PROPOSED TECHNICAL SPECIFICATIONS

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
9. Local Power Density - High	$\leq$21.0 kw/ft (5)	$\leq$21.0 kw/ft (5)
10 DNBR - Low	$\geq$1.25 (5)	$\geq$1.25 (5)
11. Steam Generator Level - High	$\leq$93.7% (4)	$\leq$94.589% (4)

TABLE NOTATION

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

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

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

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

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

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

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

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

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

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

2.2.1 REACTOR TRIP SETPOINTS

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

The DNBR - Low and Local Power Density - High are digitally generated trip setpoints based on Limiting Safety System Settings of 1.25 and 21.0 kw/ft, respectively. Since these trips are digitally generated by the Core Protection Calculators, the trip values are not subject to drifts common to trips generated by analog type equipment. The Allowable Values for these trips are therefore the same as the Trip Setpoints.

To maintain the margins of safety assumed in the safety analyses, the calculations of the trip variables for the DNBR - Low and Local Power Density - High trips include the measurement, calculational and processor uncertainties and dynamic allowances as defined in CEN 305-P, "Functional Design Requirement for a Core Protection Calculator," July 1985; CEN-304-P, "Functional Design Requirements for a Control Element Assembly Calculator," July 1985; CEN-310-P, "CPC and Methodology Changes for the CPC Improvement Program," October 1985 and CEN-308-P, "CPC/CEAC Software Modifications for the CPC Improvement Program," August 1985.

Manual Reactor Trip

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

Linear Power Level-High

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

Logarithmic Power Level-High

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

DNBR - Low

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

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

b. Reactor Coolant System pressure from pressurizer pressure measurement;

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

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

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

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

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

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

BASES

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

Steam Generator Level-High

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

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 MSS-NA2-P, "Arkansas Nuclear One-Unit 2 Core Protection Calculator Addressable Constant Determination Methodology" dated August 1981.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 LINEAR HEAT RATE

LIMITING CONDITION FOR OPERATION

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

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

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

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER

ACTION:

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

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

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

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

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

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

SURVEILLANCE REQUIREMENTS

4.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.1.2 The linear heat rate shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the linear heat rate, as indicated on any OPERABLE CPC channel, is within the limit shown on Figure 3.2-1.

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

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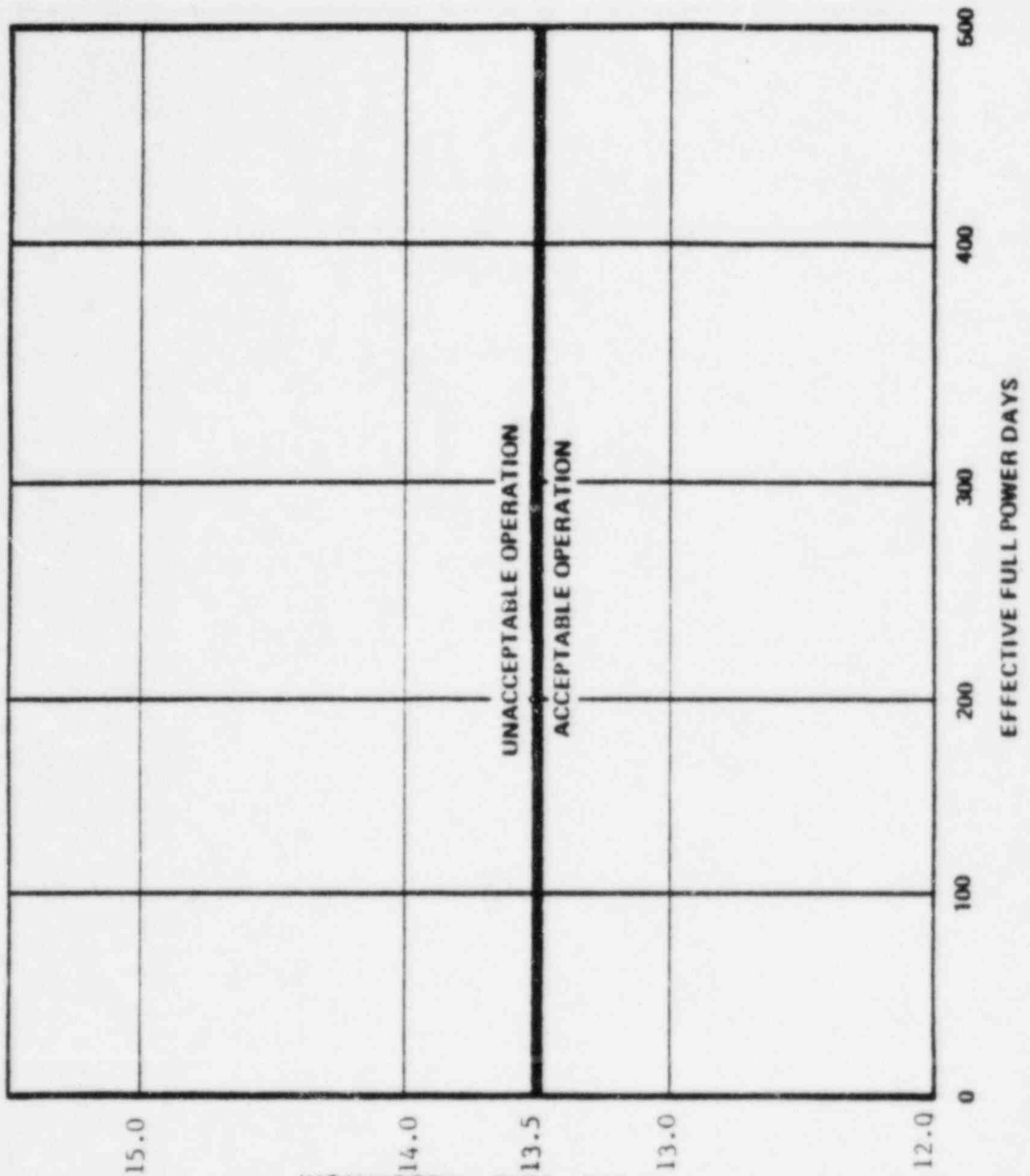

Figure 3.2-1 Allowable Peak Linear Heat Rate vs Burnup
(COLSS out of service)

3/4.2.4 DNBR MARGIN

LIMITING CONDITIONS FOR OPERATION

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

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

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

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

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

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

ACTION:

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

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

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

within 15 minutes initiate corrective action to restore the DNBR to within the limits, and either:

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

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

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS (continued)

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

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

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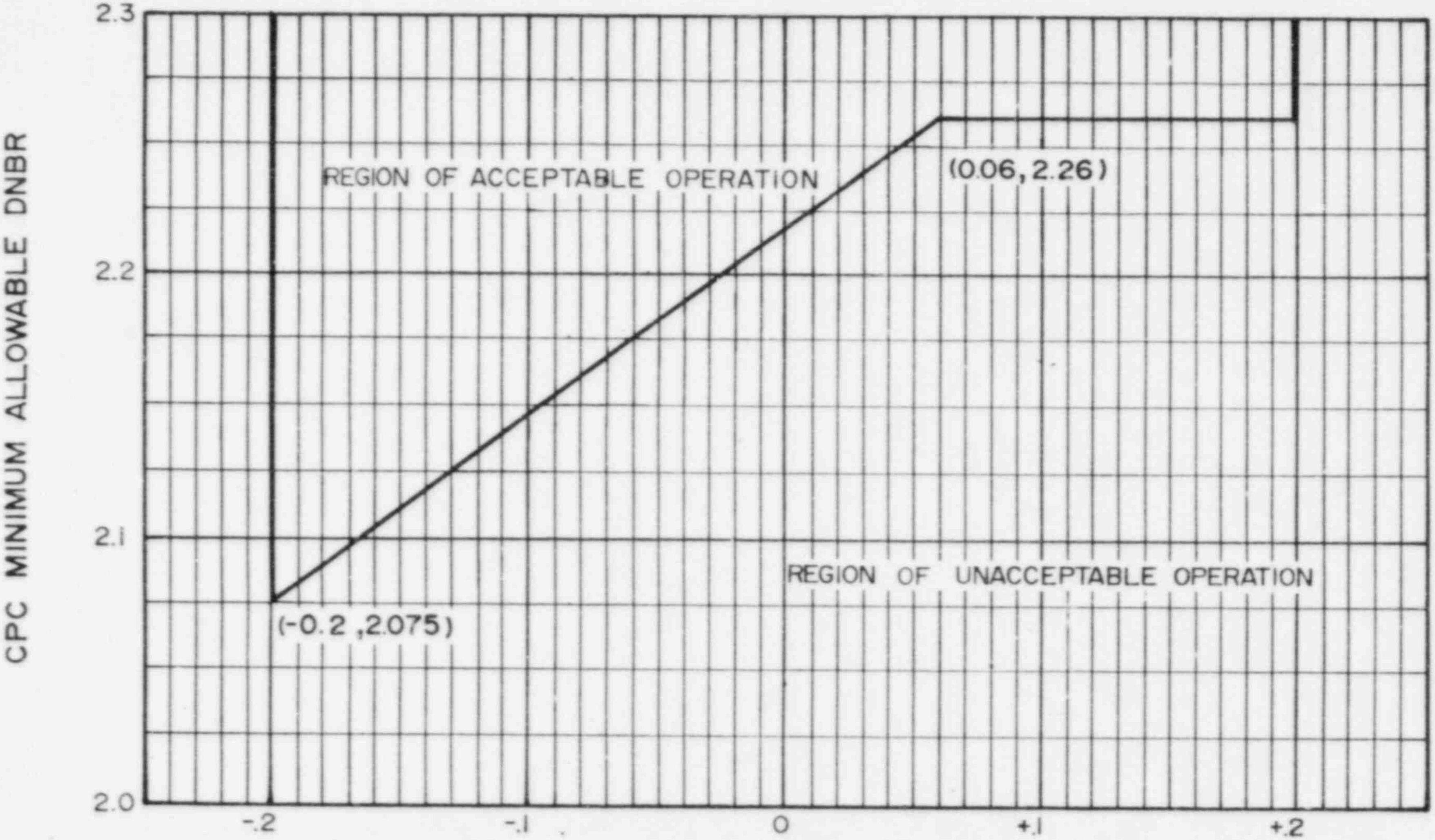
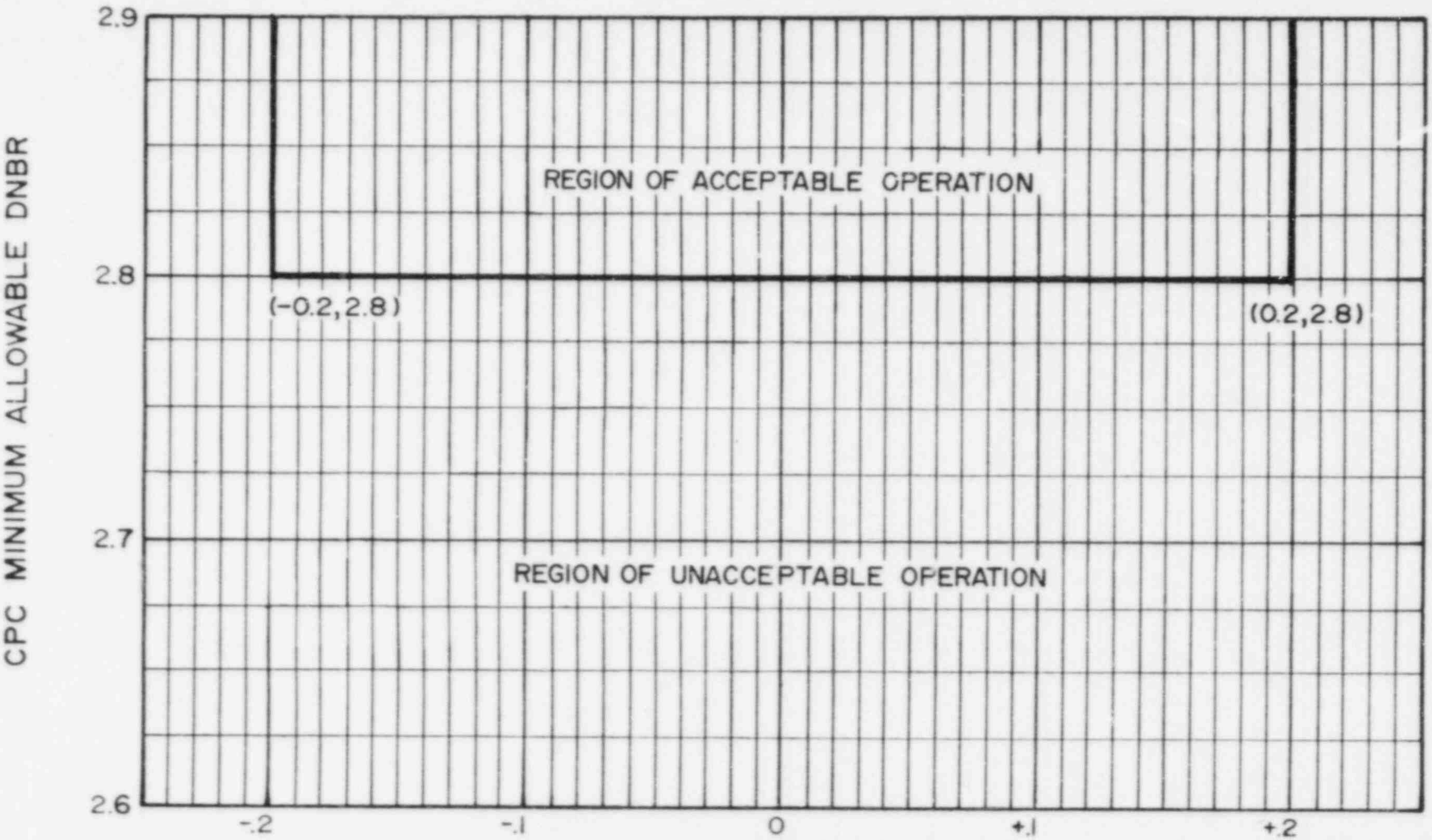


CORE AVERAGE AXIAL SHAPE INDEX

FIGURE 3.2-2

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



CORE AVERAGE AXIAL SHAPE INDEX

FIGURE 3.2-3

DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS

(COLSS OUT OF SERVICE, BOTH CEACS INOPERABLE)

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

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

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

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

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

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

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

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

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

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

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

   2. Within 4 hours:

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

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

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

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

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

TABLE 3.3-2 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

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

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

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

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

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

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

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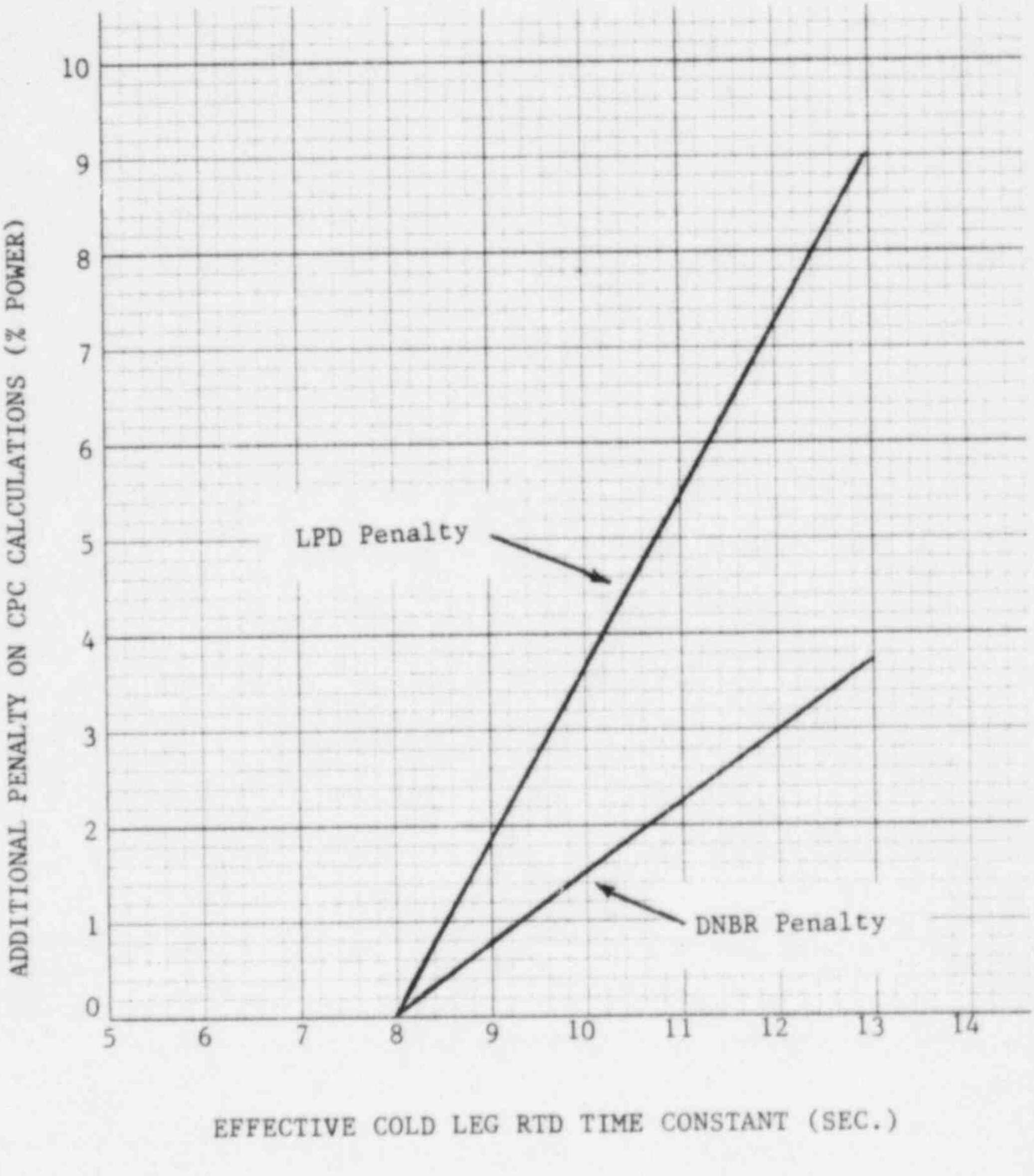
FIGURE 3.3-1

CPC PENALTY VS. EFFECTIVE RTD TIME CONSTANT



3/4.2 POWER DISTRIBUTION LIMITS

BASES

3/4.2.1 LINEAR HEAT RATE

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

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

The COLSS calculated core power and the COLSS calculated core power operating limits based on linear heat rate are continuously monitored and displayed to the operator. A COLSS alarm is annunciated in the event that the core power exceeds the core power operating limit. This provides adequate margin to the linear heat rate operating limit for normal steady state operation. Normal reactor power transients or equipment failures which do not require a reactor trip may result in this core power operating limit being exceeded. In the event this occurs, COLSS alarms will be annunciated. If the event which causes the COLSS limit to be exceeded results in conditions which approach the core safety limits, a reactor trip will be initiated by the Reactor Protective Instrumentation. The COLSS calculation of the linear heat rate limit includes appropriate uncertainty and penalty factors necessary to provide a 95/95 confidence level that the maximum linear heat rate calculated by COLSS is greater than or equal to that existing in the core. To ensure that the design margin to safety is maintained, the COLSS computer program includes an F_{xy} measurement uncertainty factor of 1.053, an engineering uncertainty factor of 1.03, a THERMAL POWER measurement uncertainty factor of 1.02 and appropriate uncertainty and penalty factors for rod bow.

Parameters required to maintain the operating limit power level based on linear heat rate, margin to DNB and total core power are also monitored by the CPCs. Therefore, in the event that the COLSS is not being used, operation within the limits of Figure 3.2-1 can be maintained by utilizing a predetermined local power density margin and a total core power limit in the CPC trip channels. The above listed uncertainty and penalty factors are also included in the CPCs.

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

3/4.2.4 DNBR MARGIN

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

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the DNBR channels in the Core Protection Calculators (CPCs), provide adequate monitoring of the core power distribution and are capable of verifying that the DNBR does not violate its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core operating limit corresponding to the allowable minimum DNBR. The COLSS calculation of core power operating limit based on DNBR includes appropriate uncertainity and penalty factors necessary to provide a 95/95 confidence level that the core power at which a DNBR of less than 1.25 could occur, as calculated by COLSS, is less than or equal to that which would actually be required in the core. To ensure that the design margin to safety is maintained, the COLSS computer program includes an F_{xy} measurement uncertainty factor of 1.053, an engineering uncertainty factor of 1.03, a THERMAL POWER measurement uncertainty factor of 1.02 and appropriate uncertainty and penalty factors for rod bow.

Parameters required to maintain the margin to DNB and total core power are also monitored by the CPCs. Therefore, in the event that the COLSS is not being used, operation within the limits of Figure 3.2-2 for CEAC operable or Figure 3.2-3 for both CEACs inoperable can be maintained by utilizing a predetermined DNBR as a function of AXIAL SHAPE INDEX and by monitoring the CPC trip channels. The above listed uncertainty and penalty factors are also included in the CPC.

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

3/4.3 INSTRUMENTATION

BASES

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

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

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

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

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

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

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

CPC IMPROVEMENT PROGRAM TECHNICAL SPECIFICATION CHANGES

	Section No.	Nature of Changes	Reason for Change
1.	2.2.1 Table 2.2-1	LPD trip setpoint = 21.0 kw/ft	CIP will use generic LPD trip setpoint with adjustments, if necessary, via addressable constants.
	Basis	Update references	CPCS modification.
		Modify CPC range limits	CIP specifies generic range limits.
2.	3/4.2.1 & Basis	Revise format	Provide clear and consistent COLSS and COLSS-out-of-service monitoring, action and surveillance requirements.
3.	3/4.2.4 & Basis 3.3.1 Table 3.3-1 Actions 5.a & b	Revise format	Provide clear and consistent COLSS, COLSS-out-of-service and CEAC inoperable monitoring, action and surveillance requirements.
		Update figures for COLSS out of service monitoring	Consistent with revised format and CIP methodology and results.
4.	3.3.1 Tables 3.3-2 and 3.3-3	Revised note to change RTD response time to 8 seconds for cold leg, 13 seconds for hot leg.	CIP dynamic compensation algorithms and constants assume 8 seconds for cold leg, 13 seconds for hot leg.
		Revise Figure 3.3-1 and Table 3.3-3.	Consistent with CPC revision and transient analysis assumptions and results.

DESCRIPTION OF AMENDMENT REQUEST

1. This proposed change would revise Technical Specification 2.2.1, "Reactor Trip Setpoints," and its associated Bases. Table 2.2-1, "Reactor Protective Instrumentation Trip Setpoint Limits," requires that the setpoints for trip values of the Reactor Protective System (RPS) be set at specified values and kept within a specified allowable value range. The Local Power Density--High Trip Setpoint of Table 2.2.1 specifies the required trip setpoint which is the value at which the Core Protection Calculator System (CPCS) acts to prevent the peak linear heat rate from exceeding its safety limit for transients and anticipated operational occurrences and to mitigate the consequences of accidents. This proposed change revises the value of the Local Power Density--High Trip Setpoint and Allowable Value of the Limiting Safety System Setting (LSSS). Specifically, Table 2.2-1 Functional Unit 9, currently requires that both the Trip Setpoint and Allowable Value for the Local Power Density--High Trip be 20.3 kw/ft. This setpoint has been derived by reducing the actual Local Power Density Specified Acceptable Fuel Design Limit (SAFDL) by a certain amount which accounts for thermal dynamic effects in the CPCS calculations. The revised setpoint no longer includes these dynamic allowances; instead, these effects are adjusted for using the addressable constants. The proposed change increases both the Trip Setpoint and Allowable Value to 21.0 kw/ft, and reflects this revised setpoint in the Bases. This is a generic setpoint for the CPC Improvement Program (CIP) and reflects an improved methodology and simplification of adjusting for dynamic effects on a cycle dependent basis. The effective CPCS trip setpoint remains the same. Effectively, the CPCS local power density protection is not being changed.

General Design Criterion 10, Reactor Design, requires that the reactor core and associated coolant, control and protection systems be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. The specified trip settings result in confidence that the SAFDLs will not be exceeded during normal operation or as the result of anticipated operational occurrences.

In the Bases for Technical Specification 2.2.1, the references will be updated to reflect the appropriate CIP modifications and methodology documents. Also, the CPC range limits will be modified to generic range limits specified by the CIP. The RCS Cold Leg Temperature--Low limit will change from 465 to 490 °F and the High limit will change from 605 to 585 °F. The Pressurizer Pressure--Low limit will change from 1750 to 1785 psia and the High limit will change from 2400 to 2415 psia. These generally more restrictive limits will reduce the range of inputs (and therefore the calculational complexity) of the safety analyses and promote consistency between the software of CPCS-equipped plants.

2. This proposed change will revise the format of Technical Specification 3/4.2.1, "Linear Heat Rate," and the associated Bases, to provide clear and consistent monitoring, action and surveillance requirements when the Core Operating Limits Supervisory System (COLSS) is available or out of service. Technical Specification 3/4.2.1, "Linear Heat Rate" requires that the linear heat rate limit be maintained by operating within the region of acceptable operation as indicated by either the COLSS or the CPC.

The proposed change to Technical Specification 3.2.1 replaces the existing Limiting Condition for Operation (LCO) with two parts, 3.2.1.a and 3.2.1.b. 3.2.1.a states that when COLSS is in service, the COLSS calculated core power must be maintained less than or equal to the COLSS calculated Power Operating Limit (POL) based on linear heat rate. 3.2.1.b states that when COLSS is out of service, the linear heat rate limit is maintained by operating within the region of acceptable operation of the new Figure 3.2-1 using any operable CPC channel. Also, Surveillance Requirement 4.2.1.2 is revised to allow use of any operable CPC channel for monitoring the linear heat rate limit with COLSS out of service. This proposed change recognizes that it is acceptable to monitor any one channel for control purposes during steady state operation. It is not necessary that the monitored channel be the most limiting since appropriate uncertainty allowances are already implemented in the CPCS calculations and trip setpoints. The CPCS continues to provide the required protection during transient operation. The old Figure 3.2-1 will be deleted and replaced with words to the same effect. Figure 3.2-2 will become the new Figure 3.2-1. The linear heat rate limit value of Figure 3.2-1 will be lowered from 14.5 kw/ft to 13.5 kw/ft due to the removal of the flux peaking augumentation factors (in accordance with prior NRC approval for CE plants as described in the SER related to Calvert Cliffs, Docket No. 50-317, Technical Specification Amendment No. 104) and to accomodate longer fuel cycles. No new data has been developed which would require further analysis of clad collapse for ANO-2. The Combustion Engineering (CE) fuel rod manufacturing process has not changed in any way that would adversely affect the present clad collapse and augmentation factor analysis results. Therefore, flux peaking augmentation factors may be removed from the calculation of linear heat rate since the maximum potential augmentation factor is insignificant with respect to other power distribution uncertainties. The proposed change to Technical Specification 3.2.1 also modifies the existing action statement to be more clear and consistent with the LCO as described above.

3. This proposed change would revise Technical Specifications 3/4.2.4, "DNBR Margin," 3/4.3.1, "Reactor Protective Instrumentation," and the associated Bases, to provide clear and consistent monitoring, action and surveillance requirements for the various conditions with COLSS in service/ out of service and Control Element Assembly Calculators (CEACs) operable/ inoperable. Technical Specification 3/4.2.4 requires that the departure from nucleate boiling ratio (DNBR) margin be maintained by operating within the region of acceptable operation as indicated by either the COLSS or the CPC. Technical Specification 3/4.3.1 requires that the Reactor Protective Instrumentation System (RPIS) be operable and defines the number and type of RPIS channels required, response times and periodic testing required to assure operability and actions to be taken when the required RPIS is out of service. This proposed change consists of the following two parts:

a. The proposed change to Technical Specification 3.2.4 replaces the existing LCO with four parts, 3.2.4.a through 3.2.4.d. 3.2.4.a states that when COLSS is in service and at least one CEAC is operable, the COLSS calculated core power must be maintained less than or equal to the COLSS calculated POL based on DNBR. This is consistent with the requirements previously presented graphically by Figure 3.2-3. The existing Figure 3.2-3 will thus be deleted and replaced with words to the same effect. 3.2.4.b states that when COLSS is in service and neither CEAC is operable, the COLSS calculated core power must be maintained less than or equal to the COLSS calculated POL based on DNBR decreased by a penalty factor of 13.0% of rated power.

3.2.4.c states that when COLSS is out of service and at least one CEAC is operable, CPC calculated DNBR on any operable channel must be kept within the limits of the new Figure 3.2-2, which is identical to the old Figure 3.2-4. Section 3.2.4.d states that when COLSS is out of service and neither CEAC is operable, CPC calculated DNBR on any operable channel must be kept within the limits of Figure 3.2-3. The new Figure 3.2-3 is a power independent figure similar to the new Figure 3.2-2, but it accommodates the increased margin required when both CEACs are inoperable. The proposed change to Technical Specification 3.2.4 also modifies the existing action statement to be more clear and consistent with the LCO as described above. Also, Surveillance Requirement 4.2.4.2 is revised to allow use of any operable CPC channel for monitoring the DNBR with COLSS out of service. This proposed change recognizes that it is acceptable to monitor any one channel for control purposes during steady state operation. It is not necessary that the monitored channel be the most limiting since appropriate uncertainty allowances are already implemented in the CPCS calculations and trip setpoints. The CPCS continues to provide the required protection during transient operation.

b. This proposed change would revise Technical Specification 3/4.3.1, Table 3.3-1, ACTION 5 which provides conditions under which operation may continue for various operability conditions of the CEACs. ACTION 5.a will be revised to allow operation to continue after 7 days provided ACTION 5.b is met, in which more restrictive actions must be taken. ACTION 5.b addresses operation with both CEACs inoperable and will be revised to be consistent with the changes to 3/4.2.4, described above, by referencing the appropriate requirement of 3/4.2.4, depending on COLSS in service or COLSS out of service. The COLSS calculated core power operating limit based on linear heat rate remains conservative without additional penalty for both CEACs inoperable. ACTION 5.b.2.c) is revised to clarify that the CEDMCS may be removed from the "Off" mode as specified in ACTION 5.b.2.a).

4. This proposed change will revise Technical Specification 3/4.3.1, "Reactor Protective Instrumentation," and the associated Bases. Technical Specification 3/4.3.1 requires that the Reactor Protective Instrumentation System (RPIS) be operable and defines the number and type of RPIS channels required, response times and periodic testing required to assure operability and actions to be taken when the required RPIS is out of service. Table 3.3-2 defines the maximum reactor protection instrumentation response times in order to verify that the maximum response times for the RPIS assumed in the Final Safety Analysis Report (FSAR) are not exceeded. The Table notes that these response times are based on a resistance temperature detector (RTD) response time of less than or equal to 6.0 seconds, the value used in the accident analyses for Cycle 5. If the effective RTD response time constant for a CPC channel exceeds 6.0 seconds, the DNBR and linear heat rate penalties for the affected channel(s) are required to be increased per Figure 3.3-1, and the DNBR POL decreased per Table 3.3-3. The CIP and related accident analyses now assume effective response times of 8.0 seconds for the reactor coolant system (RCS) cold leg temperature RTDs, and 13.0 seconds for the RCS hot leg temperature RTDs. The RTD response times and related actions specifications are therefore being modified to be consistent with the CIP changes to the CPCS dynamic compensation algorithms and the assumptions used in the uncertainty analysis. This change will revise the maximum RTD response time from 6.0 to 8.0 seconds in Note ## appended to item 10.c, "Cold Leg Temperature" of Table 3.3-2.

A new Note ### is appended to item 10.d, "Hot Leg Temperature," to reflect the allowable hot leg RTD response time of 13.0 seconds. Figure 3.3-1 is revised accordingly, and Table 3.3-3 is deleted since the analyses assumes longer response times and adjustments are no longer needed for RTD response times $\leq$ 13.0 seconds. Also, the definition of RTD response time is moved from Table 3.3-2 to a more appropriate location in the related Bases, and the Bases will be modified to reflect the revised RTD response times.

BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION

The proposed change does not involve a significant hazards consideration because operation of Arkansas Nuclear One Unit 2 in accordance with this change would not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated. This change results from enhancements to the Core Protection Calculator System (CPCS) as a part of AP&L's participation in the CPCS Improvement Program (CIP). A part of the implementation of this program at ANO-2 was a review of the plant specific transient analyses to determine the effect of the functional changes to the CPCS on the reference cycle (Cycle 5) analyses. In each case, it was found that the revised CPCS software provides for protection system action at least as quickly as credited in the reference cycle analyses. Therefore, the existing analyses remain bounding.

(2) create the possibility of a new or different kind of accident from any previously analyzed. It has been determined, in conjunction with the analyses described in (1) above, that a new or different kind of accident will not be possible due to implementation of the CPCS software enhancements for the CIP at ANO-2.

(3) involve a significant reduction in a margin of safety. As stated in (1) above, the exisiting safety analyses remain bounding with the implementation of this change. This change does not involve a significant reduction in a margin of safety. Because this change is the product of the CIP, which involves CPCS enhancements, performance improvements and reductions in the possibility of spurious protection system actuations, it may in fact increase the margin of safety.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve significant hazards consideration. Example (i) relates to a purely administrative change to the Technical Specifications: For example, a change to achieve consistency throughout the Technical Specifications, correction of an error or a change in nomenclature. Example (ii) relates to changes which may constitute an additional limitation, restriction or control not presently included in the Technical Specifications. Example (vi) relates to a change which either may result in some increase in the probability or consequences of a previously analyzed accident or may in some way reduce a safety margin, but where the results of the change are clearly within all acceptance criteria with respect to the system or component specified in the Standard Review Plan (SRP): For example, a small refinement of a previously used calculation model or design method. The proposed changes are similar to one or more of these examples. The specifics of how each proposed change is similar to the examples of 48 FR 14870 are discussed below:

1. Section 7.2, "Reactor Trip System," requires that the reactor protection system automatically initiate a reactor trip to assure that specified acceptable fuel design limits are not exceeded. This change is similar to Example (vi) of 48 FR 14870. Although the increased LPD trip setpoint may be perceived to reduce in some way a margin of safety, adjustments for dynamic effects which were previously included in the trip setpoint are now accounted for elsewhere in the CPCS algorithms. The net effect is that the CPCS with the revised setpoint will continue to initiate a reactor trip to assure that specified acceptable fuel design limits are not exceeded. Therefore, the proposed change satisfies SRP Section 7.2 acceptance criteria and is similar to Example (vi) of 48 FR 14870.

2. This proposed change replaces the existing Figure 3.2-1 with words to the same effect, revises existing Figure 3.2-2 and makes it the new Figure 3.2-1 and also revises the existing Limiting Conditions for Operation (LCO), Surveillance Requirements and associated Bases of 3/4.2.1 to provide clear and consistent monitoring, action and surveillance requirements. This modification is similar to Example (i) of 48 FR 14870 in that it relates to a purely administrative change to the Technical Specifications. Replacement of a Figure with words to the same effect and revisions of the LCO, Action and Surveillance Requirements will be made to clarify the requirements. The proposed change pertains to a revision of a graphic representation of the LCO with a set of plain administrative control statements easy for understanding. The reduction of the linear heat rate limit value of the new Figure 3.2-1 is similar to Example (ii) in that it constitutes an additional limitation, restriction or control not presently included in the Technical Specifications.

3. The proposed change described in (a) revises Figures 3.2-3 and 3.2-4, replaces the existing LCO with four parts, i.e., Sections 3.2.4.a through 3.2.4.d, and also revises the related Surveillance Requirements and associated Bases to provide clear and consistent monitoring, action and surveillance requirements. This modification is similar to Example (i) of 48 FR 14870 in that it relates to a purely administrative change to Technical Specifications by imposing four applicable administrative control methods and two new Figures in lieu of two existing Figures to maintain an adequate DNBR margin under different states of plant operations. 3.2.4.a and 3.2.4.b replace the existing Figure 3.2-1 with words to the same effect when COLSS is in service. Additionally, both new Figures supplant the existing Figure 3.2-4 for compliance with 3.2.4.c and 3.2.4.d when COLSS is out of service. Thus, DNBR will be maintained by 3.2.4.a (or 3.2.4.c) when at least one CEAC is operable, and by 3.2.4.b (or 3.2.4.d) when neither CEAC is operable. Since the proposed change pertains to a revision of graphic representations of an LCO with a set of plain administrative control statements for easy understanding and two consolidated figures for simplification, it is a change within the scope contemplated by Example (i). This change may also be considered similar to

Example (ii) in that the addition of the new Figure 3.2-3 constitutes an additional operating restricting, not presently included in the Technical Specifications, for monitoring DNBR with COLSS and both CEACs out of service.

The proposed change described in Part (b) revises the ACTION statements in Table 3.3-1 of Technical Specification 3.3.1. Specifically, ACTION 5.a is revised to allow continued operation after 7 days with one CEAC inoperable, providing the more restrictive requirements of 5.b are met. ACTION 5.b is revised to provide consistency and reflect changes to 3.2.4.b and d. This is therefore similar to Example (i) in that it is purely administrative change to promote consistency throughout the Technical Specifications. The change to ACTION 5.a is similiar to Example (ii) in that it requires compliance with ACTION 5.b, which in turn references more restrictive requirements of 3.2.4, after 7 days of operation with one CEAC inoperable.

4. This change revises the RTD response time requirements to reflect the CPCS software revisions and assumptions used in the uncertainty analysis for the implementation of the CPCS Improvement Program (CIP). This change may be considered similar to Example (vi) of 48 FR 14870 in that it reflects the refinement of a previously used calculation model and design method. Although the allowed RTD response times are increased, the CPCS dynamic compensation algorithms and constants reflect these increases and the net margin of safety is preserved.

Therefore, based upon the discussion and reasoning presented above, AP&L has determined that this Technical Specifications amendment package does not involve a significant hazards consideration.