - T_q

The limitations on the AZIMUTHAL POWER TILT are provided to ensure that design safety margins are maintained. An AZIMUTHAL POWER TILT greater than 0.10 is not expected and if it should occur, operation is restricted to only those conditions required to identify the cause of the tilt. The tilt is normally calculated by COLSS. The surveillance requirements specified when COLSS is out of service provide an acceptable means of detecting the presence of a steady state tilt. It is necessary to explicitly account for power asymmetries because the radial peaking factors used in the core power distribution calculations are based on an untilted power distribution.

AZIMUTHAL POWER TILT is measured by assuming that the ratio of the power at any core location in the presence of a tilt to the untilted power at the location is of the form:

$$P_{tilt}/P_{untilt} = 1 + T_q \, g \cos (\Theta - \Theta_o)$$

where:

T_q is the peak fractional tilt amplitude at the core periphery

g is the radial normalizing factor

Θ is the azimuthal core location

Θ_o is the azimuthal core location of maximum tilt

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. 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 any anticipated operational occurrence.

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. The COLSS calculation of core power operating limit based on DNBR includes appropriate uncertainity 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 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 3.2-2 for CEAC operable or Figure 3.2-3 for both CEACs inoperable 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.

3/4.3.1 and 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION

The OPERABILITY of the protective and ESF instrumentation systems and bypasses ensure that 1) the associated ESF action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available for protective and ESF purposes from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the accident analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

RTD response time is defined as the time interval required for the RTD output to achieve 63.2% of its total change when subjected to a step change in RTD temperature. The RTD response time for the Core Protection Calculator System (CPCS) is expressed as an effective time constant. For hot leg temperatures, the effective time constant for a given CPC channel is defined as the mean time constant for averaged pairs of hot leg RTD inputs to the channel. This is done because the CPCS utilizes the mean hot leg temperature in its calculations. The maximum hot leg effective time constant allowable for use in the CPCS is 13.0 seconds. For cold leg temperatures, the effective time constant to be used in Figure 3.3-1 is the maximum time constant of the two cold leg RTD inputs for a given channel. The CPCS utilizes the more conservative cold leg temperature in the various DNBR and LPD calculations. The maximum cold leg effective time constant allowable for use in the CPCS is 13.0 seconds.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

3/4.3.3.2 INCORE DETECTORS

The OPERABILITY of the incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magniture of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix "A" of 10 CFR Part 100. The instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO FACILITY OPERATING LICENSE NO. NPF-6

ARKANSAS POWER AND LIGHT COMPANY

ARKANSAS NUCLEAR ONE, UNIT 2

DOCKET NO. 50-368

1.0 INTRODUCTION

By letter dated February 27, 1986, Arkansas Power and Light Company (AP&L) submitted a request to revise the Technical Specifications, Appendix A to Facility Operating License No. NPF-6 for the Arkansas Nuclear One - Unit 2 (ANO-2) plant. The proposed Technical Specification revisions are necessary in order to implement the Core Protection Calculator (CPC) Improvement Program at ANO-2. The CPC Improvement Program has been developed by the CPC Oversight Committee, consisting of AP&L, Arizona Nuclear Power Project, Louisiana Power and Light Company and Southern California Edison, with Combustion Engineering (CE) as the technical consultant. Several goals of the program include the implementation of appropriate modifications and methodology improvements to reduce future reload efforts as well as to reduce unnecessary plant trips.

By letter dated April 21, 1986, AP&L also submitted responses to the NRC request for additional information concerning the proposed Technical Specification changes. The staff's evaluation of these proposed changes follows.

2.0 EVALUATION

The trip setpoint and allowable value for the local power density (LPD) high trip has been changed to 21.0 kw/ft. The change revised Table 2.2-1, "Reactor Protective Instrumentation Trip Setpoint Limits," of Technical Specification 2.2.1 and its associated Bases. The LPD trip setpoint specifies the setpoint required to prevent the peak linear heat rate, in the limiting fuel pin in the core, from exceeding the value which corresponds to the centerline fuel melting temperature during anticipated operational occurrences. It also assists in mitigating the consequences of accidents. The modification increases the trip setpoint from 20.3 kw/ft to 21.0 kw/ft, which is the linear heat generation rate corresponding to fuel centerline melting as determined by approved methods. Previously, the trip setpoint incorporated an adjustment for dynamic effects that will now be accounted for elsewhere in the CPC algorithms. Also, the

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21.0 kw/ft value has been previously approved for the San Onofre Units 2 and 3, the Palo Verde Units 1 and 2 and the CESSAR System 80 Technical Specification LPD trip setpoint. Based on this and on the fact that, effectively, the CPC LPD protection is not being changed, the modification is acceptable.

The Bases for Technical Specification 2.2.1 has been updated to reflect the appropriate CPC Improvement Program modifications and methodology documents which have been reviewed and approved by the NRC. In addition, the CPC range limits have been modified to generic range limits specified by the CPC Improvement Program. The RCS Cold Leg Temperature-Low limit has been changed from 465°F to 490°F and the High limit from 605°F to 585°F. The Pressurizer Pressure-Low limit has been changed from 1750 psia to 1785 psia and the High limit from 2400 psia to 2415 psia. These changes will reduce the range of inputs (and, therefore, the calculational complexity) of the safety analyses and promote consistency between the software of CPC plants. The changes to the Bases of 2.2.1 are, therefore, acceptable.

The previous Limiting Condition for Operation (LCO) of Technical Specification 3.2.1, "Linear Heat Rate," has been replaced with two parts, 3.2.1.a and 3.2.1.b. Specification 3.2.1.a is applicable when COLSS (Core Operating Limits Supervisory System) is in service and requires the COLSS calculated core power to be maintained less than or equal to the COLSS calculated Power Operating Limit (POL) based on linear heat rate. These words replace the old Figure 3.2-1 and state the same requirement. The change is, therefore, acceptable. Specification 3.2.1.b is applicable when COLSS is out of service and requires operation within the allowable region of the new Figure 3.2-1 using any operable CPC channel. The old Figure 3.2-1 has been deleted and replaced with the words of Specification 3.2.1.a which state the same requirements. The new Figure 3.2-1 is similar to the previous Figure 3.2-2 and gives the allowable peak linear heat rate as a function of burnup when COLSS is out of service. The linear heat rate limit value in this figure has been lowered from 14.5 kw/ft to 13.5 kw/ft in order to bound future cycles. Since this lower limit is more restrictive and in a conservative direction, the change is acceptable.

The previous LCO of Technical Specification 3.2.4, "DNBR Margin," has been replaced with four parts, a, b, c, and d. Specification 3.2.4.a is applicable when COLSS is in service and at least one Control Element Assembly Calculator (CEAC) is operable and requires the COLSS calculated core power to be maintained less than or equal to the COLSS calculated POL based on DNBR. This is consistent with the previous requirement except that Figure 3.2-3 has been deleted and replaced with words to the same effect. The change is acceptable. Specification 3.2.4.b is applicable when COLSS is in service and neither CEAC is operable and requires the COLSS calculated core power to be maintained less than or equal to the COLSS calculated POL based on DNBR decreased by a penalty factor of 13% of rated power.

The 13% penalty factor is calculated based on a single group 6 CEA withdrawal event and accounts for the power rise and increased core power peaking during the event. It provides sufficient margin for CEA misoperation events during periods when neither CEAC is operable and the CPC system does not have CEA position information and is, therefore, acceptable. Specification 3.2.4.c is applicable when COLSS is out of service and at least one CEAC is operable. In this case, the CPC calculated DNBR is maintained within the limits of a new Figure 3.2-2. This is identical to the old Figure 3.2-4 and the change is acceptable. Specification 3.2.4.d applies when COLSS is out of service and neither CEAC is operable. In this case, the CPC calculated DNBR is maintained within a new Figure 3.2-3. This new figure accommodates the increased margin required when both CEACs are inoperable and the change is, therefore, acceptable. The staff also concludes that it does not matter which of the four CPC channels is chosen to monitor DNBR when COLSS is out of service since appropriate uncertainty allowances are already implemented in the CPC system calculations and trip setpoints. The four redundant CPC channels are still required for protection purposes to ensure appropriate protective action during transients and accidents.

Action 5 of Table 3.3-1 in Technical Specification 3.3.1 has been revised. Action 5 provides conditions under which operation may continue for various operability conditions of the CEACs. Action 5.a is revised to allow operation with one inoperable CEAC to continue after 7 days provided Action 5.b is met. Since Action 5.b addresses operation with both CEACs inoperable, it requires more restrictive actions to be taken. The change is, therefore, acceptable. In addition, Action 5.b is revised to reference the appropriate requirement of Specification 3.2.4, as described and approved above, and to clarify the setting of the "RSPT/CEAC Inoperable" addressable constant in the CPCs. These changes are also acceptable.

The resistance temperature detector (RTD) response time of 6 seconds for a CPC channel, shown in the footnote to Table 3.3-2 of Technical Specification 3.3.1, has been revised. The CPC Improvement Program and related accident analyses now assume effective response times of 8 seconds for the RCS cold leg temperature RTDs and 13 seconds for the RCS hot leg temperature RTDs. The footnotes to Table 3.3-2 are revised accordingly. Figure 3.3-1 is also revised to give the DNBR and linear heat rate penalties to be applied if the effective cold leg RTD response time falls between 8 seconds and 13 seconds. Table 3.3-3 is deleted since the analyses now assumes response times up to 13 seconds and adjustments are no longer required for RTD response times less than this. These changes are acceptable.

3.0 EVALUATION SUMMARY

The staff has concluded, based on the considerations discussed above, that the proposed changes to the ANO-2 Technical Specifications are acceptable.

4.0 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.

5.0 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.

Principal contributor to this SE was L. Kopp

Dated: July 22, 1986