Docket File
DMB
MS-016

JUN 19 1981

Docket No.: 50-368

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

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JUN 25 1981
U.S. NUCLEAR REGULATORY COMMISSION

See Rept.

Dear Mr. Cavanaugh:

SUBJECT: OPERATION OF ANO-2 DURING CYCLE 2

The Commission has issued the enclosed Amendment No. 24 to Facility Operating License No. NPF-6 for the Arkansas Nuclear One - Unit 2 plant. This amendment consists of changes to the license in accordance with the satisfactory completion of certain conditions to the license. It also consists of changes to the Technical Specifications in accordance with your Cycle 2 Reload Report and request dated February 20 and March 5, 1981 as supplemented by information identified in the Reference Section of the attached Safety Evaluation.

This amendment authorizes Cycle 2 operation subject to the condition in the license which temporarily restricts operation of the facility to seventy percent of the licensed full power level of 2815 MWt pending completion of the staff's review of the core protection calculator system, and with the following changes.

- Changes in the Core Protection Calculator System (CPCS) to reflect utilization of the CE-1 critical heat flux correlation and associated thermal hydraulic methodology.
- Changes in the CPCS to reflect utilization of the Statistical Combination of Uncertainties (SCU) thermal hydraulic methodology for the combination of system parameter uncertainties.
- Changes in the RPS and ESFAS trip setpoints to reflect a change in signal transmitter design and to reflect staff approval of the licensee's equipment trip setpoints.
- Changes in the minimum required shutdown margin to lenghten the time available for operator action during a boron dilution event.
- Changes required to maintain acceptable results for the steamline break analysis.

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- Some demonstration fuel assemblies to test new fuel designs.

- Numerous other miscellaneous changes of a clarifying, editorial and administrative nature.

- Other changes in the Technical Specification to incorporate requirements resulting from the detailed physics and thermal hydraulic analysis of the Cycle 2 reload core.

Some portions of your proposed Technical Specifications have been modified to meet our requirements. These modifications have been discussed with and agreed to by your staff.

The license is also modified by this amendment to reflect completion of the matters addressed by the following license conditions.

2.C.3.a Fuel Performance

2.C.3.d Instrument Trip Setpoints Drift Allowance

2.C.3.f Overpressure Mitigating System

2.C.3.l CEA Guide Tube Surveillance Program

The enclosed Safety Evaluation supporting this amendment addresses our evaluation of the satisfaction of the above license conditions and also addresses our evaluation of:

- Limiting containment pressure, temperature and relative humidity to control the differential pressure in event of an inadvertent containment spray actuation.

- High pressurizer pressure trip setpoint.

In the process of our evaluation of your request, we find the following items need your attention as documented herein. Some of these requested analyses result from expanded staff reviews and will be requested of other licensees to coincide with their reload reviews.

1. Provide a positive means to alert the control room operators of a boron dilution event when the reactor is shutdown. Your description of such a means would be submitted to the staff within 120 days of the date of this amendment. If the positive means involves the installation of hardware, it should be completed as soon as practical.

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2. Provide analysis of the reactor coolant pump shaft seizure event taking into consideration the single failure criterion. Results of the analysis including calculation of the radiological consequences should be provided on a timely schedule.

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

Sincerely,

Original signed by
Robert A. Clark

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

Enclosures:
1. Amendment No. 24 to NPF-6
2. Safety Evaluation
3. Notice of Issuance

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See next page

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

June 19, 1981

Docket No.: 50-368

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

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- Numerous other miscellaneous changes of a clarifying, editorial and administrative nature.

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Sincerely,

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

Enclosures:
1. Amendment No. 24 to NPF-6
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures
See next page

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Washington, D. C. 20460

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Region VI Office
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cc w/enclosure(s) and incoming
dated: 2/6/81, 3/5/81

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

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

ARKANSAS POWER AND LIGHT COMPANY

ARKANSAS NUCLEAR ONE, UNIT NO. 2

DOCKET NO. 50-368

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 24
License No. NPF-6

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

   A. The application for amendment by Arkansas Power and Light Company (the licensee) dated February 20 and March 5, 1981, as supplemented, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by deletion of License Conditions 2.C.3.a, d, f, and i and changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraphs 2.C.(1) and 2.C.(2) of Facility Operating License No. NPF-6 are hereby amended to read as follows:

   (1) Maximum Power Level

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 24 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: June 19, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 24

FACILITY OPERATING LICENSE NO. NPF-6

DOCKET NO. 50-368

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

Pages

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DEFINITIONS

# INDEX

## SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS



## BASES

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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BASES

INDEX

BASES

## $\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.19 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MEV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

## STAGGERED TEST BASIS

1.20 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

## FREQUENCY NOTATION

1.21 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

## AXIAL SHAPE INDEX

1.22 The AXIAL SHAPE INDEX shall be the power generated in the lower half of the core less the power generated in the upper half of the core divided by the sum of these powers.

## REACTOR TRIP SYSTEM RESPONSE TIME

1.23 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until electrical power is interrupted to the CEA drive mechanism.

ENGINEERED SAFETY FEATURE RESPONSE TIME

1.24 The ENGINEERED SAFETY FEATURE RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

PHYSICS TESTS

1.25 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

SOFTWARE

1.26 The digital computer SOFTWARE for the reactor protection system shall be the program codes including their associated data, documentation and procedures.

PLANAR RADIAL PEAKING FACTOR - $F_{xy}$

1.27 The PLANAR RADIAL PEAKING FACTOR is the ratio of the peak to plane average power density of the individual fuel rods in a given horizontal plane, excluding the effects of azimuthal tilt.

# 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 $\geq$ 1.24.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the DNBR of the reactor core has decreased to less than 1.24, 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.

## 2.2 LIMITING SAFETY SYSTEM SETTINGS

### REACTOR TRIP SETPOINTS

2.2.1 The reactor protective instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a reactor protective instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

## 2.2 LIMITING SAFETY SYSTEM SETTINGS (Continued)

### CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

2.2.2 Core Protection Calculator Addressable Constants are defined in Table 2.2-2. Type I Addressable Constants are expected to change frequently during plant operation. Type II Addressable Constant values are determined (or confirmed) during PHYSICS TESTS following each fuel loading and are not expected to change during plant operation. Changes to Type I Addressable Constants outside the Allowable Value range require Plant Safety Committee review prior to implementation. Changes to Type II Addressable Constants made other than as a result of post fuel loading PHYSICS TESTS shall require Plant Safety Committee review prior to implementation unless the changes are required for Technical Specification Compliance.

APPLICABILITY: As shown for Core Protection Calculators in Table 3.3-1.

ACTION: With a Core Protection Calculator Addressable Constant found to be non-conservative, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 until the channel is restored to OPERABLE status.

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

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

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

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

| FUNCTIONAL UNIT | TRIP SETPOINT | ALLOWABLE VALUES |
|---|---|---|
| 9. Local Power Density - High | $\leq$ 20.3 kw/ft (5) | $\leq$ 20.3 kw/ft (5) |
| 10. DNBR - Low | $\geq$ 1.24 (5) | $\geq$ 1.24 (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.

(6) The minimum allowable value of the addressable constant BERR1 in each OPERABLE channel is 1.174. Upon NRC approval of the Statistical Combination of Uncertainties methodology as described in CEN-139(A)-P, the minimum allowable value of BERR1 is 1.055 .

## 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 | ≥0.90 |
| 65 | KCAL | Neutron flux power calibration constant | ≥0.85 |
| 66 | DNBRPT | DNBR pretrip setpoint | Unrestricted |
| 67 | LPDPT | Local power density pretrip setpoint | Unrestricted |

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 |
| 98 | CORR1 | Temperature shadowing correction factor multiplier |
| 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 |

## 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 1.24 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 1.24 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.

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. $\Delta T$ 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 ($\Delta T$) 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 1.24 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.

| | | |
|---|---|---|
| a. | RCS Cold Leg Temperature-Low | ≥ 465°F |
| b. | RCS Cold Leg Temperature-High | ≤ 605°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 | ≥ 1750 psia |
| f. | Pressurizer Pressure-High | ≤ 2400 psia |
| g. | Integrated Radial Peaking Factor-Low | ≥ 1.28 |
| h. | Integrated Radial Peaking Factor-High | ≤ 4.28 |
| i. | Quality Margin-Low | ≥ 0 |

Steam Generator Level-High

The Steam Generator Level-High trip is provided to protect the turbine from excessive moisture carry over. 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.

2.2.2 CPC Addressable Constants

The Core Protection Calculator (CPC) addressable constants are provided to allow calibration of the CPC system to more accurate indications such as calorimetric measurements for power level and RCS flowrate and incore detector signals for axial flux shape, radial peaking factors and CEA deviation penalties. Other CPC addressable constants allow penalization of the calculated DNBR and LPD values based on measurement uncertainties or inoperable equipment. Administrative controls on changes and periodic checking of addressable constant values (see also Technical Specifications 3.3.1.1 and 6.8.1) ensures that inadvertent misloading is unlikely. The methodology for determination of CPC addressable constant values is described in AP&L letter 2CAN058113 dated May 26, 1981.

## SHUTDOWN MARGIN - $T_{avg} \leq 200°F$

### LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be $\geq$ 5.0% $\Delta k/k$.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN < 5.0% $\Delta k/k$, immediately initiate and continue boration at $\geq$ 40 gpm of 1731 ppm boric acid solution or equivalent until the required SHUTDOWN MARGIN is restored.

### SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be $\geq$ 5.0% $\Delta k/k$:

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

b. At least once per 24 hours by consideration of at least the following factors:

   1. Reactor coolant system boron concentration,
   2. CEA position,
   3. Reactor coolant system average temperature,
   4. Fuel burnup based on gross thermal energy generation,
   5. Xenon concentration, and
   6. Samarium concentration.

## BORON DILUTION

### LIMITING CONDITION FOR OPERATION

3.1.1.3 The flow rate of reactor coolant through the reactor coolant system shall be ≥ 3000 gpm whenever a reduction in Reactor Coolant System boron concentration is being made.

APPLICABILITY: ALL MODES.

ACTION:

With the flow rate of reactor coolant through the reactor coolant system < 3000 gpm, immediately suspend all operations involving a reduction in boron concentration of the Reactor Coolant System.

### SURVEILLANCE REQUIREMENTS

4.1.1.3 The flow rate of reactor coolant through the reactor coolant system shall be determined to be ≥ 3000 gpm within one hour prior to the start of and at least once per hour during a reduction in the Reactor Coolant System boron concentration by either:

a. Verifying at least one reactor coolant pump is in operation, or

b. Verifying that at least one low pressure safety injection pump is in operation and supplying ≥ 3000 gpm through the reactor coolant system.

## MODERATOR TEMPERATURE COEFFICIENT

### LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be:

a. Less positive than $0.5 \times 10^{-4}$ $\Delta k/k/^{o}F$ whenever THERMAL POWER is $\leq$70% of RATED THERMAL POWER,

b. Less positive than 0.0 $\Delta k/k/^{o}F$ whenever THERMAL POWER is >70% of RATED THERMAL POWER, and

c. Less negative than $-2.8 \times 10^{-4}$ $\Delta k/k/^{o}F$ at RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*.

ACTION:

With the moderator temperature coefficient outside any one of the above limits, be in at least HOT STANDBY within 6 hours.

### SURVEILLANCE REQUIREMENTS

4.1.1.4.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

4.1.1.4.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. At any THERMAL POWER, within 7 EFPD after reaching a RATED THERMAL POWER equilibrium boron concentration of 800 ppm.

c. At any THERMAL POWER, within 7 EFPD after reaching a RATED THERMAL POWER equilibrium boron concentration of 300 ppm.

---

*With $K_{eff} \geq 1.0$.

## MINIMUM TEMPERATURE FOR CRITICALITY

### LIMITING CONDITION FOR OPERATION

3.1.1.5 The Reactor Coolant System lowest operating loop temperature ($T_{avg}$) shall be $\geq$ 525°F when the reactor is critical.

APPLICABILITY: MODES 1 and 2#*.

ACTION:

With a Reactor Coolant System operating loop temperature ($T_{avg}$) < 525°F, restore $T_{avg}$ to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

### SURVEILLANCE REQUIREMENTS

4.1.1.5 The Reactor Coolant System temperature ($T_{avg}$) shall be determined to be $\geq$ 525°F:

a. Within 15 minutes prior to achieving reactor criticality, and

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System $T_{avg}$ is less than 535°F.

---

#With $K_{eff} \geq 1.0$.

*See Special Test Exception 3.10.5.

## 3/4.1.2 BORATION SYSTEMS

### FLOW PATHS - SHUTDOWN

### LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths and one associated heat tracing circuit shall be OPERABLE:

a. A flow path from the boric acid makeup tank via either a boric acid makeup pump or a gravity feed connection and charging pump to the Reactor Coolant System if only the boric acid makeup tank in Specification 3.1.2.7a is OPERABLE, or

b. The flow path from the refueling water tank via either a charging pump or a high pressure safety injection pump to the Reactor Coolant system if only the refueling water tank in Specification 3.1.2.7b is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one injection path is restored to OPERABLE status.

### SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperature of the heat traced portion of the flow path is above the temperature limit line shown on Figure 3.1-1 when a flow path from the boric acid makeup tanks is used.

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

## FLOW PATHS - OPERATING

### LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths and one associated heat tracing circuit shall be OPERABLE:

a. Two flow paths from the boric acid makeup tanks via either a boric acid makeup pump or a gravity feed connection, and a charging pump to the Reactor Coolant System, and

b. The flow path from the refueling water tank via a charging pump to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 5% Δk/k at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

### SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperature of the heat traced portion of the flow path from the boric acid makeup tanks is above the temperature limit line shown on Figure 3.1-1.

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. At least once per 18 months during shutdown by verifying that each actuated valve in the flow path actuates to its correct position on a SIAS test signal.

## CHARGING PUMP - SHUTDOWN

### LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one charging pump in the boron injection flow path required OPERABLE pursuant to Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency bus.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no charging pump OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one of the required pumps is restored to OPERABLE status.

### SURVEILLANCE REQUIREMENTS

4.1.2.3 No additional Surveillance Requirements other than those required by Specification 4.0.5.

## CHARGING PUMPS - OPERATING

### LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 5% $\Delta k/k$ at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

### SURVEILLANCE REQUIREMENTS

4.1.2.4 No additional Surveillance Requirements other than those required by Specification 4.0.5.

## BORIC ACID MAKEUP PUMPS - SHUTDOWN

### LIMITING CONDITION FOR OPERATION

3.1.2.5 At least one boric acid makeup pump shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if only the flow path through the boric acid makeup pump in Specification 3.1.2.1a above, is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no boric acid makeup pump OPERABLE as required to complete the flow path of Specification 3.1.2.1a, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one boric acid makeup pump is restored to OPERABLE status.

### SURVEILLANCE REQUIREMENTS

4.1.2.5 No additional Surveillance Requirements other than those required by Specification 4.0.5.

## BORIC ACID MAKEUP PUMPS - OPERATING

### LIMITING CONDITION FOR OPERATION

3.1.2.6 At least the boric acid makeup pump(s) in the boron injection flow path(s) required OPERABLE pursuant to Specification 3.1.2.2a shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if the flow path through the boric acid makeup pump(s) in Specification 3.1.2.2a is OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one boric acid makeup pump required for the boron injection flow path(s) pursuant to Specification 3.1.2.2a inoperable, restore the boric acid makeup pump to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 5% $\Delta k/k$ at 200°F; restore the above required boric acid pump(s) to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

### SURVEILLANCE REQUIREMENTS

4.1.2.6 No additional Surveillance Requirements other than those required by Specification 4.0.5.

## BORATED WATER SOURCES - OPERATING

### LIMITING CONDITION FOR OPERATION

3.1.2.8 Each of the following borated water sources shall be OPERABLE:

a. At least one boric acid makeup tank and one associated heat tracing circuit per tank with the contents of the tank in accordance with Figure 3.1-1, and

b. The refueling water tank with:

1. A contained borated water volume of between 464,900 and 500,500 gallons (equivalent to an indicated tank level of between 91.7% and 100%, respectively),

2. Between 1731 and 2250 ppm of boron,

3. A minimum solution temperature of 40°F, and

4. A maximum solution temperature of 100°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the above required boric acid makeup tank inoperable, restore the make up tank to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 5% $\Delta k/k$ at 200°F; restore the above required boric acid makeup tank to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.1.2.8 Each of the above required borated water sources shall be demonstrated OPERABLE:

SURVEILLANCE REQUIREMENTS (Continued)

a. At least one per 7 days by:

   1. Verifying the boron concentration in each water source,

   2. Verifying the contained borated water volume in each water source, and

   3. Verifying the boric acid makeup tank solution temperature.

b. At least once per 24 hours by verifying the RWT temperature.

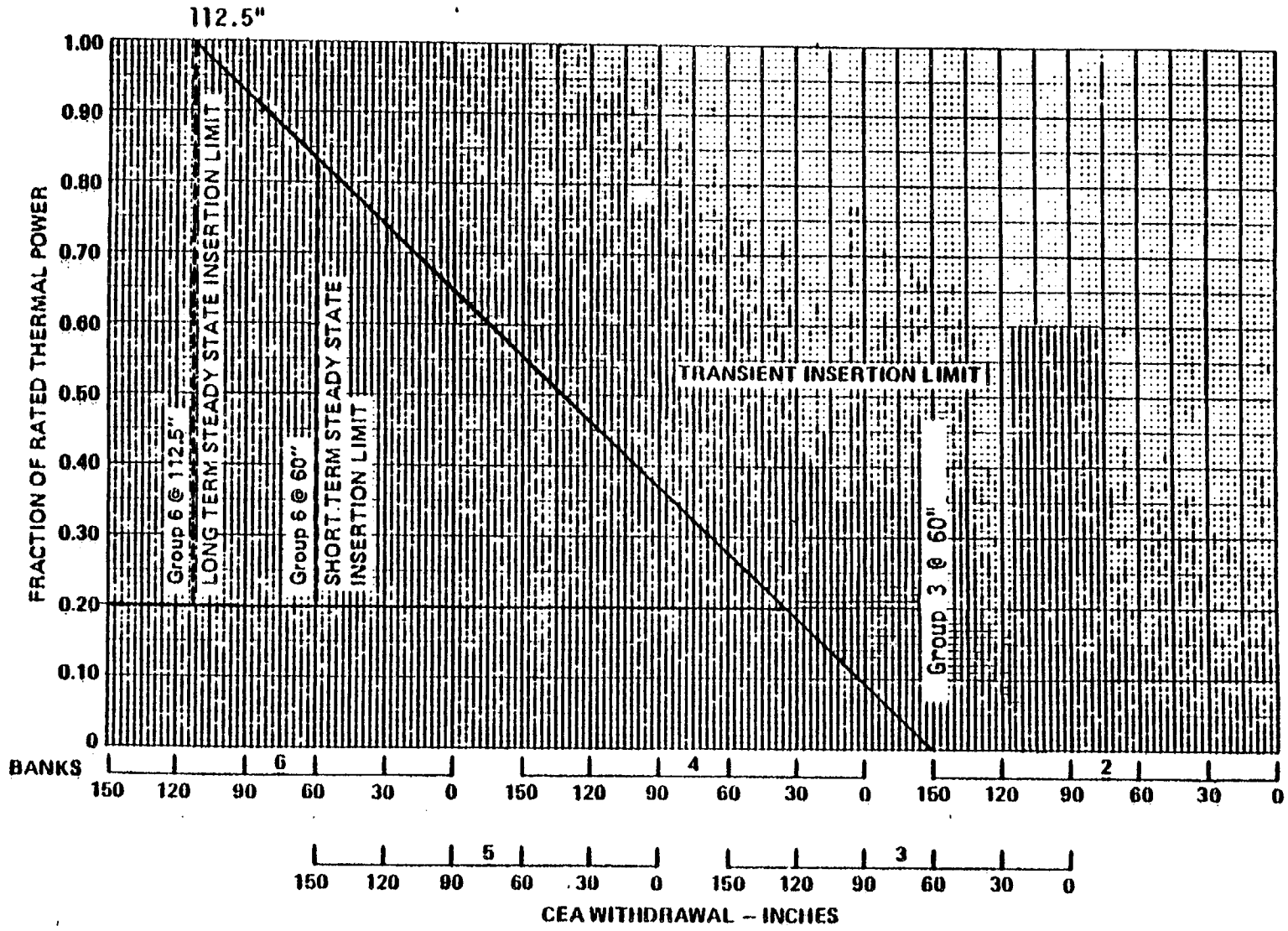


FIGURE 3.1-2
CEA Insertion Limits vs THERMAL POWER

## 3/4.2 POWER DISTRIBUTION LIMITS

### LINEAR HEAT RATE

### LIMITING CONDITION FOR OPERATION

3.2.1 The linear heat rate margin shall be maintained by operating within the region of acceptable operation of Figures 3.2-1 or 3.2-2 as applicable.

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

ACTION:

With the linear heat rate exceeding its limits, as indicated by either (1) the COLSS calculated core power exceeding the COLSS calculated core power operating limit based on kw/ft; or (2) when the COLSS is not being used, any OPERABLE Local Power Density channel exceeding the linear heat rate limit, 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 all OPERABLE Local Power Density channels, 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 kw/ft.

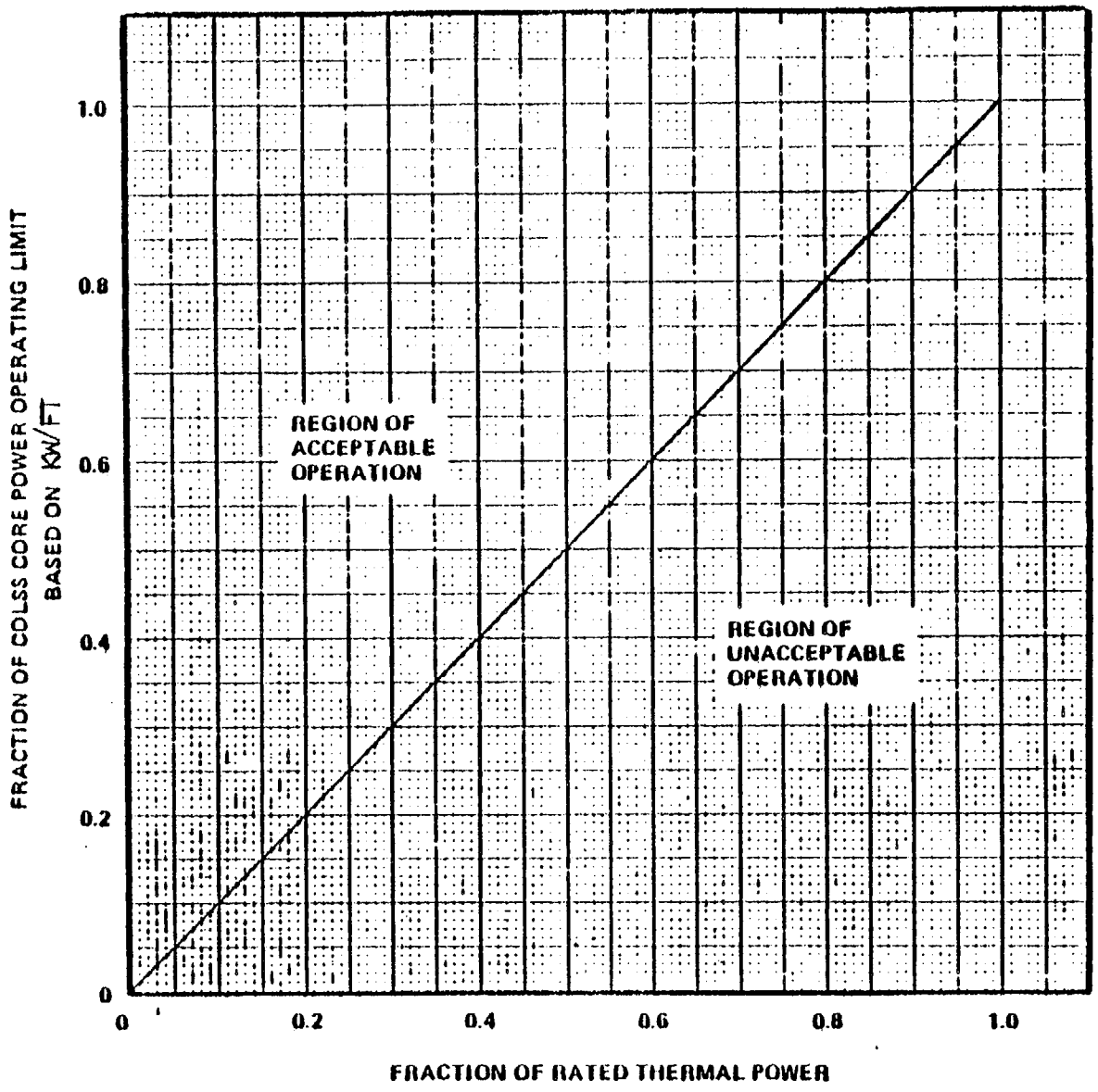


Figure 3.2- 1

KW/FT Margin Operating Limit Based on COLSS

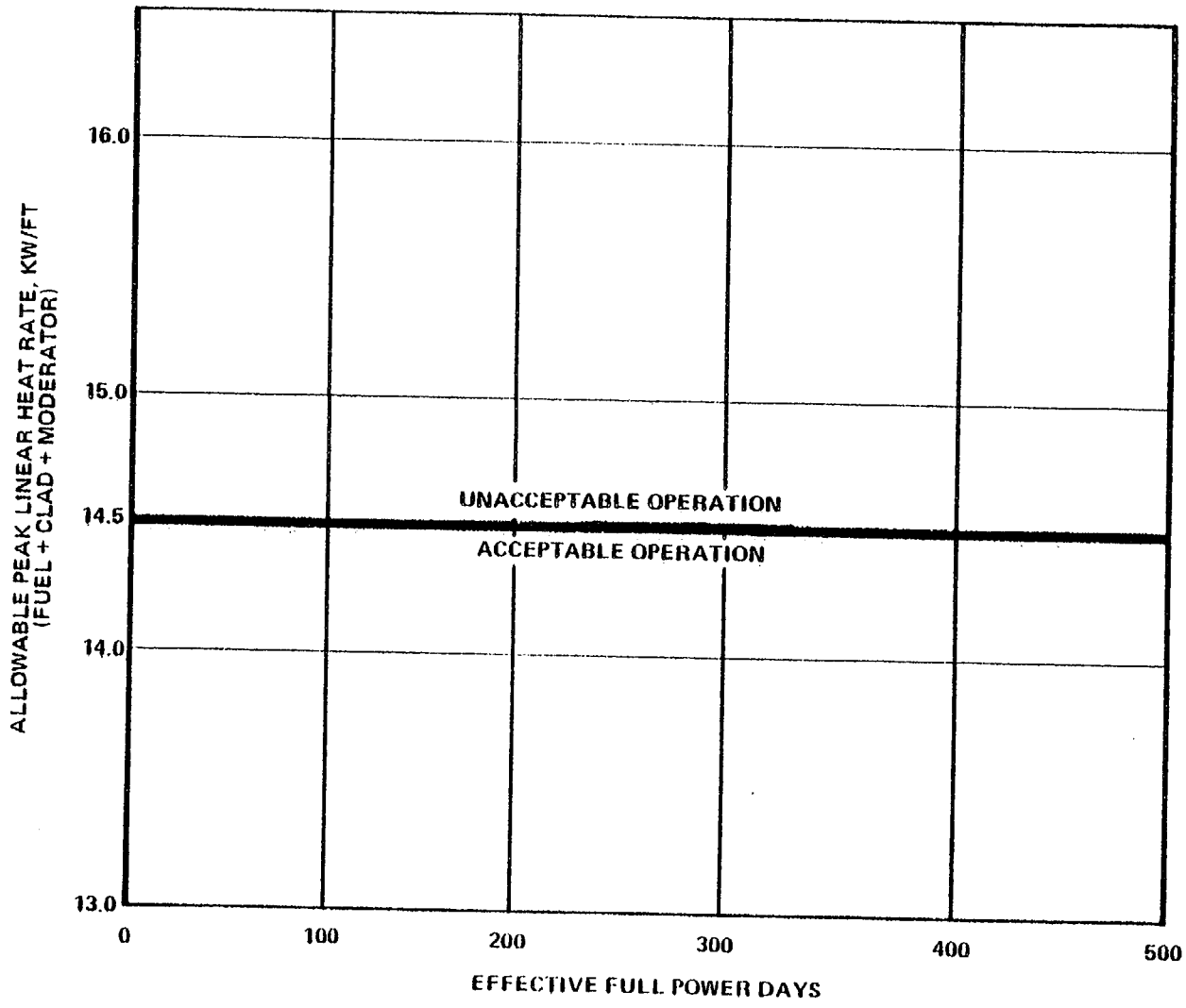


Figure 3.2-2 Allowable Peak Linear Heat Rate vs Burnup
(COLSS out of service)

## 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.

## AZIMUTHAL POWER TILT - $T_q$

### LIMITING CONDITION FOR OPERATION

3.2.3 The AZIMUTHAL POWER TILT ($T_q$) shall be less than or equal to the AZIMUTHAL POWER TILT Allowance used in the Core Protection Calculators (CPCs).

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

ACTION:

a. With the measured AZIMUTHAL POWER TILT determined to exceed the AZIMUTHAL POWER TILT Allowance used in the CPCs but $\leq$ 0.10, within two hours either correct the power tilt or adjust the AZIMUTHAL POWER TILT Allowance used in the CPCs to greater than or equal to the measured value.

b. With the measured AZIMUTHAL POWER TILT determined to exceed 0.10:

1. Due to misalignment of either a part length or full length CEA, within 30 minutes verify that the Core Operating Limit Supervisory System (COLSS) (when COLSS is being used to monitor the core power distribution per Specifications 4.2.1 and 4.2.4) is detecting the CEA misalignment.

2. Verify that the AZIMUTHAL POWER TILT is within its limit within 2 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Linear Power Level - High trip setpoints to $\leq$ 55% of RATED THERMAL POWER within the next 4 hours.

3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the AZIMUTHAL POWER TILT is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

*See Special Test Exception 3.10.2.

## SURVEILLANCE REQUIREMENTS

4.2.3 The AZIMUTHAL POWER TILT shall be determined to be within the limit above 20% of RATED THERMAL POWER by:

a. Continuously monitoring the tilt with COLSS when the COLSS is OPERABLE.

b. Calculating the tilt at least once per 12 hours when the COLSS is inoperable.

c. Verifying at least once per 31 days, that the COLSS Azimuthal Tilt Alarm is actuated at an AZIMUTHAL POWER TILT greater than the AZIMUTHAL POWER TILT Allowance used in the CPCs.

d. Using the incore detectors at least once per 31 days to independently confirm the validity of the COLSS calculated AZIMUTHAL POWER TILT.

## DNBR MARGIN

### LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

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

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

### SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

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

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

SURVEILLANCE REQUIREMENTS (Continued)

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

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

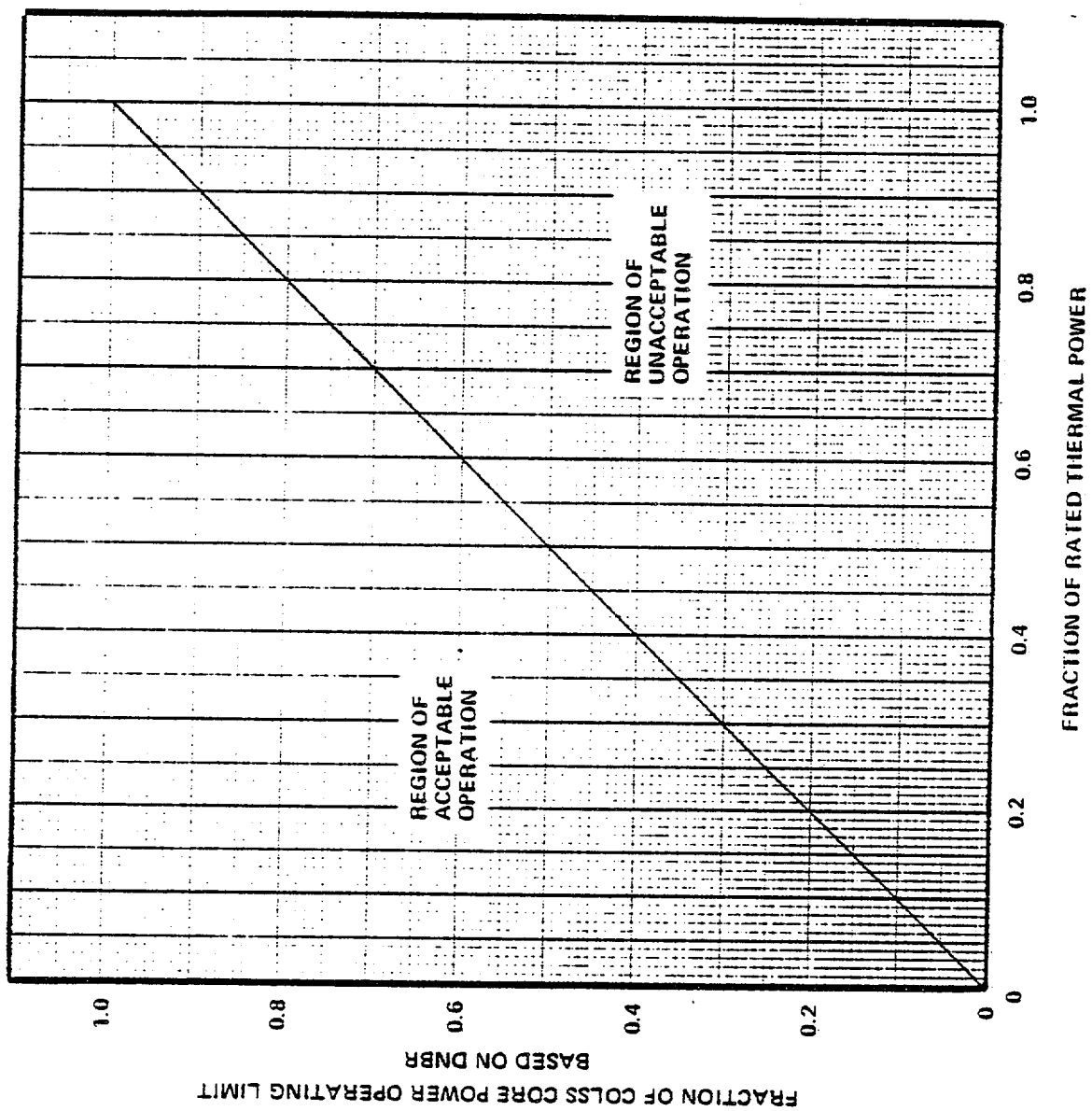


Figure 3.2-3

DNBR Margin Operating Limit Based on COLSS

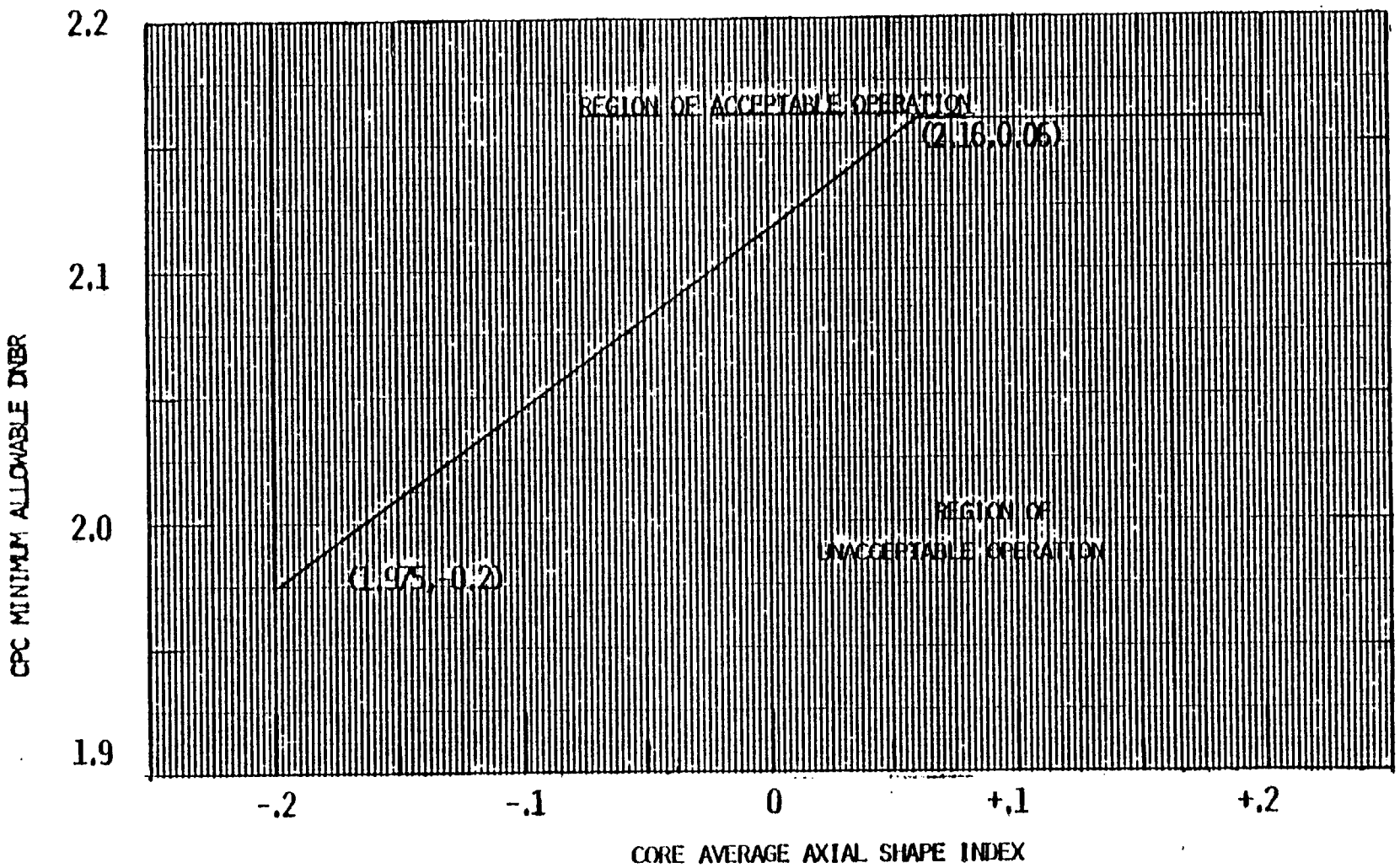


FIGURE 3.2-4

DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS

(COLSS OUT OF SERVICE)

## RCS FLOW RATE

### LIMITING CONDITION FOR OPERATION

3.2.5 The actual Reactor Coolant System total flow rate shall be greater than or equal to $120.4 \times 10^6$ lbm/hr.

APPLICABILITY: MODE 1.

ACTION:

With the actual Reactor Coolant System total flow rate determined to be less than the above limit, reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

### SURVEILLANCE REQUIREMENTS

4.2.5 The actual Reactor Coolant System total flow rate shall be determined to be within its limit at least once per 12 hours.

## REACTOR COOLANT COLD LEG TEMPERATURE

### LIMITING CONDITION FOR OPERATION

3.2.6 The Reactor Coolant Cold Leg Temperature (Tc) shall be maintained between 542°F and 554.7°F.

APPLICABILITY: MODE 1 above 30% of RATED THERMAL POWER.

ACTION:

With the Reactor Coolant Cold Leg Temperature exceeding its limits, restore the temperature to within its limit within 2 hours or reduce THERMAL POWER to less than 30% of RATED THERMAL POWER within the next 4 hours.

### SURVEILLANCE REQUIREMENTS

4.2.6 The Reactor Coolant Cold Leg Temperature shall be determined to be within its limit at least once per 12 hours.

## POWER DISTRIBUTION LIMITS

### AXIAL SHAPE INDEX

### LIMITING CONDITION FOR OPERATION

3.2.7 The core average AXIAL SHAPE INDEX (ASI) shall be maintained within the following limits:

a. COLSS OPERABLE
$-0.28 \leq ASI \leq +0.28$

b. COLSS OUT OF SERVICE (CPC)
$-0.20 \leq ASI \leq +0.20$

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

ACTION:

With the core average AXIAL SHAPE INDEX (ASI) exceeding its limit, restore the ASI to within its limit within 2 hours or reduce THERMAL POWER to less than 20% of RATED THERMAL POWER within the next 4 hours.

### SURVEILLANCE REQUIREMENTS

4.2.6 The core average AXIAL SHAPE INDEX shall be determined to be within its limits at least once per 12 hours using the COLSS or any OPERABLE Core Protection Calculator channel.

*See Special Test Exception 3.10.2.

## PRESSURIZER PRESSURE

### LIMITING CONDITION FOR OPERATION

3.2.8 The average pressurizer pressure shall be maintained between 2225 psia and 2275 psia.

APPLICABILITY: MODE 1

ACTION:

With the average pressurizer pressure exceeding its limits, restore the temperature to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

### SURVEILLANCE REQUIREMENTS

4.2.6 The average pressurizer pressure shall be determined to be within its limit at least once per 12 hours.

SURVEILLANCE REQUIREMENTS (Continued)

2. With 120 volts AC (60 Hz) applied for at least 30 seconds across the input, the reading on the output does not exceed 8 volts DC.

b. For the optical isolators: Verify that the input to output insulation resistance is greater than 10 megohms when tested using a megohmmeter on the 500 volt DC range.

4.3.1.1.5 The Core Protection Calculator System shall be determined OPERABLE at least once per 12 hours by verifying that less than three auto restarts have occurred on each calculator during the past 12 hours.

4.3.1.1.6 The Core Protection Calculator System shall be subjected to a CHANNEL FUNCTIONAL TEST to verify OPERABILITY within 12 hours of receipt of a valid High CPC Room Temperature alarm.

TABLE 3.3-1

REACTOR PROTECTIVE INSTRUMENTATION

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| 1. Manual Reactor Trip | 2 sets of 2 | 1 set of 2 | 2 sets of 2 | 1, 2 and * | 1 |
| 2. Linear Power Level - High | 4 | 2 | 3 | 1, 2 | 2# |
| 3. Logarithmic Power Level-High | | | | | |
| a. Startup and Operating | 4 | 2(a)(d) | 3 | 2 and * | 2# |
| b. Shutdown | 4 | 0 | 2 | 3, 4, 5 | 3 |
| 4. Pressurizer Pressure - High | 4 | 2 | 3 | 1, 2 | 2# |
| 5. Pressurizer Pressure - Low | 4 | 2(b) | 3 | 1, 2 and * | 2# |
| 6. Containment Pressure - High | 4 | 2 | 3 | 1, 2 | 2# |
| 7. Steam Generator Pressure - Low | 4/SG | 2/SG | 3/SG | 1, 2 and * | 2# |
| 8. Steam Generator Level - Low | 4/SG | 2/SG | 3/SG | 1, 2 | 2# |
| 9. Local Power Density - High | 4 | 2(c)(d) | 3 | 1, 2 | 2# |

TABLE 3.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| 10. DNBR - Low | 4 | 2(c)(d) | 3 | 1, 2 | 2# |
| 11. Steam Generator Level - High | 4/SG | 2/SG | 3/SG | 1, 2 | 2# |
| 12. Reactor Protection System Logic | 4 | 2 | 4 | 1, 2 and * | 4 |
| 13. Reactor Trip Breakers | 4(f) | 2 | 4 | 1, 2 and * | 4 |
| 14. Core Protection Calculators | 4 | 2(c)(d) | 3 | 1, 2 | 2# and 6 |
| 15. CEA Calculators | 2 | 1 | 2(e) | 1, 2 | 5# and 6 |

## TABLE 3.3-1 (Continued)

## TABLE NOTATION

*With the protective system trip breakers in the closed position and the CEA drive system capable of CEA withdrawal.

#The provisions of Specification 3.0.4 are not applicable.

(a) 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.

(b) Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is $\geq$ 500 psia.

(c) 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. During testing pursuant to Special Test Exception 3.10.3, trip may be manually bypassed below 1% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 1% of RATED THERMAL POWER.

(d) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(e) See Special Test Exception 3.10.2.

(f) Each channel shall be comprised of two trip breakers; actual trip logic shall be one-out-of-two taken twice.

## ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours and/or open the protective system trip breakers.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

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

1. Within 1 hour the margins required by Specifications 3.2.1 and 3.2.4 are increased and maintained at a value equivalent to $\geq$ 11% of RATED THERMAL POWER.

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 the inoperable status.

c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Off" mode except during CEA group 6 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 during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 6 as permitted by 1. 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 its 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 4.3-1

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | CHANNEL FUNCTIONAL TEST | MODES IN WHICH SURVEILLANCE REQUIRED |
|---|---|---|---|---|
| 1. Manual Reactor Trip | N.A. | N.A. | S/U(1) | N.A. |
| 2. Linear Power Level - High | S | D(2,4),M(3,4), Q(4) | M | 1, 2 |
| 3. Logarithmic Power Level - High | S | R(4) | M and S/U(1) | 1, 2, 3, 4, 5 and * |
| 4. Pressurizer Pressure - High | S | R | M | 1, 2 |
| 5. Pressurizer Pressure - Low | S | R | M | 1, 2 and * |
| 6. Containment Pressure - High | S | R | M | 1, 2 |
| 7. Steam Generator Pressure - Low | S | R | M | 1, 2 and * |
| 8. Steam Generator Level - Low | S | R | M | 1, 2 |
| 9. Local Power Density - High | S | D(2,4), R(4,5) | M, R(6) | 1, 2 |

TABLE 4.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | CHANNEL FUNCTIONAL TEST | MODES IN WHICH SURVEILLANCE REQUIRED |
|---|---|---|---|---|
| 10. DNBR - Low | S | S(7), D(2,4), M(8), R(4,5) | M, R(6) | 1, 2 |
| 11. Steam Generator Level - High | S | R | M | 1, 2 |
| 12. Reactor Protection System Logic | N.A. | N.A. | M | 1, 2 and * |
| 13. Reactor Trip Breakers | N.A. | N.A. | M | 1, 2 and * |
| 14. Core Protection Calculators | S, W(9) | D(2,4),R(4,5) | M,R(6) | 1, 2 |
| 15. CEA Calculators | S | R | M,R(6) | 1, 2 |

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

* - With reactor trip breakers in the closed position and the CEA drive system capable of CEA withdrawal.

(1) - If not performed in previous 7 days.

(2) - Heat balance only (CHANNEL FUNCTIONAL TEST not included), above 15% of RATED THERMAL POWER; adjust the Linear Power Level signals and the CPC addressable constant multipliers to make the CPC $\Delta T$ power and CPC nuclear power calculations agree with the calorimetric calculation if absolute difference is > 2%. During PHYSICS TESTS, these daily calibrations may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(3) - Above 15% of RATED THERMAL POWER, verify that the linear power subchannel gains of the excore detectors are consistent with the values used to establish the shape annealing matrix elements in the Core Protection Calculators.

(4) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) - After each fuel loading and prior to exceeding 70% of RATED THERMAL POWER, the incore detectors shall be used to determine the shape annealing matrix elements and the Core Protection Calculators shall use these elements.

(6) - This CHANNEL FUNCTIONAL TEST shall include the injection of simulated process signals into the channel as close to the sensors as practicable to verify OPERABILITY including alarm and/or trip functions.

(7) - Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by either using the reactor coolant pump differential pressure instrumentation (conservatively compensated for measurement uncertainties) or by calorimetric calculations (conservatively compensated for measurement uncertainties) and if necessary, adjust the CPC addressable constant flow coefficients such that each CPC indicated flow is less than or equal to the actual flow rate. The flow measurement uncertainty may be included in the BERRI term in the CPC and is equal to or greater than 4%.

(8) - Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by calorimetric calculations (conservatively compensated for measurement uncertainties).

(9) - The correct values of addressable constants (See Table 2.2-2) shall be verified to be installed in each OPERABLE CPC.

## 3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

### LIMITING CONDITION FOR OPERATION

3.3.2.1 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels and bypasses shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.3-3 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the Trip Setpoint value.

b. With an ESFAS instrumentation channel inoperable, take the ACTION shown in Table 3.3-3.

### SURVEILLANCE REQUIREMENTS

4.3.2.1.1 Each ESFAS instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-2.

4.3.2.1.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.2.1.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3.

## TABLE 3.3-3 (Continued)

### TABLE NOTATION

ACTION 10 - With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is demonstrated within 1 hour; one additional channel may be bypassed or placed in the tripped condition for up to 2 hours for surveillance testing per Specification 4.3.2.1.1.

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

| FUNCTIONAL UNIT | TRIP SETPOINT | ALLOWABLE VALUES |
|---|---|---|
| 1. SAFETY INJECTION (SIAS) | | |
| a. Manual (Trip Buttons) | Not Applicable | Not Applicable |
| b. Containment Pressure - High | ≤ 18.4 psia | ≤ 19.024 psia |
| c. Pressurizer Pressure - Low | ≥ 1766 psia (1) | ≥ 1712.757 psia (1) |
| 2. CONTAINMENT SPRAY (CSAS) | | |
| a. Manual (Trip Buttons) | Not Applicable | Not Applicable |
| b. Containment Pressure -- High-High | ≤ 23.3 psia | ≤ 23.624 psia |
| 3. CONTAINMENT ISOLATION (CIAS) | | |
| a. Manual (Trip Buttons) | Not Applicable | Not Applicable |
| b. Containment Pressure - High | ≤ 18.4 psia | ≤ 19.024 psia |

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

| FUNCTIONAL UNIT | TRIP VALUE | ALLOWABLE VALUES |
|---|---|---|
| 4. MAIN STEAM AND FEEDWATER ISOLATION (MSIS) | | |
| a. Manual (Trip Buttons) | Not Applicable | Not Applicable |
| b. Steam Generator Pressure - Low | ≥ 751 psia (2) | ≥ 729.613 psia (2) |
| 5. CONTAINMENT COOLING (CCAS) | | |
| a. Manual (Trip Buttons) | Not Applicable | Not Applicable |
| b. Containment Pressure - High | ≤ 18.4 psia | ≤ 19.024 psia |
| c. Pressurizer Pressure - Low | ≥ 1766 psia (1) | ≥ 1712.757 psia (1) |
| 6. RECIRCULATION (RAS) | | |
| a. Manual (Trip Buttons) | Not Applicable | Not Applicable |
| b. Refueling Water Tank - Low | 54,400 ± 2,370 gallons (equivalent to 6.0 ± 0.5% indicated level) | between 51,050 and 58,600 gallons (equivalent to between 5.111% and 6.889% indicated level) |
| 7. LOSS OF POWER | | |
| a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage) | 3120 volts (4) | 3120 volts (4) |
| b. 460 volt Emergency Bus Undervoltage (Degraded Voltage) | 423 ± 2.0 volts with an 8.0 ± 0.5 second time delay | 423 ± 4.0 volts with an 8.0 ± 0.8 second time delay |

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

| FUNCTIONAL UNIT | TRIP VALUE | ALLOWABLE VALUES |
|---|---|---|
| 8. EMERGENCY FEEDWATER (EFAS) | | |
| a. Manual (trip Buttons) | Not Applicable | Not Applicable |
| b. Steam Generator (A&B) Level-Low | ≥ 46.7% (3) | ≥ 45.811% (3) |
| c. Steam Generator ΔP-High (SG-A > SG-B) | ≤ 90 psi | ≤ 99.344 psi |
| d. Steam Generator ΔP-High (SG-B > SG-A) | ≤ 90 psi | ≤ 99.344 psi |
| e. Steam Generator (A&B) Pressure - Low | ≥ 751 psia (2) | ≥ 729.613 psia (2) |

(1) Value may be decreased manually, to a minimum of ≥ 100 psia, during a planned reduction in pressurizer pressure, provided the margin between the pressurizer pressure and this value is maintained at ≤ 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 ≥ 500 psia.

(2) 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 ≤ 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

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

(4) Inverse time relay set value, not a trip value. The zero voltage trip will occur in 0.75 ± 0.075 seconds.

## 3/4.4 REACTOR COOLANT SYSTEM

### REACTOR COOLANT LOOPS

### LIMITING CONDITION FOR OPERATION

3.4.1 Both reactor coolant loops and both reactor coolant pumps in each loop shall be in operation.

APPLICABILITY: As noted below, but excluding MODE 6*.

ACTION:

MODES 1 and 2:

FOUR PUMP OPERATION ***

With less than four reactor coolant pumps in operation be in at least HOT STANDBY within one hour.

PART LOOP OPERATION ***

a. With one reactor coolant pump not in operation, STARTUP and/or continued POWER OPERATION may proceed provided THERMAL POWER is restricted to ≤ **% of RATED THERMAL POWER and the setpoint for the Linear Power Level - High trip has been reduced to the value specified in Specification 2.2.1 for operation with three reactor coolant pumps operating.

b. With two reactor coolant pumps in opposite loops not in operation, STARTUP and/or continued POWER OPERATION may proceed provided THERMAL POWER is restricted to ≤ **% of RATED THERMAL POWER and the setpoint for the Linear Power Level - High trip has been reduced to the value specified in Specification 2.2.1 for operation with two reactor coolant pumps operating in opposite loops.

c. With two reactor coolant pumps in the same loop not in operation, STARTUP and/or continued POWER OPERATION may proceed provided the water level in both steam generators is maintained above the Steam Generator Water Level - Low trip setpoint, the THERMAL POWER is restricted to ≤ **% of RATED THERMAL POWER, and the setpoint for the Linear Power Level - High trip has been reduced to the value specified in Specification 2.2.1 for operation with two reactor coolant pumps operating in the same loop.

---

*See Special Test Exception 3.10.3.
**These values left blank pending NRC approval of safety analyses for operation with less than four reactor coolant pumps operating.
***Part loop operation is not allowed in Modes 1 and 2 pending APL submittal and NRC approval of safety analyses.

ACTION: (Continued)

MODE 3:

Operation may proceed provided two reactor coolant loops are in operation with at least one reactor coolant pump in each loop. With less than one reactor coolant pump in each loop in operation have at least one pump in each loop in operation within one hour or be in at least HOT SHUTDOWN within the next 12 hours.

MODES 4 and 5:

Operation may proceed provided at least one reactor coolant loop is in operation with an associated reactor coolant pump or shutdown cooling pump*. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

---

*All reactor coolant pumps and shutdown cooling pumps may be de-energized for up to 1 hour, provided no operations are permitted which could cause dilution of the reactor coolant system boron concentration.

SURVEILLANCE REQUIREMENTS

4.4.1 The Reactor Protective Instrumentation channels specified in the applicable ACTION statement above shall be verified to have had their trip setpoints changed to the values specified in Specification 2.2.1 for the applicable number of reactor coolant pumps operating either:

a. Within 4 hours after switching to a different pump combination if switch is made while operating, or

b. Prior to reactor criticality if switch is made while shutdown.

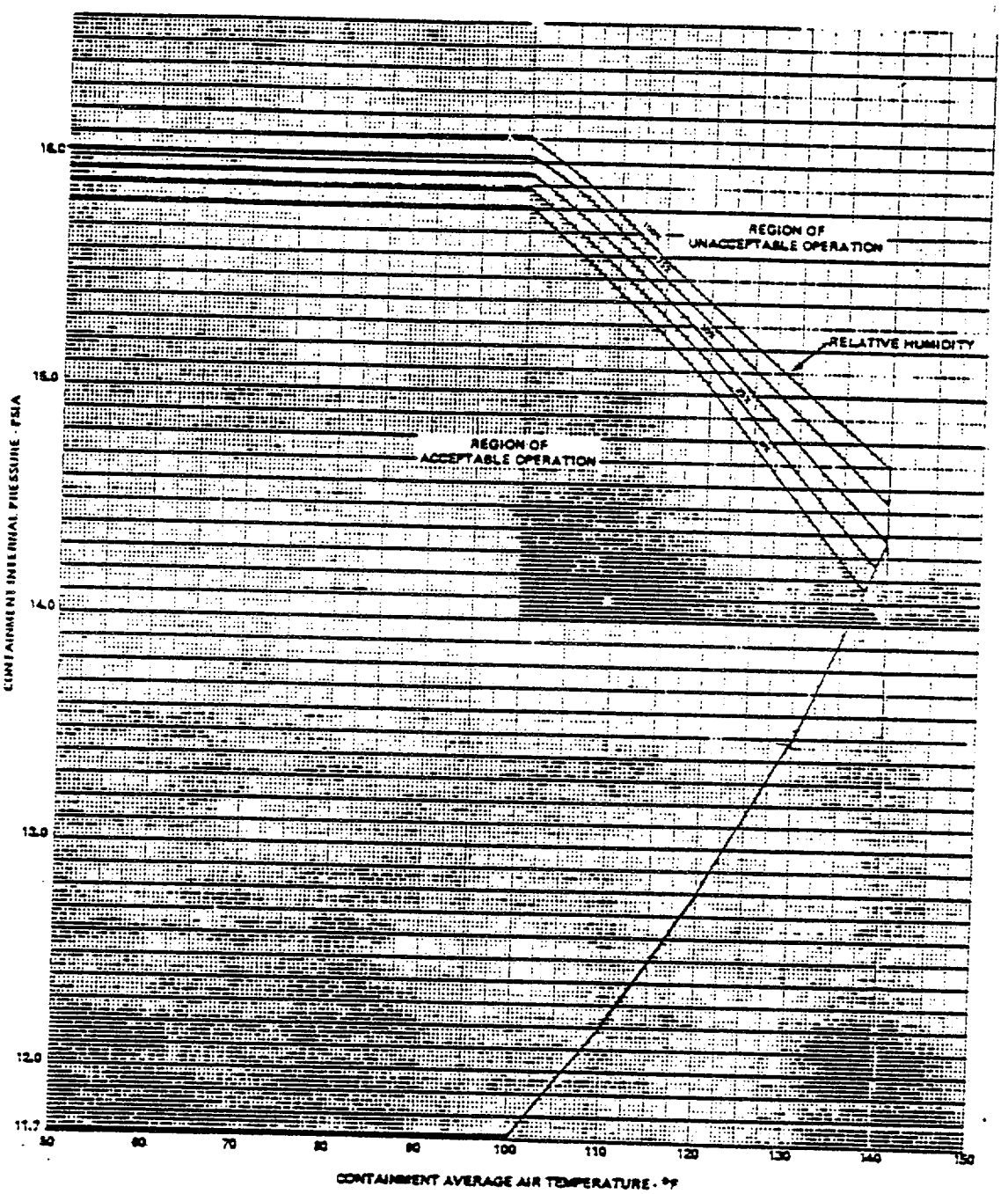


Figure 3.6-1 Containment Internal Pressure vs. Containment Average Air Temperature

## CONTAINMENT STRUCTURAL INTEGRITY

### LIMITING CONDITION FOR OPERATION

3.6.1.5 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.5.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the structural integrity of the containment not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.6.1.5.1 Containment Tendons The containment tendons' structural integrity shall be demonstrated at the end of one, three and five years following the initial containment structural integrity test and at five year intervals thereafter. The tendons' structural integrity shall be demonstrated by a visual examination (to the extent practical and without dismantling load bearing components of the anchorage) of a representative sample* of at least 21 tendons (6 dome, 5 vertical, and 10 hoop) and verifying no abnormal degradation. Unless there is evidence of abnormal degradation of the containment tendons during the first three tests of the tendons, the number of tendons examined during subsequent tests may be reduced to a representative sample of at least 9 tendons (3 dome, 3 vertical and 3 hoop).

*For each inspection, the tendons shall be selected on a random but representative basis so that the sample group will change somewhat for each inspection; however, to develop a history of tendon performance and to correlate the observed data, one tendon from each group (dome, vertical, and hoop) may be kept unchanged after the initial selection.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.1.2 Each isolation valve specified in Table 3.6-1 shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by verifying that on a containment isolation test signal, each isolation valve actuates to its isolation position.

4.6.3.1.3 The isolation time of each power operated or automatic valve of Table 3.6-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

TABLE 3.6-1

CONTAINMENT ISOLATION VALVES

| PENETRATION NUMBER | VALVE NUMBER | FUNCTION | ISOLATION TIME (SEC) |
|---|---|---|---|
| A. CONTAINMENT ISOLATION | | | |
| 2P7 | 2CV-5852-2# | "A" S/G Sample Isolation (outside) | ≤ 20 |
| | 2CV-5859-2# | "B" S/G Sample Isolation (outside) | ≤ 20 |
| 2P8 | 2SV-5833-1 | RCS & Pressurizer Sample Isolation (inside) | ≤ 20 |
| | 2SV-5843-2 | RCS & Pressurizer Sample Isolation (outside) | ≤ 20 |
| 2P9 | 2CV-6207-2 | H.P. Nitrogen to SI Tanks (outside) | ≤ 20 |
| 2P14 | 2CV-4821-1 | CVCS L/D Isolation (inside) | ≤ 35 |
| | 2CV-4823-2 | CVCS L/D Isolation (outside) | ≤ 20 |
| 2P18 | 2CV-4846-1 | RCP Seal Return Isolation (inside) | ≤ 25 |
| | 2CV-4847-2 | RCP Seal Return Isolation (outside) | ≤ 20 |
| 2P31 | 2CV-2401-1 | Containment Vent Header (inside) | ≤ 20 |
| | 2CV-2400-2 | Containment Vent Header (outside) | ≤ 20 |
| 2P37 | 2SV-5878-1 | Quench Tank Liquid Sample (inside) | ≤ 20 |
| | 2SV-5871-2 | Quench Tank Liquid Sample (outside) | ≤ 20 |
| | 2SV-5876-2 | SI Tanks Sample Isolation (outside) | ≤ 20 |
| 2P39 | 2CV-4690-2 | Quench Tank Makeup & Demin Water Supply Isolation (outside) | ≤ 20 |
| 2P40 | 2CV-3200-2 | Fire Water Isolation (outside) | ≤ 20 |
| 2P41 | 2CV-6213-2 | L.P. Nitrogen Supply Isolation (outside) | ≤ 20 |
| 2P51 | 2CV-3852-1 | Chilled Water Supply Isolation (outside) | ≤ 20 |
| 2P52 | 2CV-5236-1 | CCW to RCP Coolers Isolation (outside) | ≤ 20 |
| 2P59 | 2CV-3850-2 | Chilled Water Return Isolation (inside) | ≤ 20 |
| | 2CV-3851-1 | Chilled Water Return Isolation (outside) | ≤ 20 |
| 2P60 | 2CV-5254-2 | CCW from RCP Coolers Isolation (inside) | ≤ 20 |
| | 2CV-5255-1 | CCW from RCP Coolers Isolation (inside) | ≤ 20 |

TABLE 3.7-2

MAXIMUM ALLOWABLE LINEAR POWER LEVEL-HIGH TRIP SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING OPERATION WITH ONE STEAM GENERATOR

| Maximum Number of Inoperable Safety Valves on The Operating Steam Generator | Maximum Allowable Linear Power Level-High Trip Setpoint (Percent of RATED THERMAL POWER) |
|---|---|
| 1 | * |
| 2 | * |
| 3 | * |

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

TABLE 3.7-5

STEAM LINE SAFETY VALVES

| | VALVE NUMBER | | LIFT SETTING (± 1%)* | ORIFICE SIZE |
|---|---|---|---|---|
| | Line No. 1 | Line No. 2 | | |
| a. | 2 PSV 1002 | 2 PSV 1052 | 1078 psig | TT (26.0 in.$^2$) |
| b. | 2 PSV 1003 | 2 PSV 1053 | 1105 psig | TT (26.0 in.$^2$) |
| c. | 2 PSV 1004 | 2 PSV 1054 | 1105 psig | TT (26.0 in.$^2$) |
| d. | 2 PSV 1005 | 2 PSV 1055 | 1132 psig | TT (26.0 in.$^2$) |
| e. | 2 PSV 1006 | 2 PSV 1056 | 1132 psig | TT (26.0 in.$^2$) |

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

## REFUELING MACHINE OPERABILITY

### LIMITING CONDITION FOR OPERATION

3.9.6 The refueling machine shall be used for movement of fuel assemblies and shall be OPERABLE with:

a. A minimum capacity of 3750 pounds,

b. An overload cut off limit of ≤ 100 pounds plus the combined weight of one fuel assembly, one part length CEA, and the grapple in the "fuel only" region, and

c. An overload cut off limit of ≤ 100 pounds plus the combined weight of one fuel assembly, one part length CEA, the grapple, and the hoist box in the "fuel plus hoist box" region.

APPLICABILITY: During movement of CEAs or fuel assemblies within the reactor pressure vessel.

ACTION:

With the requirements for refueling machine OPERABILITY not satisfied, suspend its use from operations involving the movement of CEAs and fuel assemblies within the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.9.6 The refueling machine shall be demonstrated OPERABLE within 72 hours prior to the start of movement of fuel assemblies within the reactor pressure vessel by performing a load test of at least 3750 pounds and demonstrating automatic load cut offs when the crane loads exceed 100 pounds plus the applicable loads.

## CRANE TRAVEL - SPENT FUEL POOL BUILDING

### LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2000 pounds shall be prohibited from travel over fuel assemblies in the spent fuel pool.

APPLICABILITY: With fuel assemblies in the spent fuel pool.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.9.7 The crane electrical power disconnect which prevents crane travel over the spent fuel pool shall be verified open under administrative control at least once per 7 days, or the crane travel interlock which prevents crane travel over the spent fuel pool shall be demonstrated OPERABLE within 4 hours prior to each use of the crane for lifting loads in excess of 2000 pounds.

## 3/4.10 SPECIAL TEST EXCEPTIONS

### SHUTDOWN MARGIN

### LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of CEA worth and shutdown margin provided reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from OPERABLE CEA(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any full length CEA not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at ≥ 40 gpm of 1731 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length CEAs inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at ≥ 40 gpm of 1731 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

### SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length CEA required either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

## SPECIAL TEST EXCEPTIONS

## GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

### LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7 and the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is restricted to the test power plateau which shall not exceed 85% of RATED THERMAL POWER, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.2.2 below.

APPLICABILITY: During startup and PHYSICS TESTS.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while any of the above requirements suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or

b. Be in HOT STANDBY wthin 6 hours.

### SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which any of the above requirements are suspended and shall be verified to be within the test power plateau.

4.10.2.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specifications 4.2.1.3 and 3.3.3.2 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which any of the above requirements are suspended.

## 3/4.1.1 BORATION CONTROL

### 3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS $T_{avg}$. The most restrictive condition occurs at EOL, with $T_{avg}$ at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 5.0% $\Delta k/k$ is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With $T_{avg} \leq 200°F$, the reactivity transients resulting from any postulated accident are minimal and a 5% $\Delta k/k$ shutdown margin provides adequate protection.

### 3/4.1.1.3 BORON DILUTION

A minimum flow rate of at least 3000 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 3000 GPM will circulate an equivalent Reactor Coolant System volume of 9,975 cubic feet in approximately 25 minutes. The reactivity change rate associated with boron concentration reductions will therefore be within the capability of operator recognition and control.

### 3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on MTC are provided to ensure that the assumptions used in the accident and transient analysis remain valid through each fuel cycle. The surveillance requirements for measurement of the MTC during each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurances that the coefficient will be maintained within acceptable values throughout each fuel cycle.

## 3/4.1.1.5 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 525°F. This limitation is required to ensure 1) the moderator temperature coefficient is within its analyzed temperature range, 2) the protective instrumentation is within its normal operating range, 3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and 4) the reactor pressure vessel is above its minimum $RT_{NDT}$ temperature.

## 3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid makeup pumps, 5) associated heat tracing systems, and 6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 5.0% $\Delta k/k$ after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires boric acid solution from the boric acid makeup tanks in the allowable concentrations and volumes of Specification 3.1.2.8 or 56,455 gallons of 1731 ppm borated water from the refueling water tank.

With the RCS temperature below 200°F one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

BASES

The boron capability required below 200°F is based upon providing a 5% Δk/k SHUTDOWN MARGIN after xenon decay and cooldown from 200°F to 140°F. This condition requires either 8185 gallons of 1731 ppm borated water from the refueling water tank or boric acid solution from the boric acid makeup tanks in accordance with the requirements of Specification 3.1.2.7.

The contained water volume limits includes allowance for water not available because of discharge line location and other physical characteristics. The 35,250 gallon limit for the refueling water tank is based upon having an indicated level in the tank of at least 2%.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

The limits on contained water volume and boron concentration of the RWT also ensure a pH value of between 8.9 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of CEA misalignments are limited to acceptable levels.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met.

The ACTION statements applicable to a stuck or untrippable CEA, to two or more inoperable CEAs, and to a large misalignment (≥ 19 inches) of two or more CEAs, require a prompt shutdown of the reactor since any of these conditions may be indicative of a possible loss of mechanical functional capability of the CEAs and in the event of a stuck or untrippable CEA, the loss of SHUTDOWN MARGIN.

For small misalignments (< 19 inches) of the CEAs, there is 1) a small effect on the time dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints, 2) a small effect on the available SHUTDOWN MARGIN, and 3) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the ACTION

statement associated with small misalignments of CEAs permits a one hour time interval during which attempts may be made to restore the CEA to within its alignment requirements. The one hour time limit is sufficient to (1) identify causes of a misaligned CEA, (2) take appropriate corrective action to realign the CEAs and (3) minimize the effects of xenon redistribution.

The CPCs provide protection to the core in the event of a large misalignment ($\geq$ 19 inches) of a CEA by applying appropriate penalty factors to the calculation to account for the misaligned CEA. However, this misalignment would cause distortion of the core power distribution. This distribution may, in turn, have a significant effect on 1) the available SHUTDOWN MARGIN, 2) the time dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints, and 3) the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with the large misalignment of a CEA requires a prompt realignment of the misaligned CEA.

The ACTION statements applicable to misaligned or inoperable CEAs include requirements to align the OPERABLE CEAs in a given group with the inoperable CEA. Conformance with these alignment requirements brings the core, within a short period of time, to a configuration consistent with that assumed in generating LCO and LSSS setpoints. However, extended operation with CEAs significantly inserted in the core may lead to perturbations in 1) local burnup, 2) peaking factors and 3) available SHUTDOWN MARGIN which are more adverse than the conditions assumed to exist in the safety analyses and LCO and LSSS setpoints determination. Therefore, time limits have been imposed on operation with inoperable CEAs to preclude such adverse conditions from developing.

Operability of at least two CEA position indicator channels is required to determine CEA positions and thereby ensure compliance with the CEA alignment and insertion limits. The CEA "Full In" and "Full Out" limits provide an additional independent means for determining the CEA positions when the CEAs are at either their fully inserted or fully withdrawn positions. Therefore, the ACTION statements applicable to inoperable CEA position indicators permit continued operations when the positions of CEAs with inoperable position indicators can be verified by the "Full In" or "Full Out" limits.

## 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 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. Reactor operation at or below this calculated power level assures that the limits of Figure 3.1-1 are not exceeded.

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 flux peaking augmentation and 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 3.2-2 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

$\Theta$ is the azimuthal core location

$\Theta_o$ 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, of which the loss of flow transient is the most limiting. Operation of the core with a DNBR at or above this limit provides assurance that an acceptable minimum DNBR will be maintained in the event of a loss of flow transient.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the DNBR channels in the Core Protection Calculators (CPCs), provide adequate monitoring of the core power distribution and are capable of verifying that the DNBR does not violate its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core operating limit corresponding to the allowable minimum DNBR. Reactor operation at or below this calculated power level assures that the limits of Figure 3.2-3 are not violated. The COLSS calculation of core power operating limit based on DNBR includes appropriate uncertainty and penalty factors necessary to provide a 95/95 confidence level that the core power at which a DNBR of less than 1.24 could occur, as calculated by COLSS, is less than or 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.

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.

## 3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR above 1.24 during all normal operations and anticipated transients. STARTUP and POWER OPERATION may be initiated and may proceed with one or two reactor coolant pumps not in operation after the setpoints for the Power Level-High, Reactor Coolant Flow-Low, and Thermal Margin/Low Pressure trips have been reduced to their specified values. Reducing these trip setpoints ensures that the DNBR will be maintained above 1.24 during three pump operation and that during two pump operation the core void fraction will be limited to ensure parallel channel flow stability within the core and thereby prevent premature DNB.

A single reactor coolant loop with its steam generator filled above the low level trip setpoint provides sufficient heat removal capability for core cooling while in MODES 2 and 3; however, single failure considerations require plant cooldown if component repairs and/or corrective actions cannot be made within the allowable out-of-service time.

## 3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating shutdown cooling loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2750 psia. The combined relief capacity of these valves is sufficient to limit the Reactor Coolant System pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached (i.e., no credit is taken for a direct reactor trip on the loss of turbine) and also assuming no operation of the steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

## 3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves against water relief. The steam bubble functions to relieve RCS pressure during all design transients.

The requirement that 150 KW of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss-of-offsite power condition to maintain natrual circulation at HOT STANDBY.

## 3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 0.5 GPM per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 0.5 GPM per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

BASES

## 3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the accident analyses.

## 3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

## 3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

## 3/4.9.4 CONTAINMENT PENETRATIONS

The requirements on containment penetration closure and OPERABILITY of the containment purge and exhaust system HEPA filters and charcoal adsorbers ensure that a release of radioactive material within containment will be restricted from leakage to the environment or filtered through the HEPA filters and charcoal adsorbers prior to discharge to the atmosphere. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE. Operation of the containment purge and exhaust system HEPA filters and charcoal adsorbers and the resulting iodine removal capacity are consistent with the assumptions of the accident analyses.

## 3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during CORE ALTERATIONS.

## 3/4.9.6 REFUELING MACHINE OPERABILITY

The OPERABILITY requirements for the refueling machine ensure that: 1) the refueling machine will be used for movement of CEAs with fuel assemblies and that it has sufficient load capacity to lift a fuel assembly, and 2) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

## 3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE BUILDING

The restriction on movement of loads in excess of the nominal weight of a fuel assembly, CEA and associated handling tool over other fuel assemblies in the storage pool ensures that in the event this load is dropped (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the accident analyses.

## 3/4.9.8 COOLANT CIRCULATION

The requirement that at least one shutdown cooling loop be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 10,295 ± 400 cubic feet at a nominal $T_{avg}$ of 545°F.

## 5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

## 5.6 FUEL STORAGE

CRITICALITY - SPENT FUEL

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with a nominal 12.8 inch center-to-center distance between fuel assemblies having a maximum enrichment of 4.3 weight percent U-235 placed in the storage racks to ensure a $k_{eff}$ equivalent to $\leq$ 0.95 when flooded with unborated water. The $k_{eff}$ of $\leq$ 0.95 includes a conservative allowance of 1.7% $\Delta k/k$ for uncertainties as described in Section 9.1.2.3 of the FSAR. In addition, fuel in the storage pool shall have a U-235 loading of $\leq$ 47.8 grams of U-235 per axial centimeter of fuel assembly.

CRITICALITY - NEW FUEL

5.6.1.2 The new fuel storage racks are designed and shall be maintained with a nominal 25.0 inch center-to-center distance between new fuel assemblies such that $K_{eff}$ will not exceed 0.98 when fuel having a maximum enrichment of 3.7 weight percent U-235 is in place and aqueous foam moderation is assumed and $K_{eff}$ will not exceed 0.95 when the storage area is flooded with unborated water. The calculated $K_{eff}$ includes a conservative allowance of 1.0% $\Delta k/k$ for uncertainties.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 399' 10 1/2".

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 486 fuel assemblies.

## 5.7 COMPONENT CYCLIC OR TRANSIENT LIMITS

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

| COMPONENT | CYCLIC OR TRANSIENT LIMIT | DESIGN CYCLE OR TRANSIENT |
|---|---|---|
| Reactor Coolant System | 500 system heatup and cooldown cycles at rates ≤ 100°F/hr. | Heatup cycle - $T_{avg}$ from ≤ 200°F to ≥ 545°F; cooldown cycle - $T_{avg}$ from ≥ 545°F to ≤ 200°F. |
| | 500 pressurizer heatup and cooldown cycles at rates ≤ 200°F/hr. | Heatup cycle - Pressurizer temperature from ≤ 200°F to ≥ 653°F; cooldown cycle - Pressurizer temperature from ≥ 653°F to ≤ 200°F. |
| | 10 hydrostatic testing cycles. | RCS pressurized to 3110 psig with RCS temperature ≥ 60°F above the most limiting components' NDTT value. |
| | 200 leak testing cycles. | RCS pressured to 2250 psia with RCS temperature greater than minimum for hydrostatic testing, but less than minimum RCS temperature for criticality. |
| | 400 reactor trip cycles. | Trip from 100% of RATED THERMAL POWER. |
| | 40 turbine trip cycles with delayed reactor trip. | Turbine trip (total load rejection) from 100% of RATED THERMAL POWER followed by resulting reactor trip. |
| | 200 seismic stress cycles. | Subjection to a seismic event equal to one half the design basis earthquake (DBE). |

## 6.7 SAFETY LIMIT VIOLATION

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

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

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

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

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

## 6.8 PROCEDURES

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

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

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

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

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

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

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

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

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

## 6.9 REPORTING REQUIREMENTS

### ROUTINE REPORTS AND REPORTABLE OCCURRENCES

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

### STARTUP REPORT

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

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

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

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

ARKANSAS POWER AND LIGHT COMPANY

ARKANSAS NUCLEAR ONE, UNIT NO. 2

DOCKET NO. 50-368

8106260498

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## 1.0 INTRODUCTION

By application dated February 20, 1981 and March 5, 1981, and supplemental information as listed in the reference section of this report, Arkansas Power & Light Company (AP&L Co. or the licensee) requested an amendment to Facility Operating License No. NPF-6 for the Arkansas Nuclear One-Unit No. 2 plant (ANO-2 or the facility). The amendment request consists of:

- Appendix A (Safety) Technical Specification (TS) changes resulting from the analysis of the Cycle 2 reload fuel and other matters as discussed in this report.

- Proposed changes to the reactor protection system's core protection calculator system computer software to accommodate new methodology for calculating departure from nucleate boiling ratio trip limits.

The associated specific TS changes are described in section 3.0 of this report. In addition this report addresses our evaluation of:

- The completion of the requirements of conditions to the license related to Fuel Performance, Instrument Trip Setpoints Drift Allowance, Overpressure Mitigation System, and the CEA Guide Tube Surveillance Program. These evaluations are presented in section 2.6 of this report.

- The issuance of TS changes for matters not necessarily related to the review of the reload analyses but which may be conveniently addressed in this evaluation. These include TS changes on (1) limiting the containment pressure, temperature and relative humidity so as to control containment differential pressure in the event of an inadvertent actuation of containment spray, (2) the high pressurizer pressure trip setpoint. These evaluations are presented in section 2.7 of this report.

The information provided to support the staff's review of this reload and other issues included in this report are listed in the reference section (7.0) of this report.

## 2.0 DISCUSSION AND EVALUATION

We have reviewed the information provided in support of the ANO-2 Cycle 1 reload to determine whether the design objectives continue to be met and to determine whether the proposed reload changes have resulted in a reduction of previously approved design margins. Our evaluations, as described in the following sections of this report, are complete for the purposes of authorizing Cycle 2 operation at the licensed full power level of 2815 MWt except for certain detailed matters within the thermal hydraulic review. The status of the thermal hydraulic review is discussed as follows.

By letter dated December 1, 1980 (Ref. 2.3-1), AP&LCo submitted new methodology on the statistical combination of uncertainties in the calculation of the minimum DNB ratio, prepared by Combustion Engineering, Inc. (CE), for use in Cycle 2 and future ANO-2 reloads. By letter dated January 9, 1981, AP&L Co. submitted descriptions of revised software for the CPC/CEAC system to implement the CE-1 departure from nucleate boiling ratio correlation for Cycle 2 and future cycles. These reports, in conjunction with other information submitted and in support of the Cycle 2 Reload Report, provide the basis for the Cycle 2 Limiting Conditions for Operation (LCOs).

The staff has determined that insufficient time is available to complete all details of the review of these reports prior to the scheduled attainment of core criticality for Cycle 2 operation. AP&L Co. has been requested to provide additional information to enable the staff to complete its review of the remaining details. The nature of the staff's concerns relates to whether or not sufficient margins have been represented in the core protection calculator system software changes for Cycle 2 to account for the uncertainties associated with the following: (1) the CE-1 DNBR correlation, (2) the CETOP-D code, (3) the CETOP-2 code, and (4) the statistical combination of uncertainties. While the staff's review of all details of these matters has not yet been completed, we have examined these issues in depth, we judge the basic changes to be reasonable, and conclude that the completion of our review will not reveal the application of these changes for ANO-2 Cycle 2 operation to be significantly in error. Since these questions relate primarily to the adequacy of available thermal margins to account for anticipated operational occurrences (AOOs) at full power conditions, we have concluded that it is acceptable for the plant to start up and operate at a reduced power level for a short period pending the completion of our reviews. Operation at a reduced power level will provide additional thermal margins to account for the uncertainties discussed above while we complete our review. The licensee has submitted additional information on the Linear Power Level - High Trip required to limit operation to seventy percent of the licensed full power level of 2815 MWt.

Further details regarding these matters are presented in Section 2.3 of this report. On the basis of the information discussed above, including the licensee's Linear Power Level - High Trip value which will provide additional protection to the plant from AOOs at the reduced power level, we conclude that operation during this interim period at the reduced power level is acceptable. Upon completion of our review of these matters, another SE will be issued.

## 2.1 CYCLE 2 FUEL DESIGN

The ANO-2 Cycle 2 core will be comprised of 177 fuel assemblies of the 16x16 geometry that were manufactured by Combustion Engineering, the original NSSS vendor. The major changes to the core for Cycle 2 are the removal of 60 Batch A fuel assemblies. These assemblies will be replaced by 40 Batch D assemblies and 20 Batch D* assemblies. The Cycle 2 core loading inventory is given in Table 1.

The Cycle 1 fuel management pattern (Refs. 1 and 2) was developed to accommodate an EOC-1 core-average exposure of 12.5 GWd/t, which was the actual exposure achieved (Ref. 3). After the reload, the BOC-2 exposure will be 7.9 GWd/t, and the EOC-2 exposure is predicted to be 19.0 GWd/t. The maximum EOC-2 exposure of any individual assembly will be 25.2 GWd/t.

Two Batch D fuel assemblies will serve as carriers for 42 DOE high-burnup demonstration rods (Ref. 4). Among the test rods are designs such as annular fuel pellets, large-grain-sized pellets, graphite coatings on cladding inner surfaces, and segmented fuel rods. It is anticipated that the performance information to be obtained from these test rods will contribute to establishing the bases upon which future batch-average exposures may be increased to as much as 53 GWd/t.

All other fuel comprising Cycle 2 is of the standard FSAR design except 4 C-E test rods (Ref. 5) of a proprietary design. These rods are contained in a Batch 4 fuel assembly that was previously burned in Cycle 1.

Evaluation of the C-E 16x16 fuel mechanical design is based on engineering analyses, tests, and a substantial amount of in-reactor operating experience with previous 14x14 and 15x15 fuel designs. In addition, the performance of the design is subject to continuing surveillance of operating reactors by C-E and licensees having C-E NSSS plants. These programs continually provide current performance information.

### 2.1.1 THERMAL PERFORMANCE ANALYTICAL METHODS

The C-E fuel performance evaluation model called FATES is presented in the C-E topical report CENPD-139, "Fuel Evaluation Model " (Ref. 6). This model was used to calculate fuel temperature, stored energy, linear thermal output, and augmentation (power spike) factors.

In 1976, after the approval (Ref. 7) of CENPD-139, information was made available to the NRC that lead us to question the validity of fission gas release calculations in the C-E model for fuel pellet burnups greater than 20 GWd/t. Combustion Engineering was informed (Ref. 8) of this concern and provided with a method of correcting fission gas release calculations for burnups greater than 20 GWd/t. Also, the ANO-2 license (Ref. 9) was conditioned to require resolution of this issue prior to the cycle in which a pellet burnup of 20 GWd/t was achieved.

In response to our concern, AP&LCo submitted (Refs. 1 and 2) a Cycle 2 reload analysis in which the NRC correction method has been used (Ref. 3) in all FATES analyses including that for the LOCA. Also, AP&LCo has performed (Ref. 10) a rod internal pressure analysis using the present C-E fuel performance model with the NRC correction for enhanced fission gas release. The results with the NRC correction method show that (a) the ANO-2 fuel will not exceed the LOCA acceptance criteria of 10 CFR 50.46, (b) rod internal pressure will remain below nominal coolant system pressure throughout Cycle 2, and (c) other burnup-dependent analyses have implicitly accommodated enhanced fission gas release.

We, therefore, conclude that enhanced fission gas release has been appropriately considered for ANO-2 Cycle 2 operation.

### 2.1.2 CLADDING CREEP COLLAPSE

Combustion Engineering has written a computer code that calculates time-to-collapse of Zircaloy cladding in a pressurized water reactor environment. This code has been approved by the NRC and is described in the report CENPD-187, "CEPAN Method of Analyzing Creep Collapse of Oval Cladding" (Ref. 11).

For Cycle 2 operation, C-E has performed time-to-cladding-collapse calculations using CEPAN and conservative input values of internal rod pressure, cladding dimensions, cladding temperature, and neutron flux. The results of this analysis showed that the minimum time-to-collapse is in excess of the design batch-average discharge lifetime of the fuel, which will not be exceeded during Cycle 2 operation. The cladding collapse analysis for the DOE demonstration and C-E test rods were included in the analysis discussed above.

We, therefore, conclude that the fuel rod cladding collapse analysis for ANO-2 Cycle 2 operation is acceptable.

2.1.3 FUEL ROD BOWING

Because fuel rod bowing in pressurized water reactors affects neutronic and thermal-hydraulic safety margins, AP&LCo has analyzed the anticipated extent of rod bowing in Cycle 2. In the analysis, AP&LCo has referenced the C-E topical report CENPD-225, "Fuel and Poison Rod Bowing" (Ref. 12).

The staff has not yet approved the CENPD-225 report. Accordingly, it is the staff position that the rod bow compensation currently specified in Technical Specification 4.2.4.4 shall remain applicable for initial Cycle 2 operation. We estimate that the peak bundle average burnup will be 20.2 GWD/t by the end of November 1981 when the rod bow compensation review is expected to be complete. The rod bow compensation required for that burnup is 11.4 percent, compared to the proposed 2%, of the ENBR limit value.

The difference in DNBR limit due to rod bow compensation methodology should by compensated by an equivalent increase in power uncertainty factor BERR 1. Based on the sensitivity study provided in the response to NRC questions 492,66, a relationship between the BERR 1 and DNBR limit is established. Using the most conservative value of -0.6 for the derivative of percentage BERR 1 with respect to the percentage ENBR, we estimate the BERR 1 value should be increased by 5.6% to account for the rod bow compensations.

We anticipate our approval of the topical report CENPD-225 by November 1981. At that time, AP&LCo may amend their Technical Specifications to reflect any reduction in the rod bowing penalty that is possible from the application of the CENPD-225 methodology.

We thus conclude that the effects of fuel rod bowing have been adequately addressed for Cycle 2 operation.

2.1.4 FUEL ASSEMBLY SHOULDER GAP

During irradiation, fuel rods and fuel assembly guide tubes undergo axial growth at different rates. To ensure that an adequate design shoulder gap exists for the fuel assemblies that will comprise the Cycle 2 core, AP&LCo has made a calculation (Ref. 3) on the lead-burnup fuel rod in a Batch B fuel assembly.

The calculation of the minumum shoulder gap in the Batch B fuel assembly was performed with the methods described in the C-E topical report, CENPD-198, "Zircaloy Growth In-Reactor Dimensional Changes in Zircaloy-4 Fuel Assemblies" (Refs. 17, 18, and 19). The calculation was made for axially averaged fast neutron fluence to $4X10^{21}$ neutrons per square centimeter, which corresponds to a maximum assembly exposure of 22.5 GWd/t, as specified in our approval (Ref. 20) on the CENPD-198 methodology.

For calculating differential growth at exposures beyond 22.5 GWd/t, a more conservative method, which is acceptable (Ref. 21), was utilized. The results showed that no interference between fuel rods and the upper end fitting is predicted for Cycle 2 operation. Furthermore, during the current refueling outage, shoulder gap inspection of the Cycle 1 characterized fuel assemblies verified the acceptability of the gap calculation.

Therefore, we conclude that an adequate fuel assembly shoulder gap will be maintained for Cycle 2 operation.

### 2.1.5 CEA AND FUEL ASSEMBLY GUIDE TUBE INTEGRITY

Fretting wear has been observed (for example see Refs. 22, 23, 24, and 25) in irradiated fuel assemblies taken from operating C-E reactors. These observations revealed unexpected degradation of guide tubes that were under control element assemblies. It was concluded that coolant turbulence was responsible for vibration of the normally fully withdrawn control rods and, where these vibrating rods were in contact with the inner surface of the guide tubes, wearing of the guide tube walls took place.

As a remedy, AP&LCo has installed scupper extensions on the upper guide structure flow channels and stainless-steel sleeves in all fuel assembly guide tubes to be used in CEA positions. The 4 Batch A unsleeved test assemblies (Ref. 26) that resided under CEAs in Cylce 1 will not be used in Cycle 2.

Our review of the scupper extensions and the sleeving program has been documented in the ANO-2 safety evaluation (Ref. 26). Our prior safety evaluation concluded that scupper extensions and guide tube sleeves will perform their function of mitigating fretting wear in fuel assemblies. Furthermore, to provide assurance of guide tube and sleeve integrity, the licensee performed an EOC-1 guide tube surveillance program (Ref. 27). Eddy current testing was performed on all of the guide tubes in 5 fuel assemblies from Batch A and 5 fuel assemblies from Batch C. These assemblies were spatially selected from the Cycle 1 core on the basis of where maximum wear might be expected to occur. The results (Ref. 28) indicate that the total amount of wear is negligible and that sleeves remain intact.

We conclude that the sleeved guide tubes in the Cycle 2 fuel assemblies continue to meet their design functions and are therefore acceptable. Based on the reported favorable surveillance results and continued use of guide tube sleeves under all CEAs, we consider the issue of guide tube wear resolved for future cycles of ANO-2.

While the stainless steel sleeves have precluded guide tube wear, they have probably increased the cladding wear that occurs on the control rods themselves. Therefore, during the Cycle 2 outage, eddy current testing with an encircling probe was performed on 8 CEAs. The results (Ref. 28) were consistent with similar measurements on CEAs from C-E NSSS reactors using 14X14 fuel designs after one operating cycle. Since the measured wear is within the limits for continued CEA operation, it is therefore acceptable.

To date, no inspections have revealed CEA cladding wear rates that would indicate a potential for the loss of CEA hermiticity in the near future. It, nevertheless, remains uncertain as to whether wear degradation to CEAs could ultimately reduce the CEA design lifetime. We can, however, conclude that for Cycle 2 operation, fretting wear to CEA cladding will remain at acceptably low levels.

### 2.1.6 PROGRAMMED CEA INSERTION

During ANO-2 Cycle 2 operation, AP&LCo will continue programmed CEA insertion. This program was approved (Ref. 26) and instituted during Cycle 1 operation so that the magnitude of guide tube wear at any one location would be reduced by repositioning fully withdrawn CEAs. (Specifically, the full-out insertion limits for the CEAs are extended 3 inches into the core.)

We believe that this method of apportioning the wear is acceptable, though not necessary, because the guide tube sleeve wear reported (Ref. 28) in ANO-2 is insignificant and the Cycle 2 core will contain no unsleeved assemblies. Because increased axial peaking of about 4% occurs when all of the CEAs are inserted to the 3-inch full-out insertion limit, AP&LCo might want to consider discontinuing this program in future cycles.

### 2.1.7 FUEL FAILURES

In January 1980 AP&LCo determined from primary coolant activity that a limited number of fuel rods had perforated in ANO-2. The failures were detected over a 1 to 3 day period while the plant was in the initial power ascension program. Prior to the occurrence of these failures, the testing program included preconditioning at 80% power, dropped-control-rod testing at 50% power, and then a ramp to 65% power. Following the ramp to 65% power, xenon oscillations were observed coincident with increased coolant activities.

During Cycle 1, the licensee and fuel vendor were unaware of the specific nature of the failures, but had tentatively ascertained that the fuel was operated within the recommended operating restrictions inasmuch as (a) the fuel was preconditioned prior to the occurrence of failures and (b) the rate of power ramp just prior to failure occurrence was substantially less than that allowed by the operating limits.

During the outage all 177 assemblies in the core were sipped and a total of 7 assemblies were found to contain leaking fuel rods (Ref. 29). These leakers were distributed among 2 Batch A assemblies, 3 Batch B assemblies, and 2 Batch C assemblies. Since visual inspection was not successful in locating the failed rods, each of the rods in the leaking Batch B and C fuel assemblies was removed and eddy current tested to identify the failed rods. A total of 14 failed fuel rods were found in these 5 assemblies, which were planned for reuse during Cycle 2. In addition, one poison rod in a Batch C assembly was found to be perforated. Also, 52 additional fuel rods showed questionable eddy current indications.

Consequently, the 14 failed and 52 questionable fuel rods were replaced by 66 fuel rods that were extracted from a sound Batch A fuel assembly. The perforated poison rod was replaced with a solid Zircaloy "dummy" rod. Finally, prior to reinsertion into the core, all 5 reconstituted fuel assemblies were sipped to ensure the absence of leakers.

Our interest in this issue is based on three fundamental concerns. First, that coolant activity levels be kept as low as reasonably achievable and within the Technical Specification limit and safety analysis assumptions. Second, that the cause of the failures be investigated so that such failures can be minimized or eliminated. Third, that NRC receive prompt notification of such failures so that (a) operators of similar plants can be informed and (b) NRC can watch for common trends and generic problems.

In regard to the first concern, the licensee has replaced the failed fuel rods with non-failed fuel rods of lesser enrichment and the failed poison rod with a solid Zircaloy rod. Since the licensee has determined that these substitutions do not violate the Cycle 2 physics analysis, we find these actions to be appropriate.

In regard to the second concern, the licensee has not completed the investigation and, therefore, has not determined the cause of all failures. Preliminary indications are that (a) some of the fuel rod failures were caused by fretting wear from foreign material lodged between lower end fitting flow plates and bottom Inconel grid structures, and (b) the poison rod failure may be due to fretting. The licensee is continuing the investigation and will report the findings in a written report that was scheduled for submittal in 90 days (see further discussion in Section 2.1.9 with regard to this schedule). The second concern is, therefore, being adequately addressed.

In regard to the third concern, AP&LCo has agreed to issue (a) a report describing the present damaged fuel and (b) a letter to NRC during Cycle 2 operation if additional fuel failures are inferred from variations in the equilibrium primary coolant activity level. Consequently, AP&LCo has satisfied the third concern.

### 2.1.8 GENERAL FUEL ASSEMBLY SURVEILLANCE AND GRID STRAP DAMAGE

The fuel surveillance program that was described in Section 4.2.1.1.10 of the FSAR included the visual examination of all the initial core fuel assemblies. Approximately 15 fuel assemblies were to be inspected prior to reactor startup, and the visual examination of the balance of the discharged assemblies was to be performed later.

The licensee reported (Ref. 28) preliminary results of the 15 fuel assemblies that were inspected by TV camera or periscope. There were no abnormalities observed from these assembly inspections. However, in later performing the visual examinations of the remaining 45 discharged Batch A fuel assemblies, AP&LCo observed 5 assemblies having grid strap damage. This information was verbally conveyed to NRC on May 27 and followed up by letter of June 4, 1981 (Ref. 29).

Of the 5 assemblies having grid strap damage, 2 had relatively minor damage that was confined to missing tabs, while each of the other 3 had significant damage that consisted of a missing section of one of the grid perimeter straps on that assembly. The damage apparently occurred during the Cycle 2 outage because (a) the fracture surfaces were shiny and not oxidized like the remainder of the undamaged grid surfaces and (b) fuel rods adjacent to the missing grid strap sections had abrasion markings that corresponded to finger spring locations thus indicating the presence of intact grid straps during Cycle 1 operation.

Because the grid strap damage was not detected until after the core was reloaded, the number of assemblies with damaged grids remaining in the core in unknown, but estimated (assuming a random damage distribution) by C-E to be limited to 16.

Our interest in this issue is based on three concerns. First, that the cause of the grid strap damage be determined and eliminated or the effects be reduced. Second, that any grid strap damage present during the next cycle of operation not result in unacceptable damage such as additional fuel failures. Third, that NRC receive notification of such occurrences in the future.

In regard to the first concern, AP&LCo has not yet determined the exact cause of grid strap tearing. However, as discussed above, AP&LCo is confident that the damage occurred during the Cycle 2 outage fuel shuffle. This conclusion is further supported by AP&LCo's review of refueling experiences and procedures used during the outage. From personnel interviews, it was learned that refueling machine overload/underload trips occurred frequently during withdrawal and replacement of fuel assemblies. Since the overload/underload trip set points were believed to be adequate to prevent fuel damage, routine procedure was followed after such trips occurred. That procedure consisted of reestablishing normal loading by raising or lowering the assembly and then manually shaking the hoist cable. (This procedure had been found successful for similar occurrences in ANO-1.)

The licensee is continuing this investigation to (a) quantify post-trip fuel assembly loading that occurs due to system momentum, (b) determine the effect of cable shaking, and (c) determine the loading required to cause grid damage. The first concern is thus being adequately addressed.

In regard to the second concern, if grid strap damage is present in the Cycle 2 core, there are two topics of interest: (1) the potential for fretting or fatigue damage to fuel and poison rods that might be inadequately supported in the vicinity of damaged grid sections and (2) the potential for problems associated with loose grid pieces, including the possibility of flow blockage with attendant departure from nucleate boiling.

Based on C-E out-of-pile tests on 16X16 fuel bundles, the licensee does not expect rods that are inadequately supported at one grid elevation to fail. We are not familiar with the specific tests to which AP&LCo has referred; however, it is most unlikely that these tests employed simulated grid damage. Consequently, we have no opinion on the applicability of these tests. Nevertheless, C-E has conservatively estimated that there are less than 24 fuel rods which are adjacent to damaged grid sections. Hence, the number of potential failures is limited.

Concerning potential problems due to loose grid pieces, the licensee has postulated that limited fuel rod failure could occur due to fretting at locations where grid pieces might become lodged in the fuel region. We agree that such a failure mechanism is conceivable although this type of fretting wear has not previously been observed and would be local and confined to a few rods at most.

Since (a) the rate of fuel failures due to fatigue or fretting would be slow and detectable by the letdown monitors or periodic primary coolant sampling, (b) the number of rods involved is small, and (c) this hypothetical assessment seems conservative, we conclude that the second concern is satified for Cycle 2 operation.

In regard to the last concern, the licensee has agreed (as discussed above in Section 2.1.7) to notify NRC by letter in the event that fuel failures are detected during Cycle 2 operation. Also, with respect to the general fuel assembly inspections at EOC-2, the licensee has agreed to performing these visual inspections prior to sealing up the reactor vessel (the FSAR commitment reads only "prior to reactor re-startup"). Performing these inspections prior to sealing up the system (such was also the case for this outage) will assure flexibility for inspecting additional fuel assemblies in the event that such is warranted from the observations on the selected 15 (or more) fuel assemblies. We, therefore, conclude that the third concern will be handled appropriately.

In light of the above discussions, we find that the licensee has satisfied the Cycle 2 outage surveillance requirements for pre-startup reporting of fuel assembly inspections.

### 2.1.9 SURVEILLANCE REPORTING REQUIREMENTS

According to a commitment in the FSAR, AP&LCo is also to provide a final report describing the results of fuel inspections within 90 days following refueling. However, in light of (a) the unexpected damage (i.e., torn grid straps, fuel and poison rod perforations) observed after Cycle 1, (b) the preliminary nature of the conclusions attained to date, (c) the need for additional inspections to establish conclusive damage mechanisms that were operative during Cycle 1 and the refueling outage, and (d) the time required to formulate preventative measures to be employed in the future, we will relax the 90 day reporting requirement for this specific information. Nevertheless, AP&LCo should strive to submit this information as soon as it is available and certainly no later than 90 days prior to the next refueling outage.

### 2.1.10 DEMONSTRATION AND TEST FUEL

We find the use of the DOE demonstration fuel rods (Ref. 4) in the 2 Batch D fuel assemblies acceptable since (a) they contain few fuel rods in number and thus constitute a small portion of the Cycle 2 core, (b) they are to be placed in non-limiting positions, and (c) they have been analyzed with approved methods as were the standard fuel which will comprise the core. Furthmore we encourage such demonstration and test programs because they tend to lead to improved design and safety analyses of fuel performance.

Upon the same bases, we find the continued use of the C-E test fuel rods (Ref. 5) in the 2 Batch C fuel assemblies acceptable as well.

### 2.1.11 FUEL DESIGN CONCLUSION

We have reviewed the AP&LCo reload analysis (Refs. 1 and 2) and supporting information (Refs. 3, 10, 27, 28, and 29) submitted as justification for Cycle 2 operation of ANO-2. We have determined that all applicable requirements related to the reactor fuel design have been met. Therefore, we conclude that AP&LCo's application is acceptable.

TABLE 2 (continued)

| Calculational Factors | Reference Cycle 1 | Cycle 2 |
|---|---|---|
| Engineering Heat Flux Factor | 1.03 | 1.025++ |
| Engineering Factor on Hot Channel Heat Input | 1.03 | 1.008+, ++ |
| Rod Pitch, Bowing and Clad Diameter Factor | 1.05 | 1.05 ++ |
| Fuel Densification Factor (Axial) | 1.002 | 1.002 |
| Peak Linear Heat Generation Rate (kw/ft) | 14.5 | 14.5 |
| Minimum DNBR | 1.30 | 1.24++ |

NOTES:

* Based on 1128 shims.

\+ Based on "as-built" information.

++ These factors have been combined statistically with other uncertainty factors at 95/95 confidence/probability level to define a new design limit on CE-1 minimum DNBR when iterating on power as discussed in Reference 6-7.

## 2.4 ACCIDENTS AND TRANSIENTS

### 2.4.1 ANTICIPATED OPERATIONAL OCCURRENCES

#### 2.4.1.1 CONTROL ELEMENT ASSEMBLY (CEA) WITHDRAWAL

CEA withdrawal was reanalyzed for the conditions of Cycle 2 to demonstrate that the initial margins were maintained by the applicable values of the Technical Specification Limiting Conditions for Operation (LCOs) and the reactor coolant system design pressure limits. Trip signal calculations are performed within the Core Protection Calculator (CPC) where the algorithm uses core power, heat flux and reactor constants to provide a conservative estimate of the trip signals in such a way as to prevent exceeding MDNBR, maximum local power density, or RCS pressure. The CEA transient has been calculated for withdrawal from subcritical, one percent power and full power conditions. Including feedback effects and control rod position at critical, the most critical parameter for the subcritical case is reactivity addition rate. The input values selected maximized the power increase and the margin degradation. No safety limits were exceeded.

The CEA withdrawal from one percent power was similarly analyzed. The transient is terminated by a high pressurizer pressure trip. The resulting maximum RCS pressure is 2662 psia and occurs before the high LPD or the MDNBR trip would be activated. No DNBR or LPD limits are exceeded.

The review of the CEA withdrawal indicated that none of the LCOs will be exceeded, hence, the results are acceptable.

2.4.1.2 FULL AND PART LENGTH CEA DROP

The most important factor in such a transient is the possible rate of reactivity insertion. The CPC shall initiate a trip in a manner that the initial margins be maintained by the LCO to prevent violation of the DNB, CTM or LPD design limits. Two cases were analyzed i.e., full length and part length CEA drop. The CPC constants include CEA penalty factors which account for any CEA misalignment including a drop. The penalty factors assure a conservative estimate of the transient MDNBR and maximum LPD.

The single full length control rod drop can cause an increase of the peaking factors by 17 percent over predrop values. However, the CEA penalty factors in the CPC will prevent power distributions which could violate MDNBR limits.

The part length CEA drop not only can cause severe axial and radial flux distortions but it can insert positive reactivity. However, the CPC initiated MDNBR or maximum LPD will prevent the respective limits from being exceeded.

The methods used in the analysis are consistent with those used in the FSAR which we have previously reviewed and approved.

The review of the CEA misoperation indicates that CPC originated trips will prevent violation of MDNBR, CTM or LPD limits, hence it is acceptable.

2.4.1.3 FUEL MISLOADING

The original submittal did not deal with the potential consequences of fuel misloading on the assumption that such consequences would be no more severe than those analysed for the first cycle. At our request the licensee submitted additional information for the ANO-2 Cycle 2 misloading analysis. The analysis was based in part on the analysis performed for Calvert Cliffs Unit 2 Cycle 4. The analysis demonstrates that differences in power peaking and power distribution for fuel assemblies irradiated for one or two cycles will be detectable by symmetry checks. The misloading considered includes fuel assembly interchange and assembly misrotation. When the assemblies are such that their reactivity differences are not detectable with the CEA symmetry checks, the increase in power peaking will be small and will not reduce significantly the available margins.

In addition the licensee stated that the worst misloading event which can be postulated for ANO-2 Cycle 1 or Calvert Cliffs 2 Cycle 4 cannot occur in ANO-2 Cycle 2, hence, the latter is bracketed by the existing analyses. Hence, the consequences of undetectable misloadings for ANO-2 Cycle 2 will be less severe than those evaluated and approved for ANO-2 Cycle 1 and Calvert Cliffs Unit 2 Cycle 4. We find those arguments reasonable and the misloading case acceptable.

2.4.1.4 CLOSURE OF ONE MSIV

The transients resulting from the instantaneous closure of a single Main Steam Isolation Valve (MSIV) were analyzed for Cycle 2 to determine the Core Protection Calculator (CPC) Asymmetric Steam Generator Transient Protection (ASGTP) trip setpoint. This setpoint is determined in conjunction with the initial margins maintained by the LCOs so that the DNBR and fuel center-line-to-melt (CTM) design limits are not exceeded.

CESEC II version was used to simulate the primary system response, and CETOP/CE-1 was used in MDNBR analysis. Although CESEC II version has not been approved, the staff finds it acceptable for this application.

The four events which affect a single steam generator are: (a) loss of load to one steam generator; (b) excess load to one steam generator; (c) loss of feedwater to one steam generator; and (d) excess feedwater to one steam generator.

The licensee has justified, by the detailed studies documented in reference 5, that the loss of load to one steam generator (LL/1SG) event produces by far the largest margin degradation and thus is the most limiting asymmetric event. Since this event is most limiting it was the only asymmetric event analyzed for Cycle 2 to establish the ASGTP set points. The staff has reviewed the referenced studies and finds this approach acceptable.

This event was analyzed for Cycle 1 operation prior to the installation of the Asymmetric Steam Generator Transient Protection related CPC trip in ANO-2. This Cycle 2 analysis establishes the reference analysis for future cycles in which the ASGTP trip is operational.

The event is initiated by the inadvertent closure of a single main steam isolation valve causing a loss of load to one steam generator. Upon loss of load, pressure (and temperature) in the affected steam generator will increase to the opening pressure of the main steam safety valves. The intact steam generator picks up the load loss, which causes its temperature and pressure to decrease. The cold leg temperature asymmetry leads to a reactor inlet temperature tilt which produces an azimuthal core power tilt.

Conservative assumptions were used in the analysis to account for the maximum power tilt and hot channel radial peaking factor increase (the assumption used in this case is the most negative moderator temperature coefficient of $-3.5 \times 10^{-4}$ $\Delta\rho/^{o}F$ since this maximizes the power tilt and the increase in the hot channel radial peaking factor). With this conservative assumption, the greatest asymmetry in core inlet temperature distribution, the greatest increase in hot channel radial peaking factors and the most limiting DNBR of 1.24 will result. A reactor trip is generated by the CPC low DNBR trip on high differential temperature between the cold legs associated with the two steam generators. The ASGTP trip setpoint within the CPC ensures that the acceptable DNBR limit will not be exceeded at any time during the event. A maximum allowable initial linear heat generation rate of 16.5 KW/ft could exist as an initial condition without exceeding the maximum linear heat generation rate of 21.0 KW/ft above which fuel centerline melting is predicted to occur during this transient. This amount of margin is assured by setting the Linear Heat Rate LCO based on the more limiting of the allowable linear heat rate for LOCA 14.5 KW/ft and other transients. Initiating the event from the extremes of the LCO in conjunction with the CPC (ASGT protective) trip will prevent DNBR or centerline fuel temperatures from exceeding the design limits and the maximum pressure within the RCS and main steam systems from exceeding 110% of the design pressures.

The analysis results of this transient meet the acceptance criteria of SRP section 15.2.3.3 and are acceptable.

### 2.4.1.5 BORON DILUTION

An inadvertent boron dilution event adds positive reactivity by reducing the boron concentration in the primary coolant. This produces power and temperature increases in the core and may cause an approach to both the DNBR and the fuel centerline-temperature-to-melt (CTM) limits.

The boron dilution event was reanalyzed for Cycle 2 to demonstrate that (1) sufficient time is available for the operator to identify the cause of and to terminate an approach to criticality for all subcritical modes of operation and (2) to demonstrate that sufficient scram reactivity is available in all operating modes.

In a boron dilution event during power operation (Modes 1 & 2) the core protection calculator system's DNBR trip, or, for more rapid power excursions, the high logarithmic power level trip will occur prior to reaching the DNBR or CTM limits. The high pressurizer pressure trip will prevent the RCS pressure from reaching the RCS pressure limit. These trips will provide a positive means of alerting the operator to a boron dilution event in progress and will provide adequate time to terminate the boron dilution event. We find these results acceptable.

For a boron dilution event during the subcritical modes (Modes 3, 4, 5 and 6) the cold shutdown mode (Mode 5) with the vessel water level drained down to the lip of the outlet nozzle and the refueling mode provide the most limiting times from the initiation of the event until the five percent shutdown margin is exhausted and the reactor returns to critical. Considering this assumption for Mode 5 and 6, times of 35 and 40 minutes respectively are calculated to elapse between initiation of the dilution and loss of the five percent shutdown margin. However, the time of importance, to meet the acceptance criteria of the Standard Review Plan, is the time between the provision of a positive indication to the operator and a return to criticality. This time should be at least 15 minutes for Mode 5 and 30 minutes for Mode 6. Therefore it is the staff's position that a positive means for alerting the operator to a boron dilution event in progress be installed as soon as practical. In order to be able to take credit in the analysis for this alarm it must meet the single failure criteria per section II.2.C of SRP section 15.4.6.

### 2.4.1.6 LOSS OF LOAD/LOSS OF CONDENSER VACUUM/TURBINE TRIP

The loss of load (LOL), loss of condenser vacuum (LOCV), and turbine trip events are analyzed to demonstrate that the RCS and main steam system pressures do not exceed 110% of design values (i.e., 2750 psia and 1210 psia, respectively) for Cycle 2 operation. These three events were presented in the FSAR as separate events. For Cycle 2 an analysis was performed of a single event which bounds all three FSAR events.

The bounding event considered is a Loss of Load event initiated by a turbine trip without a simultaneous reactor trip, and assuming the Steam Dump and Bypass system is inoperable. If the turbine trip was caused by a Loss of Condenser Vacuum, the main feedwater pump steam turbines would trip at the same time. Therefore, to cover these events a LOL concurrent with loss of feedwater was analyzed. The loss of load causes steam generator pressure to increase to the opening pressure of the main steam safety valves. The reduced secondary heat sink leads to a heatup of the RCS and, in the presence of the assumed positive MTC, an increase in core power. The transient is terminated by a reactor trip on high pressurizer pressure.

Conservative assumptions were used in the transient analysis to account for (a) the steam dump and bypass valves which were assumed to remain closed, (b) a positive MTC of 0.5 x $10^{-4}$ $\Delta\rho$/°F, and a least negative Doppler coefficient with a fuel temperature coefficient multiplier of 0.85, and (c) a bottom peaked axial shape which minimizes the negative reactivity insertion during the initial portion of the scram following a reactor trip and maximizes the time required to mitigate the pressure and heat flux increase.

The Loss of Load transient analysis resulted in a peak reactor coolant pressure of 2671 psia. The increase in secondary pressure is limited by the opening of the main steam safety valves. The secondary pressure peak value of 1144 psia was reached at 13.3 seconds after initiation of the event.

The results of the analysis demonstrate that the Loss of Load type event will not result in peak RCS pressure or peak main steam pressure in excess of their respective upset pressure limits and that the minimum DNBR did not decrease below 1.24.

The analysis results for this transient meet the acceptance criteria of SRP Section 15.2.1 and are acceptable.

### 2.4.1.7 LOSS OF COOLANT FLOW

The Loss of Coolant Flow event was reanalyzed for Cycle 2 to determine the minimum initial DNBR margin that must be maintained by the Limiting Conditions for Operations (LCOs) and the margin degradation rate which must be projected by the Core Protection Calculators (CPCs) such that a low DNBR trip will be initiated before the DNBR limit is exceeded.

The methods used to analyze this event are consistent with those discussed in the FSAR with the exception that the design thermal margin model CETOP was used for all DNBR calculations. The acceptability of the changes in the analytical models are discussed in Section 2.3 of this report.

The 4-Pump Loss of Coolant Flow (LOCF) produces an approach to the DNBR limit due to the decrease in the core coolant flow. Aside from the basic assumption of 4-pump LOCF without a simultaneous reactor trip, other conservative assumptions were used in the LOCF transient analysis to reflect the following initial conditions: (a) the Technical Specification LCOs, and (b) an axial shape with a negative shape index of -.18.

The analysis for this transient showed that the minimum DNBR did not decrease below the 1.24 limit. The CPC low DNBR trip assures that loss of coolant flow events initiated from within the Technical Specification LCO's will not result in a violation of the DNBR design limit. The maximum pressure within the reactor coolant and main steam systems did not exceed 110% of the design pressures.

The analysis results for this transient meet the acceptance criteria of SRP Section 15.3.1 and are acceptable.

## 2.4.2 POSTULATED ACCIDENTS

### 2.4.2.1 CONTROL ELEMENT ASSEMBLY EJECTION

A zero and full power CEA ejection accidents have been analyzed. The analytical method, detailed in topical report CENPD-190-A, has been approved. The calculational procedure computes the radial average and centerline fuel enthalpies to determine the fraction of the pins that exceed the criteria for clad damage. To assure that the calculated values bound the most adverse conditions the following assumptions were made:

(a) the beginning of cycle (BOC) Doppler coefficient was assumed which yields the least negative reactivity feedback;

(b) the BOC moderator temperature coefficient of $.5 \times 10^{-4}$ $\Delta\rho/°F$ was assumed which results in a positive reactivity feedback; and

(c) an end of cycle (EOC) delayed neutron fraction was used which results in the highest power rise during the transient.

Very low (zero) and full power conditions were analyzed with both terminating from a high linear power level trip.

The results of the analysis indicate that a small fraction (<.005) of the fuel reaches the incipient centerline melt threshold. The total energy deposited during the transient is less than 200 cal/gm criterion for clad damage. The results for Cycle 2 compare well with the corresponding results for Cycle 1. The methodology and the results of the CEA ejection accident are acceptable.

### 2.4.2.2 SEIZED RCP SHAFT

The seized shaft event was reanalyzed for Cycle 2 to demonstrate that the RCS pressure limit of 2750 psia will not be exceeded and only a small fraction of fuel pins are predicted to fail during this event which will not cause the 10 CFR 100 site boundary dose guidelines to be exceeded.

The single reactor coolant pump shaft seizure is postulated to occur as a consequence of a mechanical failure. In this hypothetical event, the RCS flow rapidly decreases to the three-pump value. A reactor trip is initiated by a low coolant flow rate which results in a rapid reduction in the margin to DNB, so that a CPC low DNBR trip occurs to terminate the transient.

The analysis for this transient used an axial shape index value of -.18. This case is selected to be consistent with the Loss of Flow case.

TABLE 1

ANO-2 CYCLE 2 CORE LOADING INVENTORY

| Assembly Designation | Number of Assemblies | Initial Enrichment w/o U235 | BOC Burnup Average (GWd/t) | EOC Burnup Average (GWd/t) |
|---|---|---|---|---|
| A | 1 | 1.93 | 13.2 | 21.0 |
| B | 60 | 2.27 | 14.1 | 24.5 |
| C | 56 | 2.94 | 9.7 | 21.6 |
| D | 40 | 3.48 | 0 | 9.7 |
| | | 3.03 | | |
| D* | 20 | 3.03 | 0 | 13.5 |
| | | 2.73 | | |
| | 177 | | | |

## 2.2 NUCLEAR ANALYSIS

The nuclear design analysis used in Cycle 1 (reference cycle) has been performed with the PDQ07 (fine mesh) computer code. The Cycle 2 analysis is based on the ROCS (coarse mesh) code in a manner consistent with Calvert Cliffs Units 1 and 2 and St. Lucie Unit 1. The ROCS code is considered as a "state-of-the-art" code which has been used extensively by Combustion Engineering, Inc. (CE) and is acceptable. The use of ROCS has been limited to the calculation of three dimensional effects while local power peaking is calculated with PDQ. Few-group cross sections for input to both codes have been computed using the DIT code, a multigroup transport theory code. The following safety parameters were calculated:

- Critical Boron Concentrations,
- Boron Worths,
- Moderator Temperature Coefficient,
- Doppler Coefficient,
- Total Delayed Neutron Fraction, $\beta_{eff}$,
- Neutron Generation Time, $\ell^*$,
- Available CEA Worths, and
- Required Worth Allowances.

### 2.2.1 NUCLEAR PARAMETERS

The expected Cycle 1 termination burnup is 12,500 MWD/MT and the corresponding expected Cycle 2 full power operation burnup is 10,500 MWD/MT. For both rodded (partial length control element assemblies (CEAs), bank 6) and unrodded configurations, the maximum power peaks occur at the BOC-2. Augmentation factors for Cycle 2 have been calculated and they include the effects of fuel densification, radial pin power distribution, single gap peaking factors, and burnup.

The Cycle 2 planar radial peaking factor uncertainty is 5.3 percent and is based on the topical report CENPD-153-P, Rev. 1-P-A which is an NRC approved report and is, therefore, acceptable. This value is conservative with respect to the maximum value of the reference cycle.

The Cycle 2 moderator coefficient is calculated to be $-.5 \times 10^{-4}$ $\Delta\rho/°F$ at BOC and $-2.3 \times 10^{-4}$ $\Delta\rho/°F$ at EOC. These values are bounded by the reference cycle values (i.e., -.5 and $-2.5 \times 10^{-4}$ $\Delta\rho/°F$). The fuel temperature coefficient (Doppler) values for Cycle 2 are slightly more negative than the values of Cycle 1. However, the extended pointwise Doppler feedback technique has been used which involves use of iterations of pointwise power distribution and pointwise fuel temperature instead of using precalculated fuel temperatures. It is estimated that the Cycle 2 Doppler coefficient values are more accurate. We find the moderator and the fuel temperature coefficient values to be acceptable.

At the beginning of Cycle 2 (BOC-2), the reactivity worth of all CEAs inserted (assuming the highest worth CEA is stuck out) is 9.3 percent $\Delta\rho$. The reactivity worth required for shutdown including the power defect from hot full power to hot zero power and the CEA bite (i.e., the fact that CEAs may be slightly inserted instead of being fullly withdrawn) is 2.5 percent $\Delta\rho$. The excess CEA worth available for normal shutdown is, therefore, 6.8 percent $\Delta\rho$. At the end of Cycle 2 (EOC-2) the corresponding excess CEA worth is 6.6 percent. The required shutdown margin is 5 percent $\Delta\rho$, hence, the available margin is negative and more than adequate to account for possible uncertainities. We find these shutdown margins acceptable.

The consequences of a dropped CEA were analyzed. The limiting safety analysis values for dropped CEA increase in radial peaking factor is 17 percent for Cycle 2 compared to 27 percent for the reference cycle. However, the Cycle 2 value is conservative compared to the actual calculated values and is acceptable.

### 2.2.2 UNCERTAINTY IN NUCLEAR PEAKING FACTORS

Incore detector measurements are used to compute the core peaking factors using the INCA code. The methodology, the required coefficients and the reduction are described in the approved topical report. As mentioned above, the planar radial power distribution measurement uncertainty is 5.3 percent and will be applied in Cycle 2 to COLSS and the CPC on-line calculations. On this basis we find these measurement uncertainties to be acceptable.

## 2.3 THERMAL AND HYDRAULIC DESIGN

We have reviewed the Cycle 2 reload to confirm that the thermal and hydraulic design of the reload core has been accomplished using acceptable analytical methods and provides acceptable margin of safety from conditions which would lead to fuel damage during normal reactor operation and anticipated operational transients.

ANO-2 Cycle 2 consists of presently operating Batch A, B, and C assemblies, along with fresh Batch D assemblies. The Cycle 1 termination burnup has been assumed to be approximately 12,500 MWD/t. Our review consisted of the following: (a) statistical combination of uncertainties to calculate minimum DNBR; (b) CETOP-D and CETOP-2 thermal hydraulic (T-H) computer codes used for DNBR analysis; (c) CE-1 correlation used for DNBR analysis; (d) effects of fuel rod bow on DNBR margin (see Section 2.1); (e) comparison of the Cycle 2 thermal-hydraulic parameters at full power with those of Cycle 1; (f) CPC and CEAC software modifications; (g) addition of asymmetric steam generator transient protection trip in CPC based on instantaneous closure of a single MSIV (ICSM) transient (see Section 2.4); (h) determination and evaluation of the most limiting event from the design basis Anticipated Operational Occurrences (AOOs) for which a DNB trip occurs and the thermal margin LCOs is maintained; and (i) proposed Technical Specifications modifications.

### 2.3.1 STATISTICAL COMBINATION OF UNCERTAINTIES

The criteria established in 10 CFR 50, Appendix A, imposes the requirement of a high degree of assurance that neither the phenomenon known as DNB (Departure from Nucleate Boiling) nor melting at the fuel centerline occurs. Calculational methods have evolved over the years that predict the conditions causing the phenomena. The results of the calculations are then entered into the reactor protection system methodology to provide the necessary assurance neither occurs.

There is a degree of uncertainty in the knowledge of the exact value of each of the variables used. This uncertainty has been handled in the past by assuming that each variable is at its extreme most adverse limit of its uncertainity range. The assumption that all factors affecting DMB and fuel centerline temperatures are simultaneously at their most adverse value is very unlikely and leads to conservative restrictions on reactor operation. The potential for greater operational flexibility has provided a strong incentive to reduce the degree of conservatism.

The licensee has proposed use of a new methodology that reduces the conservatism by statistically combining the uncertainties. The report CEN-139(A)-P, (Ref. 2.3-1) describes the methodology to calculate new MDNBR limits for ANO-2. It ensures with at least 95 percent probability and 95 percent confidence level that DNB will not occur.

CEN-139(A)-P describes methods used to statistically combine uncertainties in those variables which are not monitored while the reactor is in operation. The methods are then used to develop a new MDNBR. The variables so considered are termed system variables and include such things as reactor geometry, pin-by-pin power distributions, and inlet and outlet flow boundary conditions. The variables affecting DNB whose uncertainties are not considered are those which are monitored during reactor operation and are termed state variables. Though it is not specifically stated, the state variables are considered in the CPC and are described in other documentation supporting operation of ANO-2.

The licensee proposes that the difference in the basic DNBR limit value of 1.19 discussed in the section on the CE-1 correlation and the limit value of 1.24 is sufficient to account for these uncertainties. Our review of SCU has not progressed sufficiently to enable us to make a finding on the precise value of the thermal margin credit gained by inclusion of SCU in the DNBR limit. We are currently reviewing the individual uncertainty components of the system parameters to evaluate the SCU credit.

However, we have reviewed the basic SCU methodology and find it acceptable. Upon completion of our review, if it is required, we will require that any reduction in the credit currently proposed for SCU by the licensee be accounted for prior to authorizing full power operation. If necessary this would most likely be done through adjustment of the addressable constant on power uncertainity, BERR 1. The licensee has proposed an interim value of 1.112 for BERR 1 and we find this vaule should be adjusted upward by 5.6% to account for rod bow compensation described in the technical specifications. We, therefore, conclude that the BERR 1 value of 1.174 should be used for the interim period of operation of ANO-2 at reduced power level.

### 2.3.2 CETOP-D COMPUTER CODE

The CETOP-D computer code is used as a core thermal margin design analysis tool for the Cycle 2 reload. CETOP-D is an open lattice thermal hydraulic code which solves the same conservation equations and uses the same constitutive equations as in the TORC code (CENPD-161-P). TORC, derived from COBRA-III C is a multi-stage thermal margin code. Based on the magnitude of the changes in the CETOP-D code relative to its predecessors we undertook a complete review of the CETOP-D code as a design analysis tool.

Although our review of the CETOP-D code is near completion the details of our evaluation of the code have not been finalized. Our summary evaluation of the code is based on comparisons provided by the licensee of CETOP-D results to TORC results over a wide spectrum of operating conditions for ANO-2, Calvert Cliffs Units 1 and 2, and San Onofre Units 2 and 3. In all cases, referenced in the response to questions 492.7 and 492.68, the CETOP-D code predicts the minimum DNBR to be lower than does TORC. Since we have previously approved TORC for use in CE thermal margin plant analyses we conclude, based on the conservatism of CETOP-D relative to TORC, that the CETOP-D code is acceptable for ANO-2 thermal margin calculations, with the condition that the hot assembly inlet flow factor with the value described in response to NRC question 492.14, or a smaller value, be used.

### 2.3.3 CETOP-2 COMPUTER CODE

The staff has reviewed the CETOP-2 functional specification and has performed an audit of the functional tests of the integrated system to assure that CETOP-2 with the algorithm uncertainty factor is programmed properly and predicts minimum DNBR conservatively.

The CETOP-2 functional description is provided in the Appendix B of CEN-143. The following is a summary of the results of our review:

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

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

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

(d) CETOP-2 uses lumped channel modeling wherein the core is divided into four modeling channels, i.e., core region channel, hot assembly channel, buffer channel, and hot channel. The hot channel is a pseudo-hot channel which models a corner guide tube subchannel. The staff has raised questions (Ref. 3-2) as to how the hot channel is selected; whether the selected hot channel always predicts the lowest DNBR; whether minimum DNBR always occurs in a guide tube

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

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

(f) The algorithm uncertainty factor is a compensation applied to the core power in CPCs to ensure that the DNBR results from CETOP-2 are conservative relative to CETOP-D. The licensee has had 6400 cases run of comparisons between CETOP-2 and CETOP-D; and a compensation factor has been derived so that application of the compensation factor to the core power results in a 95/95 probability/confidence level that CETOP-2 is more conservative than CETOP-D. These cases are run using the Value of BERR 1 equal to 1.0. Using the algorithm uncertainty power compensation factor or a larger value as a core power multiplier will result in a conservative DNBR prediction from CETOP-2.

(g) Based on the compensation factor being built into the CETOP-2 software as a plant specific constant and our other findings as reported in (a) through (f) above, the staff concludes that the CETOP-2 code design as applied to ANO-2 is acceptable.

### 2.3.4 CE-1 CORRELATION

For ANO-2 Cycle 2 the critical heat flux correlation (CHF) has been changed from the W-3 correlation to the CE-1 correlation. The CE-1 correlation has previously been approved by the staff for interim plant specific application with a minimum DNBR limit of 1.19. Based on the results of our review of ANO-2 Cycle 2 operation we conlcude that the value of 1.19 is consistent with the submitted data base. Therefore, the DNBR limit for the CE-1 correlation is 1.19 before any other uncertainties are accounted for.

This value of 1.19 is consistent with the licensee's proposal and is acceptable. The accounting of other uncertainties, such as SCU and rod bow and the final value of the limit for ANO-2 Cycle 2 (1.24) are discussed in other sections of this report.

### 2.3.5 CPC/CEAC SOFTWARE MODIFICATIONS AND PHASE II TEST RESULTS:

The Core Protection Calculators (CPC) and Control Element Assembly Calculators (CEAC) of the ANO-2 Cycle 2 are basically identical hardware with a modified version of the software from that of Cycle 1. The major software modification includes (i) the use of CETOP-2 in place of CPCTH for core thermal hydraulic calculations; (ii) replacing W-3 with CE-1 correlation for CHF calculations; (iii) the use of a statistical combination of uncertainties (SCU) of system parameters to derive a new DNBR limit. The licensee has submitted a summary of the CPC/CEA software modifications over that of Cycle 1 (CEN-143(A)-P.

Since the Cycle 1 CPC/CEAC had been reviewed extensively and approved, the staff's review effort of the Cycle 2 CPC/CEAC has been concentrated on the software modification.

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

The resulting initial DNBR, initial LPD and the trip times from the single channel test fall well within the acceptance criteria for each case established from the FORTRAN simulation runs. For a few cases where the trip time fails to stay within the acceptance criteria, the cause has been identified to be the differences in interpolation of time dependent parameters between the single channel and FORTRAN simulation.

The staff has made an audit on the Phase II test and confirmed the accuracy of the report. The agreement of the Phase II testing has shown the adequacy of the implementation of the functional specification. Therefore, the staff concludes that the software modification implementation is acceptable.

### 2.3.6 THE MOST LIMITING EVENT FOR WHICH A DNB TRIP OCCURS AND THERMAL MARGIN MAINTAINED:

Results of the analyses performed by the licensee (Ref. 2) indicated most limiting AOO's on the basis of DNBR and CTM limits are (i) control element assembly withdrawal and (ii) loss of coolant flow.

CESEC II version was used to simulate the primary system response, and CETOP/CE-1 was used in DNBR analyses.

The staff has reviewed the initial conditions used in the analyses of the above transients. With the initial power level assumed to be 103% of the rated power the final transient results show that DNBR$\geq$ 1.24 and PLHR $\leq$21.0 kw/ft and the staff finds these results acceptable.

### 2.3.7 COMPARISON OF CYCLE 1 TO CYCLE 2

Comparison of thermal hydraulic design conditions for ANO-2 Cycle 1 and Cycle 2 is provided in Table 2. It can be seen that the difference in Cycle 1 and Cycle 2 design parameters is in calculational factors. This is due to application of SCU and new methodology CETOP/CE-1.

TABLE 2

Arkansas Nuclear One Unit 2
Thermal Hydraulic Parameters at Full Power

| General Characteristics | Unit | Reference Cycle 1 | Cycle 2 |
|---|---|---|---|
| Total Heat Output (Core only) | MWt<br>$10^6$ Btu/hr | 2815<br>9608 | 2815<br>9608 |
| Fraction of Heat Generated in Fuel Rod | - | 0.975 | 0.975 |
| Primary System Pressure | | | |
| Nominal | psia | 2250 | 2250 |
| Minimum in steady state | psia | 2203 | 2203 |
| Maximum in steady state | psia | 2297 | 2297 |
| Inlet Temperature (Maximum Indicated) | °F | 554.7 | 554.7 |
| Total Reactor Coolant Flow (Minimum Steady State) | gpm<br>$10^6$ lb/hr | 322,000<br>120.4 | 322,000<br>120.4 |
| Coolant Flow Through Core (Minimum) | $10^6$ lb/hr | 116.2 | 116.2 |
| Hydraulic Diameter (Nominal channel) | ft | 0.039 | 0.039 |
| Average Mass Velocity | $10^6$ lb/hr-ft$^2$ | 2.60 | 2.60 |
| Pressure Drop Across Core (Minimum steady state flow irreversible $\Delta p$ over entire fuel assembly) | psi | 15.6 | 15.5 |
| Total Pressure Drop Across Vessel (Based on nominal dimensions and minimum steady state flow) | psi | 35.9 | 35.9 |
| Core Average Heat Flux (Accounts for fraction of heat generated in fuel rod and axial densification factor) | BTU/hr-ft$^2$ | 184,720 | 184,720* |
| Total Heat Transfer Area (Accounts for axial densification factor) | ft$^2$ | 50,707 | 50,707* |
| Film Coefficient at Average Conditions | BTU/hr-ft$^2$°F | 6200 | 6200 |
| Average Film Temperature Difference | °F | 30.6 | 30.6 |
| Average Linear Heat Rate of Undensified Fuel Rod (Accounts for fraction of heat generated in fuel rod) | kw/ft | 5.40 | 5.40* |
| Average Core Enthalpy Rise | BTU/lb | 82.7 | 82.7 |
| Maximum Clad Surface Temperature | °F | 656.5 | 656.5 |

The analysis, however, did not take into consideration the single failure criterion. It is our position that the licensee should provide a confirmatory analysis that shows that the consequences of this accident, including the worst single failure, will still meet the primary system pressure limit and the 10 CFR 100 dose limit for this transient.

2.4.2.3 STEAM LINE BREAK

The steam line break (SLB) transient is an overcooling event. The full power SLB events were reevaluated for Cycle 2 to account for the reduced shutdown margin from -8.6 to -7.9% $\Delta\rho$, increased Doppler feedback, and decreased reactivity insertion during moderator cooldown. The steam generator blowdown and associated reactor coolant system (RCS) cooldown were not recalculated for Cycle 2 since the net effect of changes in the above parameters on the blowdown will be small. The Cycle 1 cases are based on cooldown curve associated with an initial allowable MTC of $-3.5 \times 10^{-4}$ $\Delta\rho/°F$, while the Cycle 2 cases are based on the cooldown curve associated with an initial allowable MTC of $-2.8 \times 10^{-4}$ $\Delta\rho/°F$. Comparison of the Cycle 1 and Cycle 2 results from these curves shows that the positive reactivity insertion due to cooldown of the moderator is less for Cycle 2 by 1.1% $\Delta\rho$ at the time of minimum negative reactivity. This improvement in the moderator cooldown behavior is sufficient to completely offset the 0.7% $\Delta\rho$ decrease in available shutdown worth and .2% $\Delta\rho$ increase in positive reactivity insertion due to Doppler feedback.

The results of the analysis of the spectrum of steam line break accidents demonstrated that the peak reactivity experienced during the transient for Cycle 2 is bounded by the FSAR results. On this basis, the licensee concluded that the FSAR results are conservative and that the conclusions presented in the FSAR remain valid for Cycle 2. The licensee's analyses showed that based on DNBR criteria, no fuel damage will result. Without fuel damage a detailed dosage reassessment is not required.

Based on the above, the staff concludes that the analysis results for this transient meet the acceptance criteria of SRP Section 15.1.5 and are acceptable.

2.4.2.4 FEEDWATER LINE BREAK

The acceptance criteria for this event as stated in SRP Section 15.2.8 are that the RCS pressure should not exceed 110% of design pressure and any fuel damage predicted to occur should be sufficiently limited such that core coolability is maintained. The feedwater line break accident was reanalyzed for Cycle 2 to determine that the RCS pressure upset limit 2750 psia is not exceeded during the transient, and that any fuel damage predicted to occur is limited.

The feedwater line break analyzed was assumed to occur during full power operation and with concurrent loss of non-emergency A-C power at time of trip. This results in the maximum initial stored energy and minimum steam generator inventory. In addition, in response to loss of non-emergency AC power upon trip, the following will occur to maximize RCS pressure increase: (1) turbine trip valves close immediately; (2) reactor coolant pumps begin to coastdown; (3) pressurizer control systems are lost; and (4) 112.4 seconds rather than 97.4 seconds are required for the automatic initiation of emergency feedwater to the unaffected steam generator. This combination of parameters maximizes the calculated RCS peak pressure. The limiting break size was determined by a parametric study performed with the methodology previously reported in the FSAR.

The results of the Cycle 2 reanalysis predicted that the RCS pressure would increase to 2705 psia. Following reactor trip on either high pressurizer pressure or low steam generator water level, the decay in core power and the action of the primary and secondary safety valves result in a reduction of RCS pressure. Subsequently, the effects of system flow coastdown due to loss of AC upon trip, continued blowdown of steam from the intact steam generator through the break and the entering of emergency feedwater to the intact steam generator cause the RCS first to go through a mild pressure increase and then a steady decrease. The decrease is reversed when the low steam generator pressure initiates the closure of Main Steam Isolation Valves (MSIV). The MSIV closure terminates the blowdown of steam through the break, thus causing the RCS to heat up once more. Eventually, the heatup is terminated by the opening of secondary safety valves.

The results of this analysis demonstrate that the Feedwater Line Break Event will not result in a peak RCS pressure which exceeds the upset pressure limit of 110% of the design pressures. The licensee's analyses showed that based on DNBR criteria no fuel damage will result. Without fuel damage a detailed dosage reassessment is not required.

Based on the above, the staff concludes that the analysis results for this transient meet the acceptance criteria of SRP Section 15.2.8 and are acceptable.

### 2.4.3 LOSS-OF-COOLANT ACCIDENT

Much of the analysis for Cycle 1 operation was used as the basis for the Cycle 2 evaluation. Only the fuel pin thermal analysis using STRIKIN-II and the PARCH codes was performed for the Cycle 2 worst break. It was not necessary to repeat the blowdown and reflood hydraulic analyses since the analyses performed for Cycle 1 are applicable to Cycle 2.

The table below compares the results of the Cycle 2 analysis with the Cycle 1 analysis. As shown, the performance requirements of 10 CFR 50.46 are not exceeded. We, therefore find the LOCA analyses acceptable.

ANO-2 LIMITING BREAK (1.0 DEG/PD) RESULTS

| Case | Peak Cladding Temperature (°F) | Peak Local Oxidation (%) | Core Wide Oxidation (%) |
|---|---|---|---|
| Cycle 1 | 2078 | 11.82 | < 0.617 |
| Cycle 2 | 2041 | 11.80 | < 0.621 |
| 10 CFR 50.46 | 2200 | 17.0 | 1.0 |

### 2.4.4 RADIOLOGICAL CONSEQUENCES OF POSTULATED ACCIDENT

The licensee provided revised evaluations where previous Cycle 1 values did not bound Cycle 2 values. As shown below, the Cycle 2 radiological consequences are acceptably small fractions of the 10 CFR 100 limits for accidents.

#### 2.4.4.1 SEIZED RCP SHAFT

Because of numerous changes in parameters and methodology the number of rods calculated to have DNBR values below the limit is lower in Cycle 2 than in Cycle 1. Therefore, the radiological consequences for the seized reactor coolant pump shaft event are no greater that previously approved values.

#### 2.4.4.2 CONTROL ELEMENT EJECTION

The licensee reevaluated the control element assembly ejection accident for Cycle 2 and concluded, using methods previously approved in CENPD-190, that the fraction of rods predicted to suffer clad damage for the limiting case is less than the fraction predicted for Cycle 1. Therefore, the radiological consequences for CEA ejection accident are no greater than previously approved values.

## 2.5 REACTOR PROTECTION SYSTEM

### 2.5.1 VERIFICATION AND VALIDATION OF CPC SOFTWARE MODIFICATIONS

The licensee proposed a number of modifications to the Reactor Protection System's Core Protection Calculator System software. The principal purpose of the changes was to implement new thermal hydraulic and physics algorithms. The acceptability of the new algorithms is discussed in Section 2.3 of this report. We have reviewed the verification and validation procedures for changes to CPCS software which were followed to assure that the new algorithms have been incorporated into the CPCS software as intended. The verification and validation procedures were originally reviewed and approved by the NRC staff during the operating licensing review for ANO-2. These procedures include provisions for confirmatory testing of a modified prototype but single channel CPCS. Based on an audit of the licensee's procedures and test program, we conclude that the procedures and tests previously accepted by the staff have been followed during the current round of CPCS modifications.

On the basis that acceptable procedures and test programs have been followed in modifying the CPCS software, we consider the implementation of the new software to be acceptable.

### 2.5.2 TECHNICAL SPECIFICATIONS TO CONTROL MODIFICATIONS TO CPC ADDRESSABLE CONSTANTS

The licensee has proposed an increase in the number of CPC "addressable constants". Addressable constants are variables which may be modified between cycles or even during reactor operation. Because the CPC is a part of the ANO-2 protection system, we believe that appropriate measures should be taken to assure that such modifications are done correctly and that the new values of the constants inserted do not decrease safety margins. Consequently, we asked that the ANO-2 Technical Specifications be amended to include provisions to control modifications to addressable constants. This request was also prompted by the proposal by the licensee to permanently connect the data links which allow transfer of information from the CPC and CEAC systems to the plant computer, an issue adressed more fully in Section 2.5.3 of this report.

Specifically, the licensee has proposed in a letter dated May 19, 1981 (Ref. 5) the following controls:

(1) All CPC addressable constants are to be identified in the Technical Specifications.

(2) The bases to the Technical Specifications (Section 2.2.2) shall reference a document which explains the methodology and procedures for obtaining modified values of addressable constants.

(3) Those addressable constants expected to be modified frequently and from the operator's console will be restricted to specified ranges unless approved by the Plant Safety Committee.

(4) Those addressable constants expected to be modified less frequently by loading from a disc storage unit shall be approved by the Plant Safety Committee unless the modification is based on a technical specification or core physics test requirement.

(5) An independent verification shall be conducted to confirm that the desired value of each constant to be modified has actually been entered.

(6) Modifications to the CPC addressable constants based on information obtained through the plant computer (CPC data links) shall not be made without prior approval of the Plant Safety Committee.

Although a complete and approved document to meet provision (2) will not be available for Cycle 2 startup, by letter dated May 26, 1981 (Ref. 6), an interim document has been provided with a commitment to provide a final document by August 17, 1981.

We consider the proposed Technical Specification controls on addressable constant modifications to be acceptable, including the delays in providing a final document specifying the methodology for modifying constants.

### 2.5.3 DATA LINKS BETWEEN THE CPC/CEAC AND THE PLANT COMPUTER

The licensee has proposed to permanently connect the plant computer, a non-safety grade computer, to the core protection calculators (CPC), and control element assembly calculators (CEAC), part of the safety grade protection system. A similar proposal was made during the operating license (OL) review but was rejected by the staff because of concerns that the connection added unnecessary complexity to the CPC/CEAC design, and that there might be an adverse indirect effect on the protection system if data from the plant computer were used in calibration of the CPC addressable constants. The issue was discussed in our safety evaluation report for ANO-2 OL, NUREG-0308, and in particular in relation to Position 20 of Table 7.1 of that document and its supplements.

The concern that data from the plant computer might be used to modify CPC addressable constants and thereby adversely affect the CPCs has been addressed by establishing controls on the modifications of CPC addressable constants in the Technical Specifications (Section 2.2.2). As discussed elsewhere in this report changes to addressable constants based on data from the plant computer may be made only upon approval of the Plant Safety Committee. We consider this to be an acceptable resolution of this concern.

The staff concern about the unnecessary complexity associated with the data link design at the time of OL review was a general concern rather than one based on a potential deficiency in the measures taken to physically isolate the plant computer from the CPCs and CEACs. The use of qualified optical isolation devices at both ends of the digital data links and use of qualified current-to-current isolation devices for the analog data links to the plant computer preclude the possibility of a fault in the plant computer being propagated to the CPCs or CEACs. Furthermore, the watch-dog timers are used to prevent delay in a needed CPC trip should an inordinate time be spent in processing data through the data links to the plant computer.

Although the existence of the data links adds some complexity to the CPC/CEAC design as stated in the OL SER (NUREG-0308), we have reconsidered the issue and believe that the possibility of an adverse impact on safety is remote. Also, the new controls on CPC addressable constant modifications will prevent an unacceptable impact on the CPCs from recalibration using plant computer data. We, therefore, conclude that the permanent connection of the data links between the CPC/CEAC and the plant computer is acceptable.

### 2.5.4 MONITORING OF CPC ROOM TEMPERATURES

During Cycle 1 operation the licensee reported instances where sequences of CPC auto restarts were attributed to high CPC room temperatures. To assure that high room temperatures do not affect CPC reliabiltiy, we requested that the licensee address this issue in the Technical Specifications. The licensee has proposed Technical Specification 4.3.1.1.6 to require a CPC channel functional test if a CPC room high temperature alarm is received. We consider this acceptable.

## 2.6 LICENSE CONDITIONS

The licensee has satisfactorily completed the requirements of the following four license conditions. Accordingly, these conditions are hereby deleted from the license. The fifth license condition in the list below has also been modified as stated.

### 2.6.1 FUEL PERFORMANCE

License Condition 2.C.3.a on fuel performance required that, prior to startup for that fuel cycle in which burnups greater than 20,000 MWd/Mt, the Commission be provided with fission gas release calculations and analyses using calculational methodology approved for burnups greater than 20,000 MWd/Mt. This matter has been acceptably resolved by the licensee as is discussed in more detail in Section 2.1.1 of this report.

### 2.6.2 INSTRUMENT TRIP SETPOINT DRIFT ALLOWANCE

License Condition 2.C.3.d required the licensee to submit values of (1) instrument drift (2) cumulative instrument bias and (3) the margin between the trip setpoint and the assumed accident analysis trip value for inclusion in the Technical Specifications.

The staff's initial request on this subject was transmitted to the licensee by letter dated March 22, 1977. The licensee responded to that letter and to the license condition by letters dated February 28, 1979 and November 27, 1979. The staff assigned review of the licensee's submittals to the Lawrence Livermore National Laboratory (LLL) under the staff's technical assistance program.

The licensee's submittals included specific values of the reactor protection system and engineered safety feature actuation system trip setpoints for inclusion in Technical Specification Tables 2.2-1 and 2.2-2, respectively. The staff's consultant (LLL) concluded that the proposed changes to the setpoint values are acceptable. Further modification of some of these values has been proposed by the licensee to accommodate the Cycle 2 Reload Report analyses requirements. We have evaluated these differences in setpoint values and on the bases that the differences are relatively small and the effects of the differences are reflected in the Cycle 2 Reload Report analyses results, we conclude that the proposed Cycle 2 values are acceptable. A related change to the pressurizer high pressure trip setpoint is discussed in Section 2.7.1 of this report.

The licensee's submittals also included a report describing the setpoint methodology that Combustion Engineering, Inc. used to determine the ANO-2 setpoints. The LLL review of the methodology concluded that the method used for determining the total equipment error is a reasonable method for determining the RPS and ESFAS trip setpoints and allowable values.

Further details of the LLL review are contained in a copy of their Technical Evaluation Report which is attached to this report. Based on our review of the LLL report and the Reload Report as described above, we have concluded that the RPS and ESF Technical Specification setpoint values as identified in the Reload Report are correct and acceptable.

### 2.6.3 RCS OVERPRESSURE MITIGATING SYSTEM

License condition 2.C.3.f required that the licensee achieve full implementation of the proposed reactor coolant system overpressure mitigation systems described in the licensee's letter dated October 11, 1977 prior to Cycle 2 startup. The staff has previously reviewed the proposed design as reported in Supplement No. 1 to the SER dated June 1978. The design was approved subject to the stipulation that the design be demonstrated to meet the following specific criteria:

(1) Provide an interlock or alarm on the isolation valves which meets the applicable IEEE Standard 279 criteria and seismic Category I criteria for valves numbered 2CV-4730-1, 2CV-4731-2, 2CV-4720-2 and 2CV-4741-1, such that if the reactor coolant system temperature drops below the proposed temperature, and all the isolation valves are not fully open, an alarm sounds in the control room or the isolation valves open automatically.

(2) The electrical portion of the permanent fix conforms to safety-grade criteria.

We requested additional information and the licensee responded with references 2 and 3. Based on our review we conclude that the design meets the above criteria and is acceptable.

### 2.6.4 CEA GUIDE TUBE SURVEILLANCE PROGRAM

License Condition 2.C.3.l required that prior to Cycle 2 startup the licensee submit the results of a surveillance program conducted on the design modifications to the CEA guide tubes. The program was to be directed toward determining whether unacceptable degradation of the guide tube components had occurred.

Our review of this matter is presented in Section 2.1.5 of this report wherein we conclude that the issue of guide tube wear is resolved for ANO-2. Therefore, on the basis of our findings and conclusions as presented in Section 2.1.5 of this report license condition 2.C.3.l is satisfied and is hereby deleted from the license.

### 2.6.5 MAXIMUM POWER LEVEL

License Condition 2.C.1, Maximum Power Level, presently contains the requirement for the licensee to complete certain preoperational tests, startup tests and other items identified in Attachment 2 to the license. Attachment 2 contained items for which completion was required prior to attaining full power following the initial licensing of the plant. These items were the remainder of the total list of tests at the time the license was issued. Accordingly, once the plant's preoperational and first startup testing program has been acceptably completed no need exists for Attachment 2. Therefore, license Condition 2.C.1 is modified to delete reference to Attachment 2 and Attachment 2 is deleted from the license.

License Condition 2.C.1 is also modified to limit plant operation to seventy percent of the licensed power level of 2815 MWt pending completion of the final details of the staff's review of the Core Protection Calculator system software changes for Cycle 2 operation as discussed in Section 2.3 of this report.

## 2.7 OTHER MATTERS

### 2.7.1 CONTAINMENT PRESSURE, TEMPERATURE, AND HUMIDITY

On November 19, 1979, the licensee submitted a proposed change to Technical Specification 3.6.1.4, on containment atmosphere conditions. The proposed change would reduce the allowable containment temperature over a range of pressures. The change is proposed to make the TS consistent with assumptions contained in the FSAR on initial containment pressure, temperature and relative humidity such that the maximum differential pressure across the containment would be 5.0 psi in the event of an inadvertent containment spray actuation. This will assure that the containment pressure, as a result of inadvertent spray operation, will not be lower than the containment's external design pressure of -5.0 psig. The current specification does not provide this assurance.

The staff has reviewed the proposed change to the Technical Specifications and on the basis of audit calculations finds that the proposed change will assure that the external design pressure will not be exceeded for inadvertent spray operation. We conclude that the proposed change is acceptable and should be implemented.

### 2.7.2 PRESSURIZER HIGH PRESSURE TRIP SETPOINT

The licensee, in its submittal of November 27, 1979, requested a change to Unit 2 Technical Specifications (TS). The high pressurizer pressure trip setpoint, Item 4 of Table 2.2-1 of the Technical Specifications, is presently $\leq$ 2345 psia. It is proposed to increase the trip setpoint by 17 psi to $\leq$ 2362 psia.

The increase in the high pressurizer pressure trip setpoint of 17 psi is to eliminate a dynamic allowance imposed prior to operation. The test data collected at startup of Cycle 1 has demonstrated an instrument channel response time less than assumed in the safety analysis. Therefore, the dynamic allowance factor is no longer required.

The existing narrow range pressurizer pressure instrument used for this trip has a range of 1500 to 2500 psi. The present trip setpoint is $\leq$ 2345 psia with an allowable drift of 8.887 psi (allowable value of $\leq$ 2353.887 psia). The proposed trip setpoint is $\leq$ 2362 with an allowable drift of 8.887 psi (allowable value of 2370.887 psia). The new trip setpoint is well within the range of the narrow range pressurizer pressure instrument and the allowable drift (8.887 psi) for the proposed setpoint is identical to that for the present setpoint. Therefore, it is concluded that the proposed trip setpoint is acceptable.

Based on our review of the licensee's submittal, we conclude that the proposed change to the technical specification - high pressurizer pressure trip setpoint of $\leq$ 2362 psia and allowable value of $\leq$ 2370.887 psia is acceptable.

## 3.0 TECHNICAL SPECIFICATION CHANGES

### 3.1 THERMAL MARGIN LIMITS

The Technical Specifications are modified to reflect the changeover from the W-3 DNBR correlation to the CE-1 DNBR correlation in conjunction with the statistical combination of uncertainties (SCU) methodology. The basic DNBR limit is changed from 1.30 (W-3) to 1.24 (CE-1/SCU).

The pages affected for the DNBR limit change from 1.30 to 1.24 are: 2-1, 2-6, B 2-1, B 2-2, B 2-6, B 3/4 2-3, B 3/4 4-1.

A change related to the change in DNBR correlations discussed above is the inclusion of limiting values on the addressable constant BERR 1, the power uncertainty factor used in the DNBR calculation, in the TS. The page affected is 2-6.

The new DNBR limit and BERR 1 values have been found acceptable in Section 2.3 of this report for the conditions of operation authorized by the related license amendment. The staff's evaluation of the DNBR limit and BERR 1 value for full power operation will be addressed further in a forthcoming Safety Evaluation accompanying an amendment authorizing full power operation.

### 3.2 PEAKING FACTOR DEFINITIONS

A definition of the planar radial peaking factor, $F_{xy}$, has been added to standardize the ANO-2 definition and symbol with other CE plants. The value of 1.053 for $F_{xy}$ is documented in a report which has been reviewed and approved by the NRC staff. The acceptability of these changes is discussed in Section 2.2.2 of the SE. Further details may be found in the licensee's letter dated May 11, 1981 response to question four. Since the 5.3 percent uncertainty value has been reviewed and found acceptable these changes should be made. The pages affected are: 1-6, 3/4 2-4, B 3/4 2-1, B 3/4 2-2, and B 3/4 2-3.

### 3.3 DNB RELATED PARAMETER LIMITS

The previous TS 3.2.6 on core average temperature is consistent with Standard Technical Specification requirements for non-CPCS CE plants. However, for the ANO-2 DNB related safety analyses, core average temperature is not an input parameter. The relevant parameters for the ANO-2 analyses are reactor coolant cold leg temperature, axial shape index and pressurizer pressure. Accordingly, the licensee has proposed TS limits on these values consistent with the values assumed in the ANO-2 Cycle 2 safety analyses. Since, as reported in Section 2.0 of this report these safety analyses have been reviewed and approved for Cycle 2, the proposed TS changes should be made. The affected pages are: 3/4 2-12, 3/4 2-13, 3/4 10-2, 3/4 2-14 and B 3/4 2-4.

## 3.4 PARTIAL PUMP OPERATION

Partial pump operation in MODES 1 and 2 has not been allowed in Cycle 1 and is not allowed in Cycle 2 since the licensee has not submitted for the Commission's review and approval the safety analyses supporting such operations. However the licensee proposes certain changes to the TS. to clarify this situation. These changes include the change of the term "ECCS" to "Safety" in various footnotes to reflect that analyses must be submitted not only for ECCS performance but for transients and other accidents as well. In addition clarifying language is added to other TS. These changes are editorial in nature, do not affect safety, and are acceptable. The affected pages are: 2-5, 3/4 4-1, 3/4 4-2 and 3/4 7-3. In addition TS 3.4.1 (page 3/4 4-2) has been subdivided to provide a separate ACTION for MODE 3 to ensure that reactor coolant pump operation in MODE 3 is consistent with the assumptions made in the main steamline break analysis. This change provides consistency between the TS and the MSLB safety analysis and is, therefore, acceptable.

## 3.5 AVAILABILITY OF BORATED WATER FROM RWT MODES 1, 2, 3 AND 4

Because of the higher core average enrichment and an increase in the available shutdown margin requirements the licensee proposes to increase the refueling water tank (RWT) required volume from 40,200 gallons at 1731 ppm to 56,455 gallons of 1731 ppm borated water. The proposed value was considered in the Cycle 2 safety analyses. Since as stated in Section 2.0 of this report, these safety analyses have been reviewed and found acceptable the proposed TS change should be made. The page affected is B 3/4 1-2.

## MODES 5 AND 6

Because of the increased shutdown margin requirements the licensee proposes to increase the RWT required volume from 4,700 gallons at 1731 ppm to 8185 gallons of 1731 ppm borated water. The proposed value was considered in the Cycle 2 safety analyses. Since, as stated in Section 2.0 of this report, these safety analyses have been reviewed and found acceptable the proposed TS changes should be made.

A typographical error is also corrected to make the BASES consistent with TS 3.1.2.7. The page affected is B 3/4 1-3.

## 3.6 SHUTDOWN MARGIN FOR MODE 5

The shutdown margin was evaluated for a boron dilution event during the cold shutdown condition. It was determined by the licensee that a 5% $\Delta K/K$ shutdown margin would be required so that at least 15 minutes would be available to the operator in order to terminate the deboration transient. We find this

proposed TS change acceptable. The pages affected are: 3/4 1-3, 3/4 1-8, 3/4 1-10, 3/4 1-12, 3/4 1-15, B 3/4 1-1, B 3/4 1-2 and B 3/4 1-3.

3.7 RPS AND ESFAS TRIP SETPOINTS

(a) The licensee proposes to change the value of certain RPS and ESFAS trip setpoints. The specific parameters are Linear Power Level-High, Pressurizer Pressure-Low, Steam Generator Pressure Low, Steam Generator Level-Low, Steam Generator Level-High, and Steam Generator P-High. The acceptability of these changes is addressed in Section 2.6.2 of this report. The pages affected are: 2-5, 2-6, 3/4 3-16, 3/4 3-17 3/4 3-18.

(b) The licensee also proposes to correct a typographical error in the refueling water tank level minimum allowable value from 5.300% indicated level to 5.111% of indicated level. The acceptability of this change is addressed in the LLL report referenced in Section 2.6.2 of this report. The page affected is 3/4 3-17.

(c) The licensee proposes to change the Pressurizer Pressure-High setpoint from 2345 psi to 2362 psi. The acceptability of this change is addressed in Section 2.72 of this report. The page affected is 2-5.

The changes discussed above in a, b and c are considered in the Cycle 2 safety analyses. Since we have reviewed and approved these safety analyses the proposed TS changes should be made.

3.8 CPCS ADDRESSABLE CONSTANTS

The licensee has proposed TS to control modifications to addressable constants. The acceptability of these changes is addressed in Section 2.5.2 of this report. The pages affected are: 2-4, 2-7, 2-8, 2-9, B 2-7, 3/4 3-8, 3/4 3-9, and 6-13.

3.9 CPCS ROOM HIGH TEMPERATURE

The licensee has proposed TS to verify the OPERABILITY of the CPCS in event of a valid high CPCS room temperature alarm. The acceptability of this TS is discussed in Section 2.5.4 of this report. The page affected is 3/4 3-1a.

3.10 RPS AND ESFAS TRIP LIMIT TABLE FOOTNOTES

In the footnotes to these tables the licensee proposes to change the terms "as pressurizer pressure is reduced" and "as steam generator pressure is reduced" to "during a planned reduction in pressurizer pressure" and "during a planned reduction in steam generator pressure" respectively. This change properly limits the manual reduction by the operator of the setpoint to occasions of controlled and planned reductions in pressure and is acceptable. The affected pages are: 2-6, 3/4 3-18.

3.11 MODERATION TEMPERATURE COEFFICIENT (MTC)

The acceptability of the MTC in TS 3.1.1.4 is supported by the discussion in Section 2.2.1 of this report. The affected page is 3/4 1-5.

3.12 STEAM GENERATOR LEVEL-LOW TRIP

The licensee proposes to change the BASES wording to reflect the fact that the ANO-2 emergency feedwater system is actuated and supplies water to the steam generators automatically upon receipt of an ESFAS versus being required to be manually actuated within a ten minute period. Since actuation and feed is automatic upon demand there is no basis for the ten minute period in the ANO-2 safety analyses. The change is acceptable. The affected page is B 2-5.

3.13 CEA INSERTION LIMITS

TS Figure 3.1-2 has been changed to reflect the provision of adequate shutdown margin for MODES 1 and 2. The acceptability of the shutdown margin is discussed in Section 2.2.1 of this report. The affected page is 3/4 1-27.

3.14 LINEAR HEAT RATE MARGIN

The licensee has proposed an additional TS Figure to provide further definition of the acceptable operating limits for the conditions of COLSS in-service and COLSS out-of-service. The previous TS did not include a figure defining the limits for these two conditions. Based on the licensee's response to items 25 and 26 in their May 6, 1981 letter the change provides clarification and does not change safety margins. Therefore the change is acceptable. The pages affected are: 3/4 2-1, 3/4 2-2, and 3/4 2-3.

3.15 DNBR OPERATING LIMIT

The licensee proposes to change Figure 3.2-4 due to Cycle 2 reanalysis of COLSS out-of-service DNBR margin requirements. These limits are reflected in the determination of the initial conditions for Cycle 2 anticipated operational occurrences which we have evaluated and found acceptable in Section 2.0 of this report. Therefore the proposed change is acceptable. The affected pages are: 3/4 2-7, 3/4 2-9, 3/4 2-10 and B 3/4 2-1.

3.16 RPS AND ESFAS APPLICABLE MODE AND ACTION NOTES

The licensee proposes to delete MODE 1 from the Table 3.3-1, page 3/4 3-2, Functional Unit 3a APPLICABLE MODES column. This trip which would occur at 0.75% power is bypassed before reaching MODE 1 and is not applicable to MODE 1 operations. Therefore, its deletion is acceptable.

The licensee proposes to add additional modes of applicability for pressurizer pressure-low and steam generator pressure-low trip setpoints to ensure acceptability of the main steamline break analyses. We have found acceptable the MSLB analyses as stated in Section 2.0 of this report. Therefore these changes should be made. The affected pages are: 3/4 3-2, and 3/4 3-7.

The licensee also proposes to add the provision that TS 3.04 is not applicable to the CEAC's in Table 3.3-1. This has previously been the case for the CPC's. With the proposed change the requirements for the CEAC's are consistent with previously approved requirements for the CPC's. We find this acceptable. The affected page is 3/4 3-3.

3.17 MARGINS WITH CEAC'S INOPERABLE

The licensee proposes to increase the required margins of TS 3.2.1 and 3.2.4 with COLSS out-of-service from greater than or equal to 8 percent to greater than or equal to 11 percent based on the Cycle 2 reanalysis of the CEAC inoperable margin requirements. This increase in the margin provided is acceptable. The affected page is 3/4 3-5a.

3.18 CONTAINMENT PRESSURE, TEMPERATURE AND HUMIDITY

By letter dated November 19, 1979 proposed additional limits on the acceptable combinations of containment pressure and temperature to be included in TS Figure 3.6-1. The acceptability of this change is addressed in Section 2.7.1 of this change is addressed in Section 2.7.1 of this report. The affected page is 3/4 6-7.

3.19 REFUELING MACHINE

The licensee proposes a change to TS 3.9.6 to delete CEAs from items to be moved by the refueling machine. The licensee states that the refueling machine does not include provisions for moving CEAs and notes that CEAs are instead moved with a manual tool or with a CEA change mechanism located over the fuel transfer mechanism upender. Therefore this change in the TS is required to reflect the actual design capabilities of the refueling machine and the handling practices of the licensee and is acceptable. The affected pages are: 3/4 9-7 and B 3/4 9-2.

## 3.20 PRESSURIZER SAFETY VALVE CAPABILITY

The licensee proposes a TS change to the BASES for pressurizer safety valve testing to amend the designated relief capability of 395,000 lb m/hr to 420,000 lb m/hr. This change is made to update the preliminary design valve (395,000) to the actual rated valve (420,000) and is acceptable. The page affected is B 3/4 4-1.

## 3.21 MISCELLANEOUS CHANGES

Various TS pages are changed due to a change in the page number due to other new pages being added, due to correction of typographical errors and other changes of an administrative non-safety related nature. These changes are acceptable. The affected pages are: 3/4 2-5, 3/4 2-6, 3/4 2-7, 3/4 2-8, 3/4 2-9, 3/4 2-11, 3/4 6-18 and 5-5.

## 4.0 PHYSICS TESTING

The startup physics test program as outlined by the licensee was reviewed. The precritical tests include control element assembly trip tests and reactor coolant flow coastdown tests. The low power tests include critical boron concentration, CEA symmetry, and temperature reactivity worth tests. Power escalation tests include core power distribution tests at 50 percent and 100 percent power and isothermal temperature coefficient and power coefficient tests at 50 and 100 percent power. The acceptance criteria supplied for each test was reviewed as well as the procedures if acceptance criteria were not met. We find this physics startup test program acceptable.

## 5.0 ENVIRONMENTAL CONSIDERATION

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

## 6.0 CONCLUSION

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

Date: June 19, 1981

## 7.0 REFERENCES

### 1.0 Introduction

There are two basic references which comprise the ANO-2 Cycle 2 Reload Report these being the submittals from the licensee dated February 20 and March 5, 1981. However numerous other documents were generated by the staff and by the licensee in support of the reload review and other issues addressed in this Safety Evaluation. Therefore, for convenience's sake on the following pages the references are listed under the SE section to which they apply..

## 2.1 Fuel Design


1. Letter from D. Trimble (AP&LCo) to R. A. Clark (NRC), Subject: Cycle 2 Reload Report, dated February 20, 1981.

2. Letter from W. Cavanaugh, III, (AP&LCo) to R. A. Clark (NRC), Subject: Cycle 2 Reload Report, dated March 5, 1981.

3. Letter from D. Trimble (AP&LCo) to R. A. Clark (NRC), Subject: Information Regarding ANO-2 Reload Report, dated April 30, 1981.

4. "The Evaluation and Demonstration of Methods for Improved Nuclear Fuel Utilization First Semi-Annual Progress Report: Inception to June 30, 1980," C-E draft report CENPD-384, October 1980.

5. "Test Fuel Rod Irradiation: 16X16 Nuclear Reactor," C-E report CENPD-256-P-A, August 1977.

6. "Fuel Evaluation Model," C-E report CENPD-139-A, July 1974.

7. Letter from O. D. Parr (NRC) to F. M. Stern (C-E), dated December 4, 1974.

8. Letter from D. F. Ross, Jr., (NRC) to A. E. Scherer (C-E), dated November 23, 1976.

9. Letter from R. S. Boyd (NRC) to W. Cavanaugh, III, (AP&LCo), Subject: Issuance of Amendment No. 1 to Facility Operating License NPF-6 (ANO-2), dated September 1, 1978.

10. Letter from D. Trimble (AP&LCo) to R. A. Clark (NRC), Subject: Responses to NRC Questions on ANO-2 Reload, dated May 6, 1981.

11. "CEPAN Method of Analyzing Creep Collapse of Oval Cladding," C-E report CENPD-187-A, March 1976.

12. "Fuel and Poison Rod Bowing," C-E report CENPD-225, Supplement 3-P, June 1979.

13. Memorandum from D. F. Ross, Jr., and D. G. Eisenhut (NRC) to D. B. Vassallo and K. R. Goller, "Interim Safety Evaluation Report on the Effects of Fuel Rod Bowing in Thermal Margin Calculations for Light Water Reactors," dated December 8, 1976.

14. Memorandum from D. F. Ross, Jr., and D. G. Eisenhut (NRC) to D. B. Vassallo and K. R. Goller, "Revised Interim Safety Evaluation Report on the Effects of Fuel Rod Bowing in Thermal Margin Calculations for Light Water Reactors," dated February 16, 1977.

15. Letter from R. A. Clark (NRC) to W. Cavanaugh, III, (AP&LCo), dated April 10, 1981.

16. "Safety Evaluation Report related to the operation of Arkansas Nuclear One, Unit 2," Section 4.4, NRC report NUREG-0308, Supplement 1, June 1978.

17. "Zircaloy Growth In-Reactor Dimensional Changes in Zircaloy-4 Fuel Assemblies," C-E report CENPD-198, December 1975.

18. "Zircaloy Growth Application of Zircaloy Irradiation Growth Correlations for the Calculation of Fuel Assembly and Fuel Rod Growth Allowances," C-E report CENPD-198, Supplement 1, Decmeber 1977.

19. "Response to Request for Additional Information on CENPD-198-P, Supplement 1," C-E report CENPD-198, Supplement 2-P, November 1, 1978.

20. Letter from R. L. Baer (NRC) to A. E. Scherer (C-E), dated August 21, 1979.

21. Letter from K. Kniel(NRC) to A. E. Scherer (C-E), dated June 22, 1976.

22. PNO-77-221, preliminary notification of event on unusual occurrence of guide tube wear, December 14, 1977.

23. Letter from A. E. Scherer (C-E) to V. Stello (NRC), dated December 23, 1977.

24. Letter from W. Johnson (MYAPCo) to V. Stello (NRC), dated February 14, 1978.

25. Letter from A. E. Lundvall, Jr., (BG&ECo) to V. Stello (NRC), dated February 17, 1978.

26. "Safety Evaluation Report related to the operation of Arkansas Nuclear One, Unit 2," Section 4.2, Supplement 2, September 1978.

27. Letter from D. Trimble (AP&LCo) to R. A. Clark (NRC), Subject: EOC-1 CEA Guide Tube Surveillance Program, dated March 30, 1981.

28. Letter from D. Trimble (AP&LCo) to R. A. Clark (NRC), Subject: Preliminary Results of ANO-2 Fuel Inspection, dated May 22, 1981.

29. Letter from D. Trimble (AP&LCo) to R. A. Clark (NRC), Subject: NRC Request for Information on Fuel Assembly Spacer Grid Damage, dated June 4, 1981.

2.2 Nuclear Analysis

1. Letter from R. A. Clark (NRC) to W. Cavanaugh, III, (AP&LCo) dated March 23, 1981 transmitting six physics questions.

2. Letter from D. C. Trimble (AP&LCo) to Director, NRR dated April 14, 1981 transmitting responses to staff's March 23, 1981 questions.

3. Letter from R. A. Clark (NRC) to W. Cavanaugh, III, (AP&LCo) dated April 29, 1981 transmitting five additional physics questions.

4. Letter from D. C. Trimble (AP&LCo) to Director, NRR dated May 11, 1981 transmitting responses to the staff's April 29, 1981 questions.

5. Letter from D. C. Trimble (AP&LCo) to Director, NRR dated May 27, 1981 transmitting information on fuel assembly misloading analyses.

2.5 Reactor Protection System

1. Letter from D. C. Trimble (AP&LCo) to Director, NRR dated September 3, 1980, CPC/CEAC - Plant Computer Datalink.

2. Letter from D. C. Trimble (AP&LCo) to Director, NRR dated February 28, 1981, CPC/CEAC - Plant Computer Datalink.

3. Letter from R. A. Clark (NRC) to W. Cavanaugh, III, (AP&LCo) dated April 10, 1981, Part II - Instrumentation and Controls System.

4. Letter from R. A. Clark (NRC) to W. Cavanaugh, III, (AP&LCo) dated May 5, 1981 requesting documentation of addressable constants modification procedures.

5. Letter from W. Cavanaugh, III, (AP&LCo) to Director, NRR dated May 19, 1981 responding to Part II of staff's April 10, 1981 letter.

6. Letter from D. C. Trimble (AP&LCo) to Director, NRR dated May 26, 1981 providing a document in response to staff's May 5, 1981 letter.

## 2.6 License Conditions

1. Letter from D. A. Reuter (AP&LCo) to J. F. Stolz (NRC) dated October 11, 1977 proposing a design for long term overpressure protection equipment.

2. Letter from D. C. Trimble (AP&LCo) to Director, NRR dated December 1, 1980 responding to the license condition and the SER open items.

3. Letter from D. C. Trimble (AP&LCo) to Director, NRR dated March 30, 1981 responding to staff questions.

## 2.7 Other Matters

1. Letter from W. Cavanaugh, III, (AP&LCo) to Director, NRR, dated November 19, 1979 requesting change to TS Figure 3.6-1 on containment pressure, temperature and relative humidity.

2. Letter from W. Cavanaugh, III, (AP&LCo) to Director, NRR dated November 27, 1979 requesting change to TS Table 2.2-1 on high pressurizer trip setpoints.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-368

ARKANSAS POWER AND LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 24 to Facility Operating License No. NPF-6, issued to the Arkansas Power and Light Company, which revised the license and the Technical Specifications for operation of Arkansas Nuclear One, Unit No. 2 (the facility) at steady state reactor core power levels not in excess of 2815 megawatts thermal, in accordance with the provisions of the license and the Technical Specifications. However, the facility is temporarily restricted from operating at full rated power pending completion of the staff's detailed review of the core protection calculator system (CPCS) changes for Cycle 2 operation. The facility is located at the licensee's site in Pope County, Arkansas. The license amendment is effective as of its date of issuance.

The amendment authorizes Cycle 2 operation at seventy (70) percent of the licensed power level of 2815 MWt with:

- Changes in the Core Protection Calculator System (CPCS) to reflect utilization of the CE-1 critical heat flux correlation and associated thermal hydraulic methodology.
- Changes in the CPCS to reflect utilization of the Statistical Combination of Uncertainties (SCU) thermal hydraulic methodology for the combination of system parameter uncertainties.
- Changes in the RPS and ESFAS trip setpoints to reflect a change in signal transmitter design and to reflect staff approval of the licensee's equipment trip setpoints.

- Changes in the minimum required shutdown margin to lengthen the time available for operator action during a boron dilution event.
- Changes required to maintain acceptable results for the steamline break analysis.
- Some demonstration fuel assemblies to test new fuel designs.
- Numerous other miscellaneous changes of a clarifying, editorial and administrative nature.
- Other changes in the Technical Specification to incorporate requirements resulting from the detailed physics and thermal hydraulic analysis of the Cycle 2 reload core.

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

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

For further details with respect to this action, see (1) the applications for amendment dated February 20 and March 5, 1981, as supplemented by references identified in the related Safety Evaluation, (2) Amendment No. 24

to License No. NPF-6 and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Arkansas Tech University, Russellville, Arkansas 72801. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland this 19th day of June, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

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