Omaha Public Power District

1623 HARNEY ■ OMAHA, NEBRASKA 68102 ■ TELEPHONE 536-4000 AREA CODE 402

January 28, 1980

Director of Nuclear Reactor Regulation
ATTN: Mr. Robert W. Reid, Chief
Operating Reactors Branch No. 4
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Reference: Docket No. 50-285

Gentlemen:

Omaha Public Power District hereby submits forty (40) copies of supplemental material in support of (1) the Application for Amendment of Operating License ("Stretch Application"), filed July 17, 1979, which seeks to amend Facility Operating License No. DPR-40 to permit Cycle 6 operation following core reload at an increased power level of 1500 MWt, and (2) the Application for Amendment of Operating License ("Reload Application), filed July 17, 1979, which seeks to permit Cycle 6 operation following core reload. Forty (40) copies of the following materials are enclosed:

(1) Revised Startup Physics Testing summary.

(2) Responses to NRC setpoint methodology questions received October 30, 1979.

(3) Proposed Technical Specifications addressing RCS heatup and cooldown pressure/temperature limitations.

(4) Discussion supporting item (3), proposed Technical Specifications.

Should you desire additional information on these materials, please advise us.

Sincerely,

W. C. Jones
Division Manager
Production Operations

WCJ/KJM/BJH:jmm

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Enclosures

cc: LeBoeuf, Lamb, Leiby & MacRae
Mr. Peter B. Erickson (NRC)

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DISCUSSION

The 400°F limit in the existing Technical Specification 2.1.1(6) was specified by Combustion Engineering as a means of protecting personnel working near the hot piping. The District has determined that the hazard presented by 400°F piping is essentially as great as the hazard presented by 500°F piping. Therefore, the 400°F limit has been eliminated in the proposed Technical Specifications.

The heatup and cooldown curves, pressure vs. temperature, are designed to indicate, for various types of operation, the limiting temperatures at which the corresponding pressure cannot be exceeded to ensure that the metal subject to the stress of that pressure remains ductile.

The minimum fracture toughness of the reactor vessel beltline, as described in Appendix G of 10 CFR 50, continually changes throughout the life of the plant. This Facility License Change 79-13 is submitted as a result of changes in the minimum fracture toughness brought about by the exposure of the reactor vessel beltline to a predicted neutron fluence of 8.94×10^{18} n/cm^2 through the end of fuel Cycle 6. The predicted fluence is based upon the results of irradiated material sample testing performed in accordance with Appendices G and H of 10 CFR 50. The CE report, "Post-Irradiation Evaluation of Reactor Vessel Surveillance Capsule W-225", dated May, 1979, documents the test results and provides a new predicted end of life surface fluence of 5.5×10^{19} n/cm^2 at 32 Effective Full Power Years (EFPY) of operation.

The resulting heatup and cooldown curves have been adjusted to account for the total shift of the Reference Nilductility Transition Temperature (RT_{NDT}) of 223°F. The shift of 223°F has been added to the baseline, non-irradiated material curves to result in the new corrected curves to be used through the end of fuel Cycle 6. At that time, a new Facility License Change will be submitted to extend the curves for operation beyond Cycle 6.

The Low Temperature Overpressure Protection (LTOP) system is designed to prevent the primary system pressure from exceeding the pressure-temperature limits (Technical Specification Figures 2-1A and 2-1B) in the event of an inadvertent mass or energy addition. LTOP system actuation setpoints, as well as temperatures for disabling High Pressure Safety Injection (HPSI) pumps, will be determined assuming failure of one of the two PORV's. Calculations will be based upon a PORV discharge coefficient of .45, which a comprehensive testing program has shown to be conservative for subcooled liquids.

Inadvertent actuation of three (3) HPSI pumps and three (3) charging pumps, coincident with the opening of one of the two PORV's, would result in a peak primary system pressure of 1190 psia. 1190 psia corresponds with a minimum permissible temperature of 320°F on Figure 2-1B. Thus, at least one HPSI pump is disabled at 320°F.

Inadvertent actuation of two (2) HPSI pumps and three (3) charging pumps, coincident with the opening of one of the two PORV's, would result in a peak primary system pressure of 1040 psia. 1040 psia corresponds with a minimum permissible temperature of 310°F on Figure 2-1B. Thus, at least two HPSI pumps will be disabled at 310°F.

Inadvertent actuation of one (1) HPSI and three (3) charging pumps, coincident with opening of one of the two PORV's, would result in a peak primary system pressure of 685 psia. 685 psia corresponds with a minimum allowable temperature of 276°F on Figure 2-1B. Thus, all three HPSI pumps will be disabled at 276°F.

Inadvertent actuation of three (3) charging pumps, coincident with opening of one of the two PORV's, would result in a peak primary system pressure of 160 psia. 160 psia corresponds with a minimum allowable temperature of 78°F (approximately the boltup temperature of 82°F) on Figure 2-1B. Thus, disabling of the charging pumps is not required.

Removal of the reactor vessel head, one pressurizer safety valve, or one PORV provides sufficient expansion volume to limit any of the design basis pressure transients. Thus, no additional relief capacity is required.

SECTION D

OMAHA PUBLIC POWER DISTRICT

FORT CALHOUN STATION

STARTUP PHYSICS TESTING

1. Introduction

The principal tests in the proposed Cycle 6 startup physics test program are listed below. These tests are sufficient to show that the as-loaded core's parameters are within the bounds of the safety analysis, thus permitting continued safe operation. Acceptance and review criteria, along with the action to be taken if these criteria are not met, are also discussed. The predicted values of the parameters being measured will not be calculated until after the end of Cycle 5 when the actual fuel exposures are known.

2. Startup Tests

2.1 Hot Functional Tests

Prior to the approach to the initial criticality of Cycle 6, normal surveillance testing and operating procedures will be completed. During this sequence, applicable surveillance tests are performed to check CEA position indication and all other interlock and control features of the rod drive system.

2.2 Initial Criticality and Low Power Physics Tests

Following the dilution to initial criticality, the following reactivity parameters will be measured at less than 10^{-1}% of rated power:

2.2.1 Critical Boron Concentration

2.2.1.1 Hot Zero Power, Partial Insertion of Group 4

2.2.1.2 Hot Zero Power, All Rods Out

2.2.2 Isothermal Temperature Coefficient - HZP, Nominal ARO

2.2.3 CEA Group Worths

2.2.3.1 Individual Bank Worths of Regulating Groups 4, 3, 2, and 1 - HZP

2.2.3.2 Sequential (overlapping) Worth of Regulating Groups 1, 2, 3, and 4

2.2.4 CEA Symmetry Checks - HZP

2.3 Power Ascension Tests

2.3.1 Tests Performed at a Nominal 50% of Rated Power

Following the acceptable comparison of measured and predicted reactivity parameters, reactor power will be increased to a nominal 50% of rated power and a power distribution verification performed. This verification will be performed in a non-equilibrium xenon state with measurement of the following parameters.

2.3.1.1 Total Unrodded Planar Radial Peaking Factor (F_{xy}^T)

2.3.1.2 Total Integrated Radial Peaking Factor (F_R^T)

2.3.1.3 Azimuthal Power Tilt, Incore Detectors

2.3.2 Tests Performed at a Nominal 70% of Rated Power

A power distribution verification will be performed at this power level after equilibrum xenon has been established with measurements of the same parameters as in Section 2.3.1.

2.3.3 Tests at Nominal 100% of Rated Power

Following completion of testing at the lower at-power levels, power will be increased to a nominal 100% rated thermal power at a rate commensurate with fuel performance guidelines. After the establishment of equilibrium xenon, the following parameters will be measured.

2.3.3.1 Isothermal Temperature Coefficient

2.3.3.2 Power Coefficient

2.3.3.3 Critical Boron Concentration, ARO

2.3.3.4 Total Unrodded Planar Radial Peaking Factor (F_{xy}^T), ARO

2.3.3.5 Total Integrated Radial Peaking Factor (F_R^T), ARO

2.3.3.6 Azimuthal Power Tilt, Incore Detectors, ARO

3. Acceptance - Review Criteria

Acceptance criteria are applied to the test results, after conservatively adding measurement uncertainty to the measured value, to insure that the core conforms to the physics design and that plant response to transients is in accordance with the safety analysis. Review criteria are also applied to highlight any lesser deviation which may indicate that the core was incorrectly loaded or to confirm that the assumptions used in the design analyses are valid. Acceptance and review criteria for the low power physics parameters measured are listed below.

Parameter	Acceptance Criteria	Review Criteria
Rod Drop	Technical Specifications	Previous values
Critical Boron Concentration	± 90 ppm of predicted	± 50 ppm of predicted
Isothermal Temperature Coefficient	Technical Specification limits of Moderator Temperature Coefficient	$\pm 0.3 \times 10^{-4}\ \Delta\rho/^{\circ}F$
CEA Group Worths	± 15% of predicted	± 15% of predicted
Total Regulating CEA Group Worth	-10% of predicted to ensure adequate shut-down margin	± 10% of predicted
CEA Symmetry Checks	None	The greater of: 1.5¢ deviation from group average or 15% deviation from group average

The review criteria for the CEA symmetry checks will be ± 1.5¢ or 15% deviation from the group average, whichever is greater. The reason that two types of criteria are stated is that for high worth rods, a percent deviation is appropriate as is applied to other rod worth measurements. For small worth rods, however, an absolute deviation is required which is the same type of allowance as specified for reactivity coefficient measurements. This is not intended to be a go-no-go criterion, but rather an indication of the degree of tilt that might cause the Azimuthal Power Tilt specification to be exceeded. If the criterion is not met during the test, the test program will be extended to reconfirm the measured value. If the values still fall outside the stated criteria, the results will be reviewed to determine the potential impact upon plant operations.

The acceptance criteria for power distribution verifications are the limits cited in the Technical Specifications. The review criterion for the comparison of the predicted and measured full core power distributions of the instrumented assemblies is a 5% standard deviation.

Acceptance and review criteria for the at-power critical boron concentration measurements are the same as for low power physics testing.

Acceptance criteria for the nominal 100% power Isothermal Temperature Coefficient (ITC) and Power Coefficient (PC) shall be the Technical Specification limits on the Moderator Temperature Coefficient (MTC).

4. Action and Review Plans

The following plan of action is provided if a measured parameter differs from the predicted value by more than the acceptance criteria.

4.1 The physics test program will be extended to reconfirm the measured value. If the total regulating CEA group worth is less than 10% of the predicted value, shutdown bank worth measurements will be made.

and/or

4.2 The predicted value will be reviewed to ensure that it accurately reflects the particular plant conditions under which the measurement was made and refined if appropriate.

4.3 If, after the above two steps, the disagreement persists, the safety analysis will be reviewed to determine whether the measured value of the particular parameter in question, when combined with all of the other safety related parameters, increases the severity or consequences of accidents or anticipated operational occurrences. If equivalent safety for the plant can be demonstrated, the test results will be deemed acceptable.

4.4 The other physics related safety parameters will be verified to be within acceptable limits by additional measurements if necessary.

4.5 If the combination of safety parameters determined above fall outside of range of safety parameters used to support the proposed operation of the plant, the plant operating limits will be adjusted to prevent conditions which could result in exceeding the specified acceptable fuel design limits.

The Plant Review Committee and Technical Services will review the results of the low power physics tests and ensure that the acceptance criteria are met prior to allowing escalation above five percent of rated thermal power. The at-power testing results will be reviewed prior to reaching 100% power.

If after review of the data it is determined that a Technical Specification limit has been exceeded, then appropriate action as required by Technical Specifications will be taken.

Results of startup testing will be submitted to the NRC within 90 days following completion of the tests. This report will summarize the test results and include a comparison of the measured and predicted values of low power physics parameters and a full core power distribution comparison including deviations between the measured and predicted relative power densities of operable instrumented assemblies. If the difference between the measured and predicted values exceed the acceptance and/or review criteria, the report will discuss the actions that were taken and also justify the adequacy of these actions.

OMAHA PUBLIC POWER DISTRICT

FORT CALHOUN STATION

DOCKET NO. 50-285

RESPONSE TO NRC SETPOINT
METHODOLOGY QUESTIONS

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QUESTION 1

List all operational maneuvers or conditions considered in generating axial power shapes. Also list the control rod configuration and burnup. Describe how the Xenon oscillations were induced.

Also describe whether the plant was base loaded or in a load follow configuration. Give the power history assumed. This information may be provided most efficiently in matrix form.

Justify that only these maneuvers need be considered for generating axial power shapes.

For Ft. Calhoun Cycle 6, how many axial power shapes were generated and how many were used in the set point analysis?

RESPONSE

The Fort Calhoun Cycle 6 core was modeled in three-dimensions with the computer code XTG[1] and depleted in a base load (1500 MWt) all rods out configuration. This reference power history and rod configuration for Cycle 6 was chosen based on the power and control rod histories of previous cycles (Cycles 3, 4, and 5) and OPPD's anticipated operating requirements for the Fort Calhoun Nuclear Power Plant throughout Cycle 6. (The plant is currently finishing up the 5th cycle.) A bar chart showing the actual plant power and control rod configuration by month for Cycles 3, 4, and 5 and the anticipated power and rod histories assumed in the analysis for the determination of Cycle 6 setpoints are shown in Figures 1.1 and 1.2.

Three different burnups, from the three-dimensional XTG Cycle 6 core depletions, were chosen as the base cases for the starting point of more than

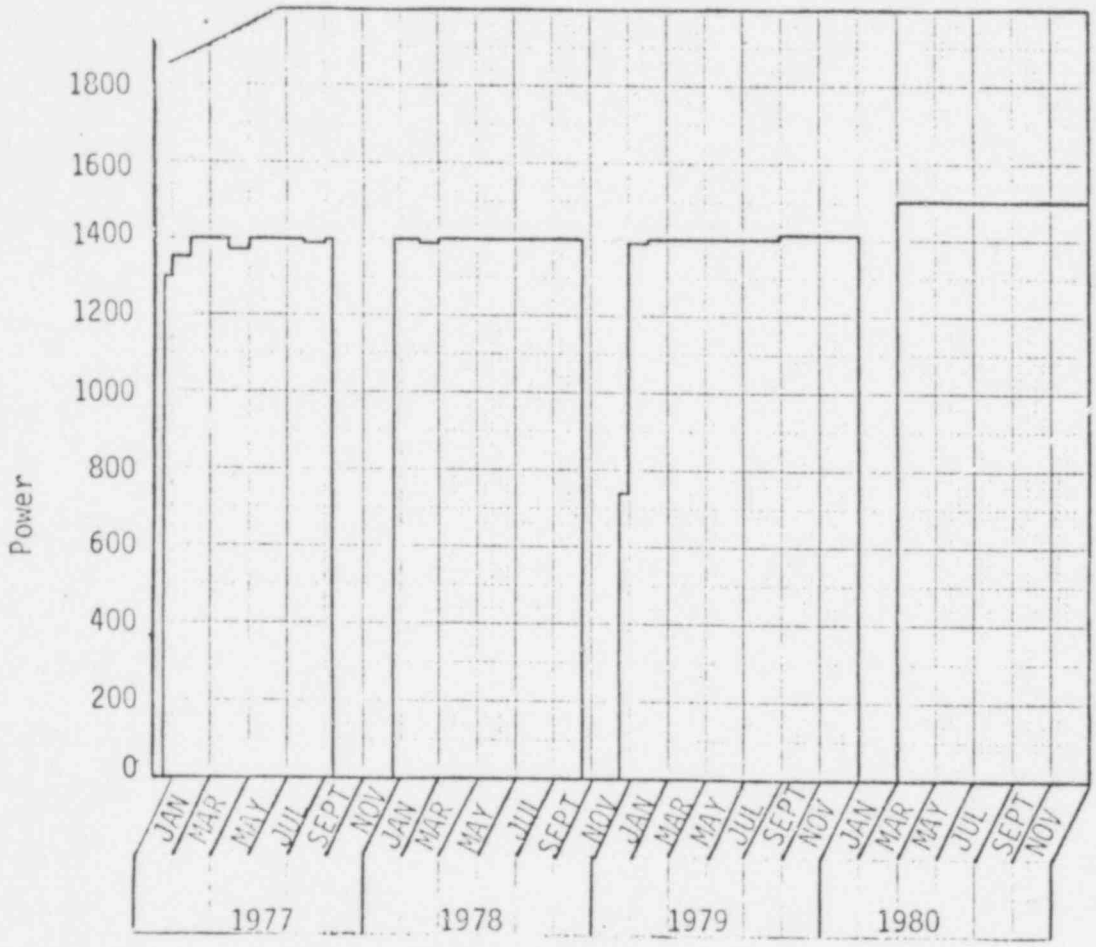
1500 axial power profiles generated and used in the determination of Cycle 6 setpoints. The burnups chosen were 0 MWD/MT (equilibrium xenon), 6000 MWD/MT, and 10,000 MWD/MT. The burnups correspond approximately to the beginning, middle and end of the Cycle 6 exposure range. At each of the above mentioned exposure points many possible Cycle 6 axial power profiles were calculated by inducing xenon oscillations. The xenon oscillations were incited by inserting the Control Element Assemblies (control rods or CEA's) to the Power Dependent Insertion Limits (PDIL) at 100% power for 8 hours and then instantly removing the CEA's.

In order to analyze the possible axial power distributions within a wide range of Axial Shape Index (ASI), severe xenon oscillations were created. The oscillations resulted in axial power distributions covering a range of Axial Shape Index (ASI) units for values ranging from -60% to +60% offset. A negative ASI is top peaked in the core. The normal axial shape index for the based loaded Fort Calhoun core covers the range from -3% to +2% ASI units. Figure 1.3 shows the anticipated ARO steady state axial shape index units calculated for Cycle 6.

Control rod effects on the axial power profiles were also determined. At selected points throughout the xenon oscillation control element assemblies were inserted to the PDIL limits in 10% power increments. At each point selected the xenon distribution was fixed allowing the axials to change depending on rod movement only. These maneuvers gave the many possible axial power profiles at partial power and CEA insertion.

Fort Calhoun

Measured and Anticipated Core
Power by Month



Month

Figure 1.1

Fort Calhoun

Measured and Anticipated Control Element
Assembly Position by Month

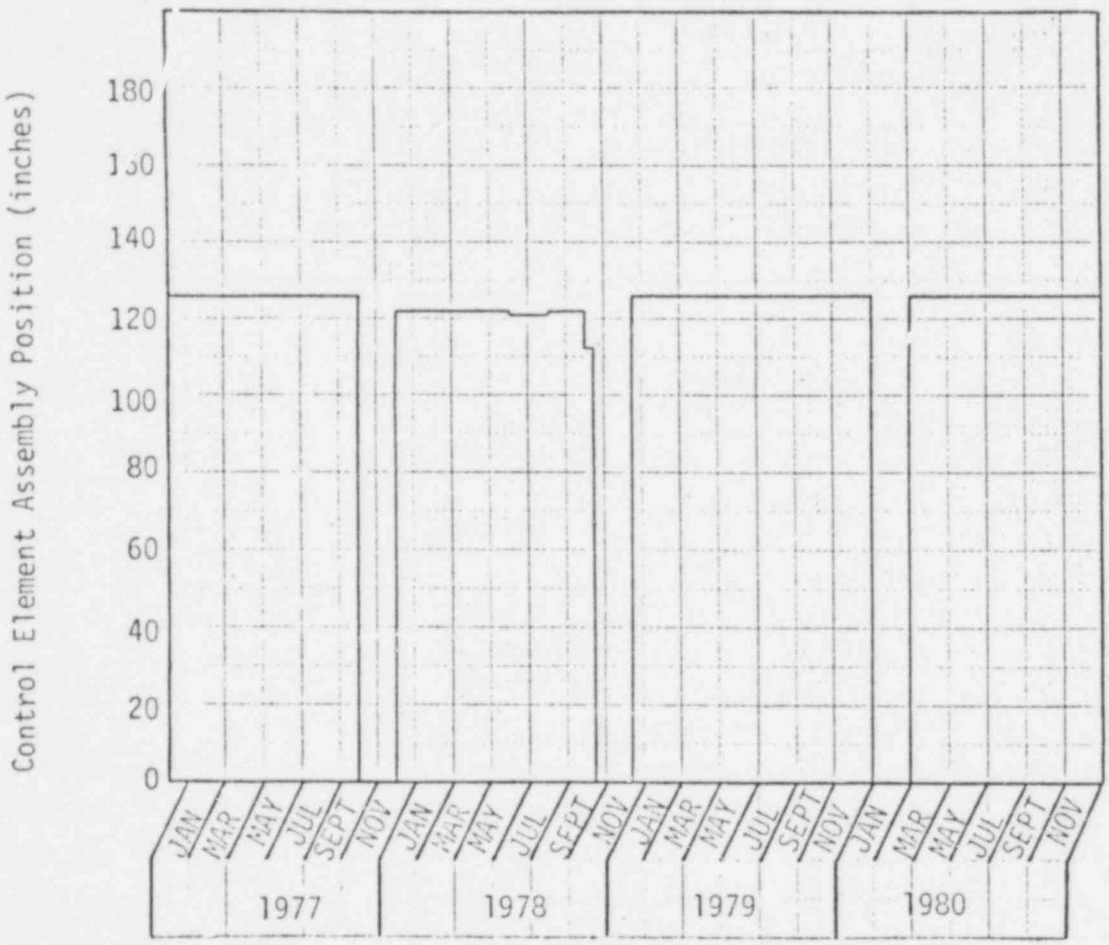


Figure 1.2

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Fort Calhoun Cycle 6 Axial Shape Index vs.
Exposure ARO, 1,500 MWt

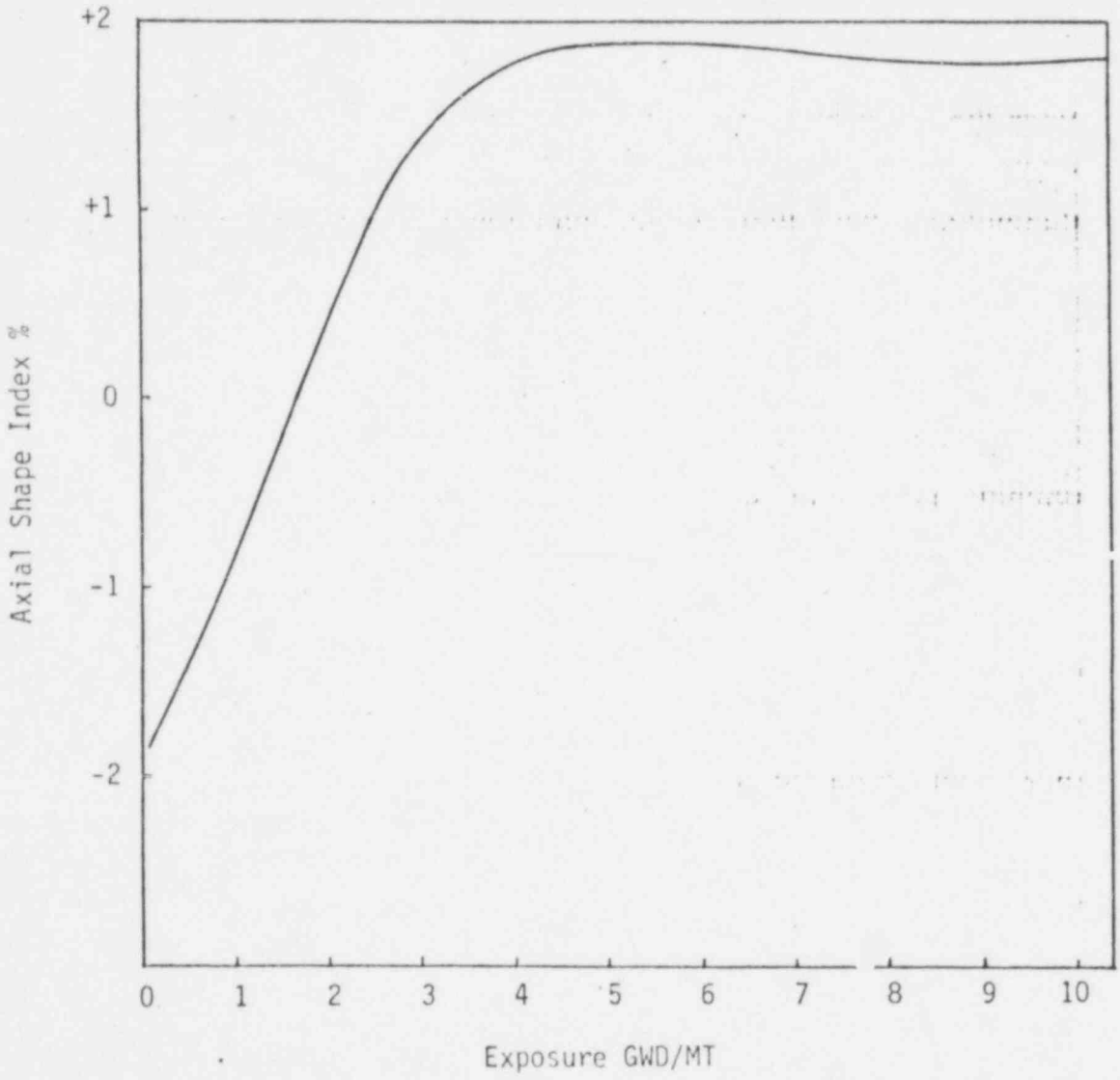

Figure 1.3

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QUESTION 2

For the axial power shpaes in Question 1, describe the analytical methods used to calculate these shapes. Provide the spatial noding schemes for each computer calculation and the size of the time steps used. Describe how reactivity feedback effects were included.
How many energy groups are used in these calculations? Justify this number of energy groups.

RESPONSE

Analytical methods used to calculate the axial power shapes adhered to in Question 1 were determined using the Exxon Nuclear Neutronic design methods for PWR's (NRC approved) described in References 2, 3, and 4 and Section 6.1 of Reference 5. Specifically the methods used to calculate the axial power shapes for Cycle 6 included the computer codes XPOSE,(6) PDQ,(7/8) and XTG(1).

The computer code XPOSE, a modified version of the industry accepted LEOPARD code, was used to generate fast and thermal spectra and cross sections in two energy groups for the reactor simulator codes, XTG and PDQ7. Detailed two-dimensional pin-by-pin radial power distributions for the core were determined with PDQ7. This information was used in conjunction with the XTG results to determine values of F_{xy} for the core.

The reactor simulator code XTG was used to determine the wide range of axial power profiles referenced in Question 1. The core was modeled and depleted in three-dimensions with the XTG code. The 3-D XTG model was used to produce reference axial power profiles and core average cross sections and exposures as a function of core height. These core average cross sections and exposures were then used in the one-dimensional XTG core model to determine

all possible core average axial power distributions for the Cycle 6 Fort Calhoun core. The one-dimensional XTG model, consisting of 1 radial node and 24 radial nodes, was normalized to the three-dimensional XTG model containing 4 radial and 12 axial nodes or 48 nodes per assembly.

In order to sustain the xenon oscillation at the beginning, middle and end of Cycle 6, reactivity feedback effects were varied in the one-dimensional XTG model. At the beginning of the cycle the reactivity feedbacks effects due to Doppler and Moderator density were removed from the calculation. A sustaining and slightly divergent xenon oscillation was not possible with these feedbacks in the calculation. The feedback effects were used in the middle and end of cycle axial power shape calculations. Without the effects of the feedback included in the calculation at the end of cycle, the xenon oscillation and corresponding power distributions became very divergent and unrealistic. One hour time steps were used throughout all the xenon oscillation calculations.

QUESTION 3

Describe in more detail the calculation of F_{xy} done to determine F_Q^T discussed in Section 4.1.1 XN-NF-507. Describe the PDQ model and show how (and which peaking factors from PDQ are used in XTG to determine F_{xy}. Also, show the XTG 3D modeling. Into how many increments are the CEA insertions divided for the various F_{xy} calculations? How many energy groups are used in these calculations? Justify using this number of energy groups.

RESPONSE

The calculation of the ratio of the power of the peak fuel pin to the average fuel pin in the core at height Z is commonly referred to as F_{xy} and is explicitly calculated in the three-dimensional XTG core simulator code referenced in Questions 1 and 2. To determine F_{xy}, XTG calculates a three-dimensional power distribution for the core. The typical core power distribution consists of powers in 12 axial planes with four radial nodes per assembly in each plane. XTG calculates an F_{xy} in each of the 12 planes by determining the peak nodal power times the local peaking factor (F_L) in each plane, the average nodal power in each plane, and divides the peak by the average to determine the planar F_{xy}. The core F_{xy} is defined as the maximum of the planar F_{xy} values. Values for the local peaking factors, F_L are directly input into the XTG calculation by assembly or by fuel type. The local peaking factors are explicitly calculated in the quarter core PDQ model.

The PDQ7 core simulator is modeled to perform detailed two-dimensional radial calculations. The core is modeled in PDQ7 on a pin-cell basis; i.e. one

mesh block per fuel cell. Each pin-cell has the appropriate nuclide concentration of the burnup history for that pin. The Fort Calhoun PDQ7 model is similar to the model described in Reference 2. Output from the PDQ/HARMONY calculations include the pin-by-pin radial power distributions, F_r. The local pin power peaking, F_L, from PDQ7 is used as input in XTG for the XTG F_r and F_{xy} calculation. Values of F_L are derived from PDQ7 results for each assembly by dividing the average assembly power peaking into the peak pin power in that assembly.

Values for F_{xy} are determined throughout Cycle 6 at full power, with the CEA's withdrawn from the core At part power the F_{xy} values were determined with the CEA's at the transient power dependent insertion limits (PDIL). The power distribution calculations were made in increments of 10% power. The CEA's were inserted to the maximum allowable transient PDIL limits at that power.

All the calculations done with the core simulator model are made in two energy groups. Methods are discussed in Reference 2, 3, and 4.

QUESTION 4

Discuss to what extent the projected (utility planned) power history is included in the set point analyses (e.g., depletion calculations). Discuss the effect of a difference between projected power history and actual power history on the set points. If this effect is considered, demonstrate with sensitivity studies for various control rod maneuvers and changes in boration that power history effects are adequately considered.

Discuss the types of power maneuvers assumed (e.g., base loaded, 100-50-100, etc.).

RESPONSE

Operating projections by OPPD for the Fort Calhoun Nuclear Power Plant in Cycle 6 show operation of the plant at 1500 MWt at a 90% capacity factor for 320 days. This operating projection was used in the analysis determining set points. The answer to Question 1 shows the power and CEA configuration assumed for Cycle 6.

The axial power shapes calculated for Cycle 6 and used in the set point analysis bound all possible axial power shapes for Cycle 6. In addition, sensitivity studies were made to determine axial power shapes possible for operation of the core at power levels below 100% as well as at CEA insertion to the long term PDIL limits. The Technical Specifications preclude operation of the core for extended periods of time with CEA's inserted past the long term PDIL limits. In all cases the calculated axial shapes or the core peaking, F_Q, versus ASI were within the bounds of the setpoint calculations. Control rod histories assumed in the sensitivity studies included CEA insertions to 121 inches and 96 inches for the entire cycle. Power levels included in the sensitivity studies included 50% and 75% power.

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QUESTION 5

Discuss the initialization of conditions for the beginning of a cycle for set point calculations.

RESPONSE

Both core simulator models used in the Fort Calhoun set point analysis, XTG and PDQ7, were verified against measured data from Cycles 1 through 5. Both simulator codes were depleted through all previous cycles; i.e. Cycle 1, 2, 3, 4, and 5. In each cycle, comparisons between measured and calculated data were made. An assembly power comparison between measured data (CECORE) and calculated data (3-D XTG, and PDQ7) for Cycle 4 at 4000 MWD/MT is shown in Figure 5.1. Agreement between the three power distributions is good.

Initial conditions for the Cycle 6 setpoint work were based on the cycle depletions from the previous cycle depletions of Fort Calhoun, namely Cycles 1 through 5. Such cycle depletions were made with both referenced simulator codes, XTG in three dimensions and PDQ7. For Cycle 6 the burnup history of the fuel was explicitly accounted for.

Fort Calhoun

Assembly Power Distribution Comparison of Measured to Calculated Cycle 4, 4,000 MWD/MT, HFP

.856	1.050	1.209	.977	1.184	1.184	
.874	1.053	1.225	.977	1.190	1.128	
.851	1.038	1.233	.980	1.202	1.160	.859
	1.042	.857	1.131	1.101	.984	.886
	1.043	.895	1.105	1.076	.966	.865
	1.043	.865	1.115	1.093	.990	.704
		1.264	1.116	1.178	1.081	.711
		1.264	1.091	1.176	1.119	.703
		1.272	1.091	1.196	1.090	
			1.241	.845	.946	
			1.179	.851	.949	
			1.202	.847	.938	
				1.123	.589	← CECORE (Measured)
				1.104	.620	← 3D XTG (ENC)
				1.123	.614	← PDQ7 (ENC)

Figure 5.1

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QUESTION 6

(a) What critical heat flux correlation is used by Exxon in calculating the SAFDL on DNBR?

(b) Describe how limitations in the range of the correlation (e.g., the 15% quality limitation of the W-3 correlation) are accommodated in this SAFDL.

(c) Provide the specific criteria used to prevent exceeding flow stability limits and provide the justification for these criteria.

(d) The fuel melting limit is given on Page 10 of XN-NF-507 as 21 kw/ft. Is this number the result of an Exxon calculation? Justify the use of this number for two different fuel designs (Combustion Engineering and Exxon). What fuel melting temperature was assumed for this calculation?

Response

The W-3 burnout heat flux correlation with correction factors for the presence of a cold wall and nonuniform axial heat flux was used to establish the specified acceptable fuel design limit (SAFDL) on the fuel burnout performance (DNBR). The application and interpretation of the W-3 correlation for determination of the Fort Calhoun reactor set points is consistent with the ENC predictive models for DNBR (XN-75-48).

The SAFDL on DNBR is protected by limiting the operating values of core power, coolant inlet temperature, and system pressure to the most conservative of the following:

- that set of operating values which results in MDNBR = 1.3.
- that set which gives rise to parameter values exceeding the range of the W-3 correlation.

Thus, the W-3 correlation is never used with parameters outside its acceptable range, and the limits on its range are implicitly included in the SAFDL.

The specific criteria to preclude potential flow instability are: (1) a calculated subchannel quality of less than or equal or 15%, and (2) a core average exit quality less than or equal to zero. Adherence to these criteria is ensured by the SAFDL on DNBR and the core saturation limits. In addition, other reactor protection system limits such as the low pressure trip and variable overpower trip preclude reactor operation in a potential flow instability mode.

The ENC fuel melting limit of 21 kw/ft was calculated via the GAPEX code (XN-73-25) in conformance with the USNRC approved ENC fuel densification model for PWR fuels (USNRC report dated February 27, 1975). The value of 21 kw/ft as an LHGR limit for the existing fuel is documented in the Fort Calhoun Technical Specifications. Thus, the value of 21 kw/ft as a maximum fuel rod LHGR for both fuel types is judged to be acceptable.

A fuel melting temperature of 2790°C was assumed at beginning-of-life, and this value was decreased at the rate of 32°C per 10,000 MWD/MTU and compared against the fuel temperatures calculated at 21 kw/ft throughout fuel lifetime.

Question 7

Provide a complete description of the determination of the shape annealing factor, SAF, for Fort Calhoun. Is the determination of this factor consistent with Exxon calculational methods?

Response

The shape annealing factor (SAF) is defined as the ratio of the internal axial shape (ASI) to the external axial shape index. It can be expressed as:

$$SAF = \frac{ASI\ (internal)}{ASI\ (external)} = \frac{ASI\ (INCA)}{ASI\ (excore)}$$

where $ASI = \frac{L-U}{L+U}$, L is the power in the lower half core and U is the power in the upper half core.

The SAF of the Fort Calhoun core was determined by the induction of an axial oscillation transient during initial startup testing. The axial oscillations were induced by inserting the control rod from ARO conditions to approximately the reactor mid-plane and then returning the control rod to the ARO position. The oscillations were characterized by the axial shape index. During the oscillation ASI (INCA) data from the incore INCA code and ASI (excore) data of the excore detectors were monitored every four hours. The ASI data from INCA and the excores were plotted and a linear least squares fit analysis was performed. The slope of the fitted curve yielded a SAF value of 2.86.

This SAF value has been used in the axial power distribution (APD) calculator since Cycle 1. A subsequent measurements of the SAF performed at mid-Cycle 1 and Cycle 2 confirmed that the SAF value used for the Fort Calhoun Station maintains a degree of conservatism.

The SAF is a function of material and geometry between the core and the detectors and the detectors themselves. It is independent of fuel type. The existing SAF and its original determination is consistent with Exxon calculational methods.

QUESTION 8

Provide the Exxon definition of rod shadowing. Describe in detail the calculations done to determine the adjustment to the core average axial shape index to account for rod shadowing effects. Give the results for Ft. Calhoun Cycle 6. Will these results change from cycle to cycle?

Combustion Engineering performs rod shadowing calculations using neutron transport theory (CENPD 188, Section 2.1.1.4.1). It appears that Exxon relies on diffusion theory. Justify this difference. Also, explain how uncertainties in this calculation are taken into account. Show the calculational models used to perform these calculations.

Discuss the effects of transients which change CEA position on rod shadowing factors. How is this effect included in the set point calculations?

RESPONSE

Effects of rod shadowing on the response of the Ft. Calhoun excore detectors are explicitly accounted for in the determination of set points. A conservative value of ±.02 Axial Shape Index units (ASI) is removed from the operating margin in the final calculation of all Limiting Safety System Settings and Limiting Conditions of operation requiring excore detector response as input. This value is presented as an uncertainty in Section 4.1.3 of Reference 5.

Rod shadowing is the effect of control rods distorting the flux in the peripheral assemblies which are the primary contributor to the signal seen by the excore detectors. The calculations determining set points use the core average axial offset or axial shape index (ASI). Therefore, the ASI measured with the excore detectors which are inputs to the set points must be adjusted for rod shadowing effects. This adjustment of the excore detectors response was determined to be less than ±0.02 ASI units for Cycle 6.

Calculational methods and models (codes) used to determine the effects of rod shadowing on the excore detectors is the transport theory code XSDRNPM,[9] PDQ7, and 3D-XTG. XSDRNPM is a one-dimensional code used to determine the attention of signal or flux encountered when traversing the water and steel region exterior to the outer assemblies. PDQ7 is used to determine the attention in signal through adjacent assemblies. The 3D-XTG code was used to calculate the ASI in all assemblies and the core average. By using the above calculations to determine the relative effect each assembly has on the excore detector response, and combining the actual effect with the ASI of each assembly, the excore response of ASI is determined. Comparing the calculated excore detector response to the calculated core average will give the adjustment in excore detector response required for Cycle 6. This value was calculated to be less than ±0.02 ASI units for Cycle 6.

Each cycle will require the determination of excore detector response due to rod shadowing. The conservative adjustment factor of ±0.02 ASI units will be adjusted accordingly on a cycle by cycle basis.

Transients affecting the CEA positions will not increase the rod shadowing adjustment factor above a value of ±0.02 ASI units. This value was used for an all rods out core configurations as well as for rods in core configurations.

Question 9

(a) For the uncertainty values listed on page 15 of XN-NF-507, justify the value listed. List any experiments used to obtain these values and give a full description of the calculations done to obtain these uncertainties from experimental results. Where values from previous cycles of operation are used, justify that these are appropriate for Exxon calculational methods. If any of these uncertainties are composed of several components, list all components and justify all the component values.

(b) For the trip overshoot, is this based on a transient analysis? Describe the analysis methods used to derive this value.

(c) For each uncertainty listed, give a statistical statement characterizing the confidence in this value.

(d) For each uncertainty, show, in detail, how it is included in the setpoint analysis.

Response

(a) and (c)

The justification for the uncertainty values listed are given below:

(1) Physics calculation measurement uncertainty

Peak LHGR	7%
F_R	6%

These uncertainties are applicable to the Technical Specification limits on F_{xy}^T and F_R^T and are effectively applied to both the limit and the measured value of these parameters. (Since the same uncertainty would be applied to both the limit and the measured value, the comparison of the two cancels out the uncertainty term. Therefore it is not applied to the Technical Specifications.) The values of F_{xy}^T and F_R^T have been and will be measured during Cycle 6 using the CECOR code and the methodology described in CENPD-145. The 7% uncertainty on peak LHGR and 6% uncertainty on F_R were accepted by NRC for CE reactors as interim uncertainties at the October 2, 1978, meeting between NRC and CE. These uncertainties were further confirmed at the March 6, 1979, CE/NRC meeting. Presently, the District is participating in the CE power distribution uncertainty program to justify lower uncertainties. The revised topical report, CENPD-153, is being prepared for submittal. The components and statistical confidence of the values is explained in drafts of this report previously given to NRC.

(2) Aximuthal tilt allowance, 3%

This allowance is the current LCO contained in the Fort Calhoun Technical Specifications. As a LCO, it represents a limit on tilt. Operation data has confirmed the suitability of this limit.

(3) Engineering tolerance uncertainty, 3%

The engineering tolerance uncertainty is included in the analysis because manufacturing tolerances in the specification of pellet density, pellet diameter, and pellet enrichment could produce an additional rod surface heat flux at a local hot spot. The engineering heat flux factor is determined from the following characteristic subfactors derived from the manufacturing tolerances:

		Subfactor
Pellet Density, TD	94.0±1.5	1.0160
Pellet Enrichment, w/o	3.5±0.05	1.0143
Pellet Diameter, in.	0.3700±0.0005	1.0014
Clad Diameter, in.	0.442+0.0035 -0.0015	1.0080
	Statistical Total	1.0229

It was assumed that the finished fuel pellet in the hot spot will deviate from nominal values by the specified allowance. The pellet density uncertainty to a 2σ variation while all other values represent an absolute limit. The resultant individual subfactors, when statistically summed, yield a tolerance uncertainty of 1.0229.

Fuel densification results in the shortening of the heated length which causes an increase in the linear heat generation rate. In-reactor shortening of the active fuel column length was conservatively evaluated from the following expression:

$$\frac{\Delta L}{L} = \frac{0.965 - \rho i}{2}$$

where:

ΔL = decrease in fuel column length
L = fuel column length
ρi = initial mean pellet density

Based on a mean pellet density of 94 percent TD and a fuel column length of 128 inches, the column shrinkage was evaluated to be 1.6 inches which yields an increase in local heat flux of 1.25 percent at constant core power. Compensating for the fuel column length decrease due to densification is the increase in fuel column length resulting from thermal expansion. This length increase is approximately 1.35 inches, which reduces the local heat flux by 1.06 percent. Thus, the net increase in local heat flux due to axial fuel column shortening is partially compensated for by the increase in fuel column length due to thermal expansion. The net 0.2 percent increase in heat flux is accounted for within the three percent engineering tolerance uncertainty.

(4) Power measurement uncertainty

2% of rated for LCO
5% of rated for LSSS

These uncertainties are a function of instrument and NSSS design and are independent of fuel type and reload cycles. These values have been used for all For Calhoun cores and are given in CENPD-199.

(5) Trip overshoot

Discussed below.

(6) Physics uncertainty in predicting CEA distribution effect on excore detectors ±.02 ASI

This uncertainty is addressed in the answer to Question 8.

(7) Physics uncertainty in applying shape annealing correction axial shape index limits 0.01 ASI

The shape annealing factor is extremely difficult to predict analytically because of the rather complex neutron scattering involved, it is determined experimentally as discussed in the response to Question 7. The procedure used involved the fitting of a linear relationship between the shape index determined from excore detector signals and the core average axial shape index. Analysis of the data including consideration of data scatter and excore detector level calibration indicate that an uncertainty in the analysis of the shape annealing factor equivalent to 0.01 ASI units conservatively represents the ability of procedures to determine the shape annealing function. Again, this uncertainty is a function of reactor geometry and is independent of fuel type and reload cycle. The statistical confidence of this uncertainty is discussed in CENPD-199.

(8) Excore detector subchannel calibration using incore detectors ±.01 ASI

This uncertainty represents the limit to which the excore ASI can be calibrated to the incore ASI. As such it represents an absolute limit which cannot be exceeded when the calibration is performed.

(9) Trip system processing ±.02 ASI

This uncertainty is currently being used at Fort Calhoun and is a function of the measurement and process instrumentation. Therefore, it is not sensitive to reload cycle or fuel type.

(b)

The trip overshoot uncertainty of 5% as listed on page 15 of XN-NF-507 is treated as an uncertainty of the existing reactor overpower trip to account for the possible variation in trip point due to calibration and measurement errors. This results in the use of a 112% overpower

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trip in the analysis of anticipated plant transients (XN-NF-79-79) and is consistent with existing Fort Calhoun Technical Specifications (section 1.3).

(d)

The application of uncertainties is explained in detail in the response to Question 10 for the TM/LP analysis, and in the response to Question 23 for the APD and DNB barn.

QUESTION 10

Section 4.2.1 lists typical uncertainties included in the TM/LP analyses. Explain in detail how each of these uncertainties are included in the set point calculations. Define what effects are covered by instrument processing error. Please state whether the depressurization transient uncertainty is the only adjustment to the TM/LP for transient effects. See Question 17.

Response

In determining the TM/LP trip function, uncertainties to account for the magnitude of nuclear peaking, engineering tolerances, and instrument processing are included in both the XCOBRA-IIIC (DNBR) and PTSPWR2 (transient) calculations. The application of the uncertainties in each of these calculations are described below.

The TM/LP trip function is derived directly from the TM/LP safety limit lines, and uncertainties included in the TM/LP safety limit analysis are implicitly included in the TM/LP trip function. The TM/LP safety limit analysis included appropriate uncertainties in the calculation of the limiting assembly LHGR as:

$$LHGR_T = \overline{LHGR} * F_M^U * F_{AT} * F_P^U$$

where,

$LHGR_T$ is the limiting assembly linear heat generation rate with uncertainties

$\overline{LHGR}$ is the nominal limiting assembly linear heat generation rate

F_M^U is the measurement/calculational uncertainty on F_R ($F_M = 1.06$)

F_{AT} is the allowable azimuthal tilt ($F_{AT} = 1.03$)

F_P^U is the core power measurement uncertainty used in TM/LP (1.03)

In addition to the above, an engineering uncertainty of 3% was applied to the DNB limiting rod (F_E^U = 1.03) and the nominal geometry of the limiting subchannel was adjusted to account for engineering tolerances associated with the fuel rod/guide tube dimensions.

The verification of the adequacy of the Cycle 6 TM/LP is accomplished by analyses of the anticipated operational occurrences (AOO) during reactor operation. The results of those analyses have been documented (XN-NF-79-79) and direct accounting of instrument response was included in the analyses. This included the following uncertainties: 2% for core power, 2°F on core inlet temperature, 22 psia on pressure plus a 25 psia operational pressure range. The response times associated with instrument processing were accounted for in the AOO analyses by assigning the appropriate total delay times to each of the LSSS trip functions as modeled in the analyses. These delay times represent the time interval associated with signal acquisition + processing, and the movement of control rods when trip conditions are encountered. The delay times used in the analyses are listed in XN-NF-79-79. The trip overshoot (XN-NF-507) was modeled as an uncertainty in the overpower trip in the AOO analyses (see response to Question 9).

No explicit accounting of the depressurization transient uncertainty was performed in determining the TM/LP trip function. Rather, the TM/LP trip function was determined from the TM/LP limit lines (SAFDL on DNBR) with sufficient margin to preclude penetration of the SAFDL on DNBR for steady-state operation and anticipated operational occurrences (see response to Question 24).

Question 11

Deleted by the NRC.

QUESTION 12

How is flux tilt included in set point calculations?

Response

A 3% azimuthal tilt allowance is included directly in the analysis of the TM/LP, and the APD and DNB barns such that the limiting rod power (peak pellet) analyzed includes the 0.03 value for T_q consistent with the proposed Cycle 6 Technical Specifications, in which the limiting peaking is defined as $F_R^T = F_R (1 + T_q)$ and $F_{xy}^T = F_{xy} (1 + T_q)$.

QUESTION 13

Will Exxon design of C.E. set points affect the Variable Overpower Trip Set point? If so, describe how Exxon designs this trip function.

Response

The ENC set point methodology does not affect the Variable Overpower Trip Set point. The existing overpower set point was modeled in the analysis in support of Cycle 6 operation for Fort Calhoun (XN-NF-79-79).

QUESTION 14

(a) Describe how the scram reactivity is included in the set point calculations. List each application of this item. Describe how the scram reactivity is determined.

(b) Describe or reference the Exxon modeling of CEA shutdown reactivity as a function of CEA position.

RESPONSE

The Generic Scram curve and worth are used directly in the analysis of the AOO's (Anticipated Operation Occurances) to verify the adequacy of the Cycle 6 set points.

Exxon Nuclear has generated a conservative generic trip reactivity curve by using a power distribution which was severly skewed toward the bottom of the core (+50% axial offset). The XTG computer code was used in the generation of the scram curve. Part of the study was made with a 3D, quarter core representation of an operating reactor. The remainder of the anlaysis was made with the ENC 1-D Power Distribution Control (PDC) model. The quarter core 3-D model was used to predict the negative scram reactivity insertion for nominal conditions (BOC and EOC) expected for steady state reactor operation. The 1-D model was used to study the effect of induced bottom peaked axial offsets up to an offset value of +50%. In both models the control rod banks were incrementally inserted in each case to determine the scram curves.

The ENC generic scram curve computed from the XTG models is based on the minimum scram negative reactivity insertion (N-1 rod worth) and the maximum allowable (by Technical Specifications) scram time. Figure 14.1 shows the Fort Calhoun scram curve for an allowable scram time of 2.5 seconds. The measured plant scram time is much shorter than 2.5 seconds.

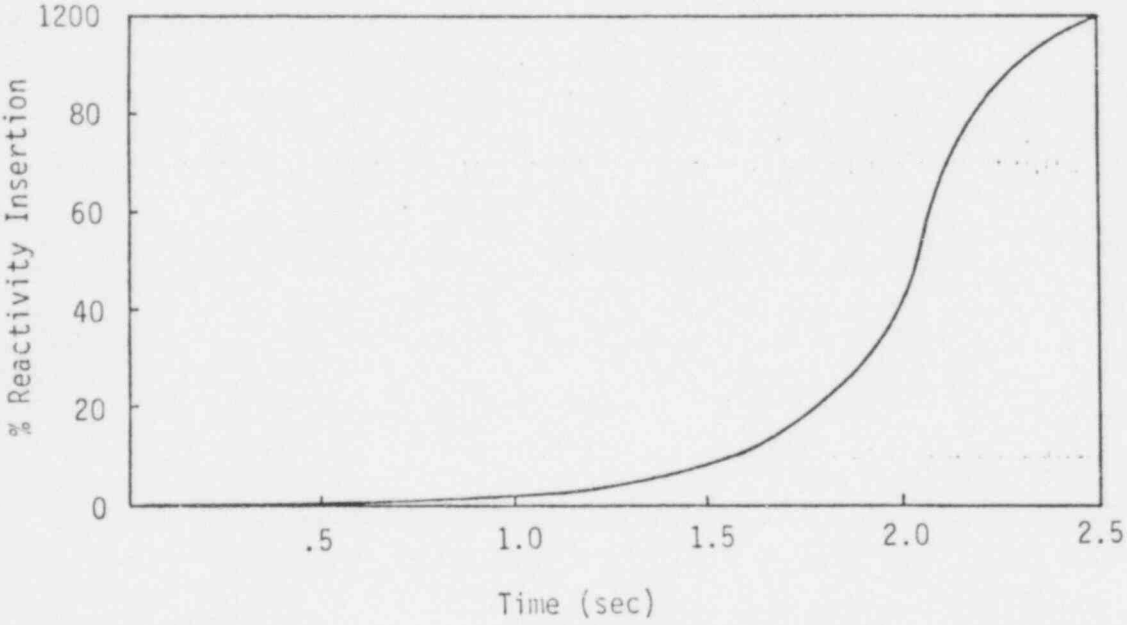

Figure 14.1 Fort Calhoun Scram Reactivity Curve

QUESTION 15

Describe in detail and provide appropriate equations to show how the various peaking factors and hot channel factors are included in the various thermal hydraulics calculations for the TM/LP trip. If this is done in the same way as other thermal hydraulics calculations and this information has already been provided to the Staff, the reference is sufficient.

Response

The XCOBRA-IIIC code in determining the TM/LP trip function allows explicit representation of appropriate peaking factors and uncertainties. The linear heat generation rate associated with the subchannel (DNBR) calculation is the limiting assembly peaking and is represented by the equation shown in the response to Question 10. The value of $\overline{LHGR}$ in that expression was selected to result in a minimum assembly flow rate for use in the subchannel calculations. This was accomplished by using the value of F_R^T as specified by the plant Technical Specifications divided by the minimum hot rod peaking anticipated throughout the fuel lifetime.

The application of the engineering heat flux factor and geometrical changes associated with tolerances on fuel rod/guide tube dimensions is discussed in the response to Question 10.

The XCOBRA-IIIC code uses the following expression to determine the local rod surface heat flux:

$$\phi\,(N,J) = LHGR_{AVG} * F_1 * F_2 * P * K * 1/\pi D$$

where,

$\phi\,(N,J)$ = local rod surface heat flux (BTU/sec-ft^2)

$LHGR_{AVG}$ = average LHGR for the model (bundle or pin)

F_1 = N^{th} pin or bundle peaking factor

F_2 = J^{th} nodal axial peaking factor

P = 1.0 for steady-state calculations

K = conversion factor (kw to BTU/sec)

D = fuel pin diameter (ft)

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QUESTION 16

Describe the calculation of the saturation limit curves. List any computer programs used for this calculation, if required. List all uncertainties included in the analyses and show how they are used.

Response

The saturation limit curves define those sets of reactor operating conditions (pressure, core power, core inlet temperature, core flow) which preclude a core average exit temperature equal to or greater than saturation temperature. The reactor average coolant exit conditions are judged not to be a function of fuel type and, as such, the existing saturation limit curves are applicable for Cyle 6 operation.

The specific criteria to preclude potential flow instability are: (1) a calculated subchannel quality of less than or equal or 15%, and (2) a core average exit quality less than or equal to zero. Adherence to these criteria is ensured by the SAFDL on DNBR and the core saturation limits. In addition, other reactor protection system limits such as the low pressure trip and variable overpower trip preclude reactor operation in a potential flow instability mode.

The ENC fuel melting limit of 21 kw/ft was calculated via the GAPEX code (XN-73-25) in conformance with the USNRC approved ENC fuel densification model for PWR fuels (USNRC report dated February 27, 1975). The value of 21 kw/ft as an LHGR limit for the existing fuel is documented in the Fort Calhoun Technical Specifications. Thus, the value of 21 kw/ft as a maximum fuel rod LHGR for both fuel types is judged to be acceptable.

A fuel melting temperature of 2790°C was assumed at beginning-of-life, and this value was decreased at the rate of 32°C per 10,000 MWD/MTU and compared against the fuel temperatures calculated at 21 kw/ft throughout fuel lifetime.

QUESTION 17

Describe how dynamic effects are accounted for in calculating the TM/LP trip. What computer codes are used? What transients are analyzed to determine the dynamic term? How are plant conditions initialized for the calculations to determine dynamic effects? List the values included for RPS delays to control rod actuation. What core power shapes are assumed? What reactivity feedback mechanisms are included? What values of reactivity feedback coefficients are used? Justify these values. Discuss the modeling of the core sensors which would measure core or loop conditions to assure adequate consideration of delays in system response.

Response

The ENC methodology for the determination of TM/LP does not consider any a priori knowledge of system transient performance to allow an explicit accounting of dynamic effects such as depressurization, system time response, trip overshoot, etc. Rather, a normalization of the set of lines representing those conditions corresponding to obtaining a SAFDL or DNB (MDNBR = 1.30) is performed so as to provide adequate protection against penetrating appropriate SAFDL values during steady-state and anticipated operational occurrences. Thus, any degradation of thermal margins due to changes in the reactor coolant conditions, time delays in instrument and scram response, power overshoot, must and are explicitly modeled in the transient calculations used to verify the adequacy of TM/LP. Thus, the ENC set point methodology includes both steady-state and transient performance evaluations.

The plant transient calculations directly determine the time dependent plant responses as they affect the core coolant conditions and the MDNBR is calculated during the transient analysis. The analyses include the thermal hydraulic

system response as well as the core neutronic response anticipated for Cycle 6. Any appropriate time delays associated with the reactor protection system are explicitly defined in the plant transient analyses (XN-NF-79-79) and are consistent with the anticipated RPS performance for Fort Calhoun. The uncertainties in nuclear peaking and control system instrumentation readings are consistent with ENC methodology (see response to Question 10). Uncertainties associated with other LSSS trip functions were applied to the nominal values in a limiting fashion, i.e., set to those values which result in increased thermal margin degradation during the analysis of the anticipated operational occurrences. The trip settings and delay times for these latter trip functions as used in the analysis are presented in XN-NF-79-79.

Question 18

Is the core ΔT-power calculator equation included in Exxon setpoint calculations for OPPD? If it is, provide a description of the procedure used to calculate the coefficients and give their values. If it will not be changed when Exxon calculates setpoints, explain why this is acceptable in view of the fact that different analytical methods will be used by Exxon. In particular, explain why this is acceptable for Fort Calhoun, which is planning operation at a higher power level.

Response

The core ΔT-power calculator equation was not included because it is independent of fuel type, power level and all other parameters associated with the reload.

QUESTION 19

Describe in detail the methods used by Exxon to incorporate into the set point calculations those transient events which require a low flow trip. Explain how Exxon treats the equivalent of the Combustion Engineering concept of a Required Overpower Margin.

Response

Loss of coolant flow transients are caused by a loss of electrical power to the primary coolant pumps and a corresponding increase in coolant temperature. This increase, combined with the reduced flow, is anticipated to reduce thermal margin (MDNBR). The two most severe transients of this type are: (1) the loss of the four primary coolant pumps, and (2) the loss of two primary coolant pumps in opposite coolant legs. Since these transients result in changes in core power and inlet temperature not covered by the TM/LP function, core protection is provided by the RPS through the low flow trip.

The adequacy of the existing low flow trip was determined by analyzing the above two loss of coolant flow transients for Cycle 6. Those analyses are reported in XN-NF-79-79.

The required overpower margin is defined as that amount of reactor power increase necessary to prevent exceeding the SAFDL on DNBR during a full length CEA drop. The CEA drop results in a non-symmetric core power distribution, and these changes in core peaking are considered in the analysis of the CEA drop. Reactor protection against SAFDL penetration is provided in determining the allowable core power as a function of axial shape index (LCO for DNB monitoring). The method used by ENC is described in detail in the response to Question 23.

QUESTION 20

Discuss in detail the method used by Exxon to incorporate into the set point calculations those transients which require no reactor trip. List all the events considered in this category. Explain the step-by-step procedure that is used to calculate the required protection.

Response

In addition to the loss-of-coolant flow transients (see response to Question 19), the CEA drop transient results in changes in core power and inlet temperature not covered by the TM/LP function. In order to provide core protection against penetration of MDNBR limits (SAFDL on DNBR), the limiting condition for operation (LCO) for DNB monitoring is determined and provided as part of the reactor operating limits.

The CEA drop transient is analyzed using standard ENC predictive methodology for DNBR (XN-75-48) and a thermal hydraulic model which is essentially the same used in the determination of TM/LP (see response to Question 10). Included in the analysis are changes in the core power distribution consequent to the CEA drop. The method used to determine the LCO for DNB monitoring is described in detail in the response to Question 23 and the results presented in the Fort Calhoun Cycle 6 Safety Analysis Report (XN-NF-79-77). Conformance to these limits provides core protection against penetration of MDNBR limits for the CEA drop (no reactor trip) transient.

QUESTION 21

Provide a comparison of ENC set point calculations with those from a previous cycle of Ft. Calhoun as discussed on page 3 of XN-NF-507. Explain the cause of any differences between the two calculations.

RESPONSE

Calculations were performed determining the possible axial power profiles for the Cycle 4 Fort Calhoun core. Methods used were those addressed in the set point document; Reference 5. Sections 4.1.1 and 6.1 of Reference 5 show several parameters calculated for the Cycle 4 core. In the analysis Cycle 4 Technical Specification values were used for the SAFDL's and LCO limits. A comparison of the ENC calculated LCO Axial Power Distribution "Barn" to that of the Fort Calhoun Cycle 4 Technical Specification, LCO, Axial Power Distribution, "Barn" is shown in Figure 5.1 of Reference 5.

Small differences show up in the comparison between the Technical Specification APD LCO curve and that generated with ENC methods. Without additional information about the calculation made to determine the APD LCO curve in the Technical Specifications, the cause of the small differences cannot be explained. Areas of possible differences could exist with the values of the uncertainties used in the two independent calculations, methods, codes, and or models. Specific details of both calculations would be required to resolve the small differences.

Question 22

Provide the curves of P_{fdn} and BOPM versus peripheral axial shape index for the CEA insertons used for the Fort Calhoun Cycle 6 setpoint calculations.

Response

Curves of P_{fdn} and BOPM versus peripheral axial shape index for the CEA insertions were not generated using the ENC setpoint methodology as detailed in the response to Question 24.

QUESTION 23

Describe in detail the method used to calculate the linear heat generation rate and DNB tents (such as Figures and in the Fort Calhoun Technical Specifications).

Response

The LCO for DNB monitoring (DNB tent) represents allowable core power as a function of axial shape index and provides administrative limits to preclude penetration of MDNBR limits during the CEA drop transient. The methodology used by ENC in determining the DNB tent is described below in detail.

o Axial Power Profiles

The core average axial power profiles used in the determination of the DNB tent were calculated using ENC neutronics methodology (XN-75-27) and represent that set of power profiles anticipated during Cycle 6 operation. The power profiles were sorted according to axial shape index. Within each small ASI increment, sufficient axial profiles were selected for analysis to ensure proper determination of the DNB tent.

o Definition of Analysis Input

The establishment of the DNBR analysis input includes allowance for reactor operating uncertainties, nuclear peaking uncertainties, and changes in peaking during the CEA drop. The core operating conditions used in the analysis were 2053 psia (minimal operating pressure minus 22 psia uncertainty), 547°F core inlet temperature (545° + 2°F uncertainty), 1530 MWt total core power (1500 + 2% power uncertainty).

The effect of the CEA drop upon hot assembly conditions was determined to result in increased nuclear peaking and decreased assembly flow. A peaking

augmentation of 121% was determined from the neutronics analysis of the worst CEA drop, and a reduction of 6% in the hot assembly flow was determined from appropriate core flow distribution calculations. The accounting for the above two effects was directly applied in the DNBR analysis.

The average LHGR associated with the limiting assembly was calculated according to the following expression:

$$LHGR = \frac{\overline{LHGR} * F_R^T * F_M^U * F_{AVG}}{F_\ell * F_G^U * F_P^U}$$

where,

$\overline{LHGR}$ = core average linear heat generation rate

F_R^T = peaking limits as defined by the Technical Specifications

F_M^U = peaking measurement uncertainty (1.06)

F_{AVG} = calculated CEA drop peaking augmentation (1.21)

F_ℓ = assembly local peaking factor (1.09)

F_G^U = calculational factor to account for changes in nominal hot subchannel area due to allowed manufacturing tolerances (.988)

F_P^U = core power measurement uncertainty (.98)

The above expression results in an LHGR reference (LHGR ref.) value of 11.46 at 100% of rated power.

o Power Interation

The limiting hot assembly LHGR is defined as that value which results in a calculated MDNBR = 1.306 ($LHGR_{crit}$). The 1.306 value is the product of the

SAFDL on DNB, engineering heat flux factor, and a correction factor to account for that fraction of total generated energy which appears as rod surface heat flux (1.036 = 1.30 * 1.03 * 0.975).

The allowable core power fraction, P, is then defined as the ratio of $LHGR_{crit}$ to $LHGR_{ref}$ for each ASI value. The ordered pair (ASI, P) defines a point on the unadjusted DNB tent. The ASI value associated with each such point is then adjusted for neutronics uncertainty and for the excore calibration uncertainty associated with the determination of the ASI value in accordance with XN-NF-507. The DNB tent as reported in XN-NF-79-77 is the loci of points thus generated.

The LCO for linear heat generation rate (LHGR) monitoring, LHGR tent, represents operating limis for allowable core power, (P) as a function of Axial Shape Index (ASI) for those times during the cycle in which the incore detectors are inoperable. The peak LCO LHGR being monitored is determined by calculating the minimum allowable LHGR resulting from either the increase in power associated with a dropped rod or the LOCA LHGR limit. The methodology used by ENC in determining the LCO LHGR tent is described below.

AXIALS

The axial power distributions used in the determination of the LCO LHGR tent were the same axials calculated and used in the LSSS APD calculation. Methods used to determine these axials are discussed in the response to Question 2. Approximately 1500 axial power profiles were used in the analysis for the LHGR LCO tent as well as the LSSS APD tent.

INPUT AND CALCULATION

Input to the LCO LHGR tent analysis includes F_Z values, all uncertainties associated with the LCO's, the Technical Specification values of F_r and F_{xy}, the core average LHGR, and the minimum of either the maximum LHGR limit that would protect the core from violating the peak kw/ft for centerline melt during a dropped rod incident or the maximum LOCA LHGR limit. The calculation performed to determine the percent allowable power as a function of ASI is

therefore:

$$P(ASI) = \% \text{ allowable power} = 100 * K/F_Q^T * \overline{LHGR} - Pm$$

$\overline{LHGR}$ = Core average linear heat rate.

where K = LHGR Limit

$F_Q^T = F_{xy} * F_Z\ (ASI) * F_U^L$

Pm = Power Measurement Overshoot = 2.0

F_{xy} = ratio of hot pin to average pin at core elevation Z.

$F_Z(ASI')$ = axial power peaking factor for axial shape index ASI'.

F_U^L = Uncertainties = $F_c * F_E * F_{LHGR} * F_{aug}$

F_C = Calculation = 1.07

F_E = Engineering = 1.03

F_{LHGR} = Linear Heat Rate = 1.005

F_{aug} = Augmentation Factors (Table A.1 of Reference 11 if dropped rod limited and 1.0 in LOCA limited)

ASI = ASI - ASIu

ASIu total .04 ASI units

(No trip system processing, Reference 5, since no reactor trip associated with LCO's.)

The above expression gives ordered pairs of (P, ASI) which results in the LHGR LCO tent for Fort Calhoun.

Response

Parts 1 through 8 of Section 6.2.3 (XN-NF-507) describe the construction of TM/LP safety limit lines. These lines are isobars on a plot of coolant inlet temperature versus allowable core power (Figure 24.1). Each isobar defines as a function of power the lowest coolant inlet temperature which precludes violation of one of the following limits:

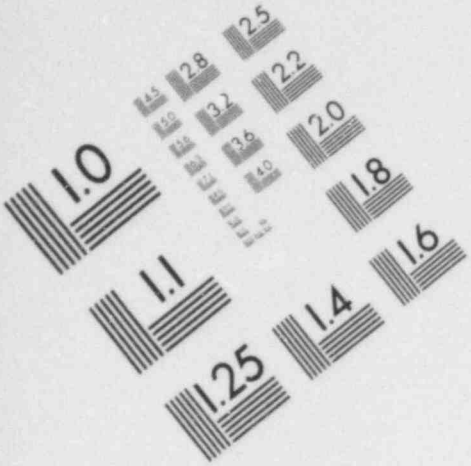
- o MDNBR $\leq$ 1.35
- o Maximum hot channel quality $\geq$ 15%
- o Core average exit temperature $\geq$ saturation temperature

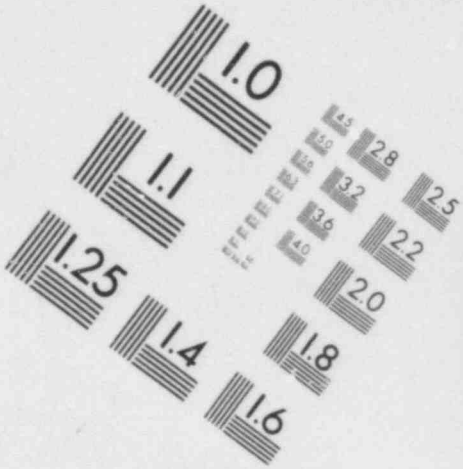
The TM/LP safety limit lines thus define steady-state temperature, pressure, and power conditions which may not be exceeded without violating the SAFDL on DNB.

(a) Selection of Worst Axial Power Distribution

An XCOBRA-IIIC calculation is performed using each of the axial shapes under consideration. The power, inlet temperature, and pressure are fixed according to Steps (1), (2) and (3). Only the axial profile is varied. The worst axial profile at the selected power is the one which yields the least MDNBR or other appropriate SAFDL value in the XCOBRA-IIIC calculation. Table 24.1 lists the worst axials determined as above are used to construct the TM/LP safety limit lines.

(b) Steps (1) through (6) establish a single point on a safety limit line. A second point on that isobar may be obtained by selecting a new power (and thus a new worst axial) and again performing a series of iterative XCOBRA-IIIC calculations to determine the limiting coolant inlet temperature. System pressure is constant in all XCOBRA-IIIC calculations used to generate the points on a single safety limit line.





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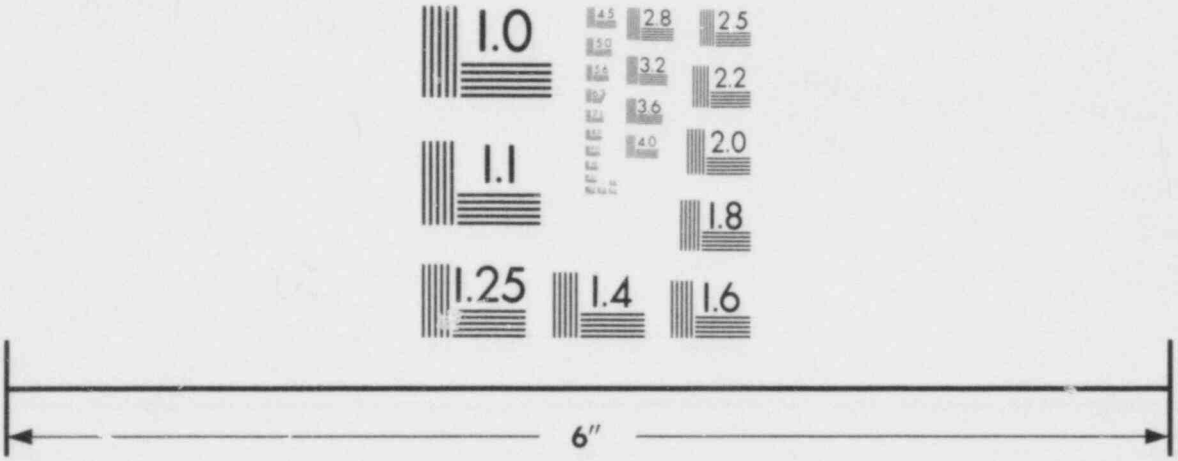
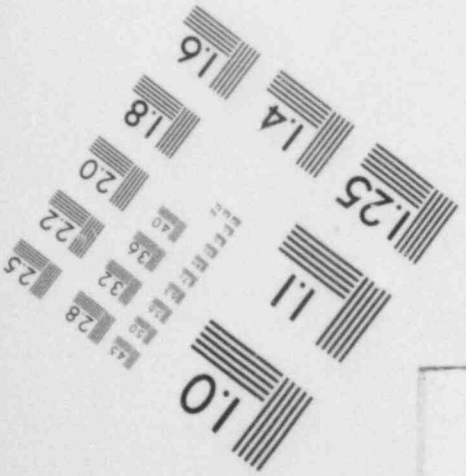
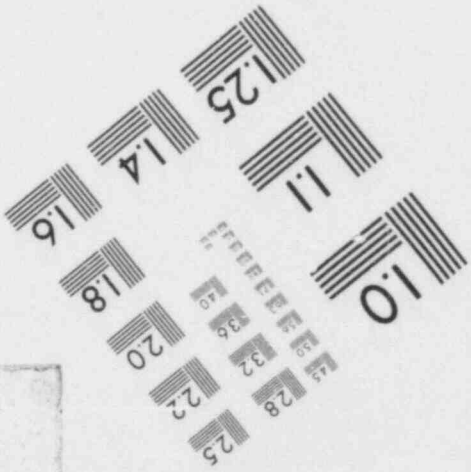
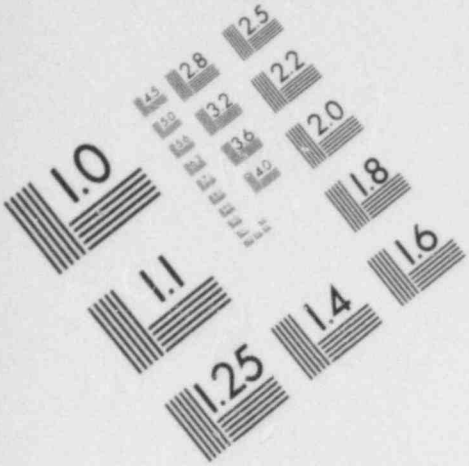


MICROCOPY RESOLUTION TEST CHART







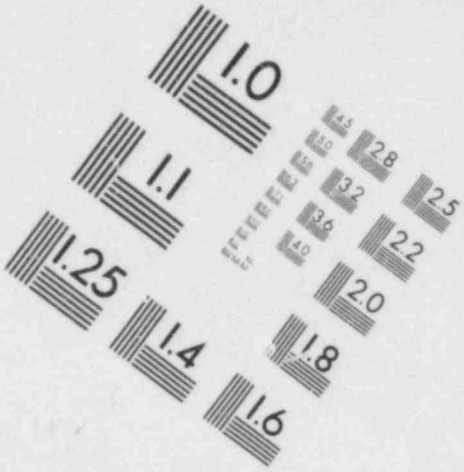
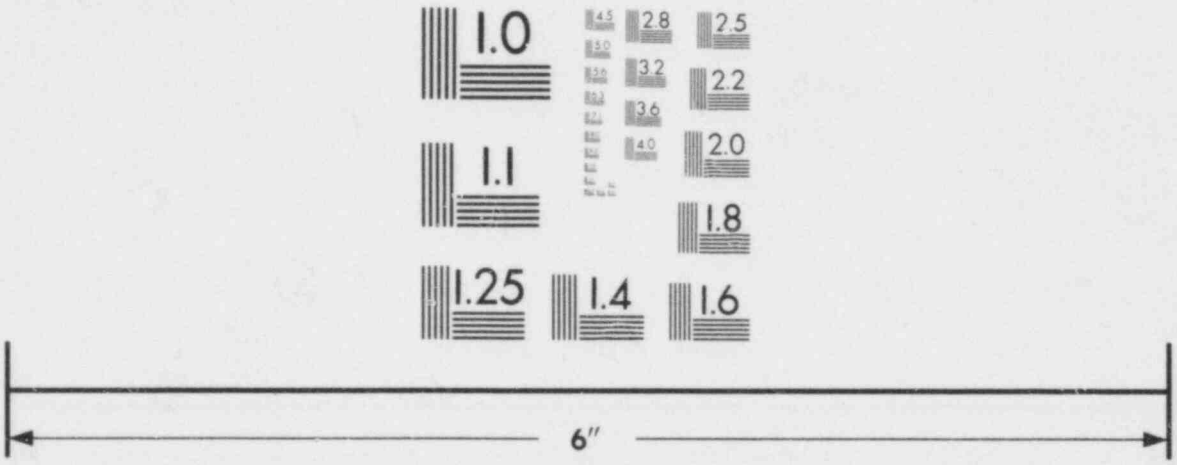


IMAGE EVALUATION
TEST TARGET (MT-3)



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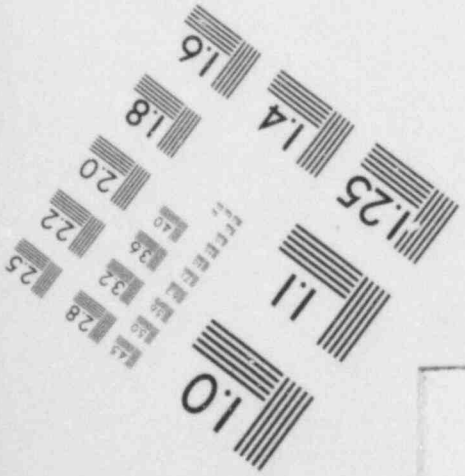
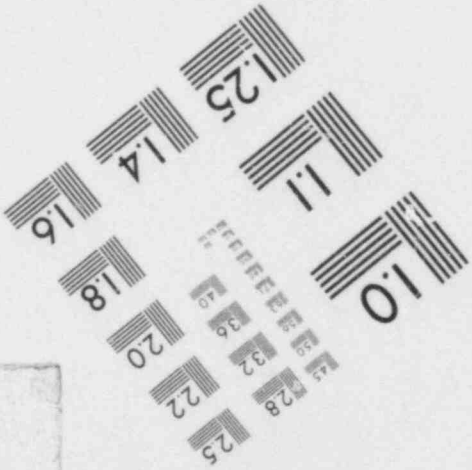
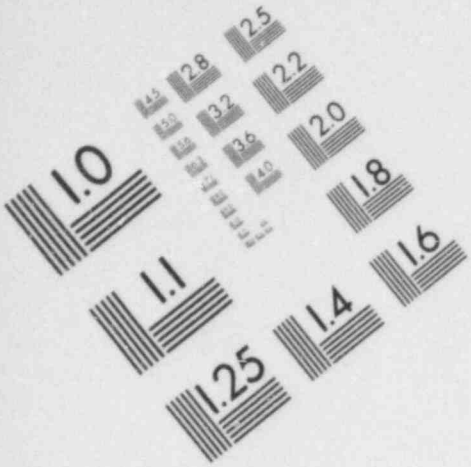






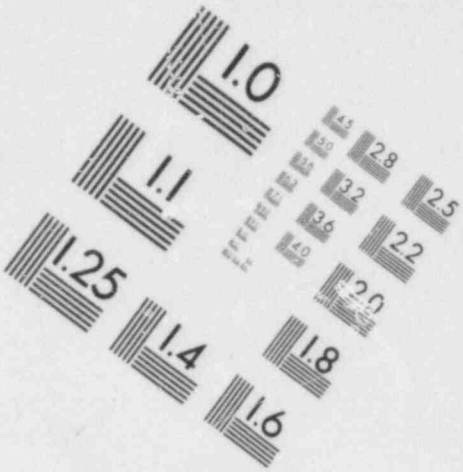
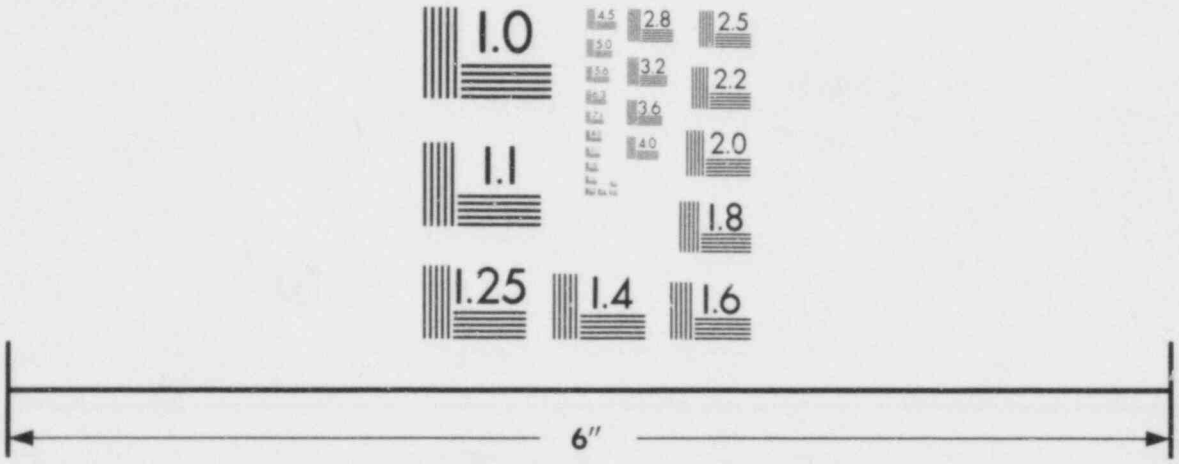


IMAGE EVALUATION
TEST TARGET (MT-3)



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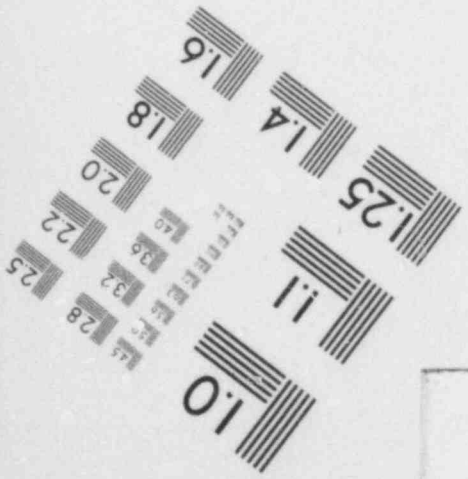

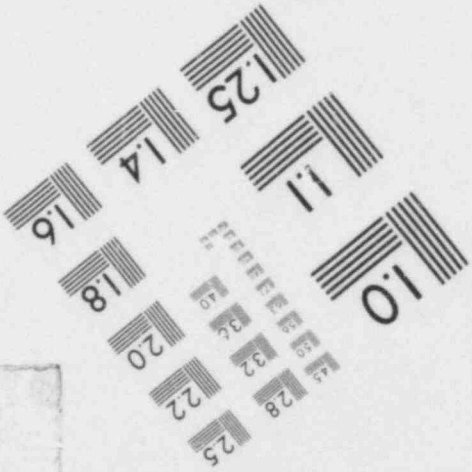

TABLE 24.1

LIMITING AXIAL PROFILES FOR TM/LP CONSTRUCTION

X/L is Axial Coordinate Normalized to the Active Fuel Length

(X/L)	@ 112% Power	@ 100% Power	@ 85% Power	@ 65% Power
.0208	.624	.342	.230	.171
.0625	.885	.494	.334	.249
.1042	.985	.574	.393	.296
.1458	.991	.615	.433	.330
.1875	.973	.655	.478	.371
.2292	.956	.694	.530	.420
.2708	.945	.736	.592	.479
.3125	.941	.782	.663	.549
.3542	.942	.831	.740	.629
.3958	.949	.882	.824	.719
.4375	.962	.936	.909	.816
.4792	.978	.991	.992	.920
.5208	.999	1.046	1.071	1.029
.5625	1.021	1.104	1.137	1.141
.6042	1.047	1.162	1.211	1.255
.6458	1.075	1.222	1.292	1.368
.6875	1.104	1.281	1.377	1.478
.7292	1.134	1.343	1.466	1.586
.7708	1.164	1.405	1.553	1.689
.8125	1.169	1.463	1.634	1.782
.8542	1.196	1.503	1.691	1.847
.8958	1.166	1.503	1.698	1.857
.9375	1.039	1.390	1.573	1.723
.9792	.736	1.046	1.181	1.296

Sufficient points are generated at a fixed pressure, plotted, and connected by a piecewise smooth curve which represents a single safety limit line.

(c) The procedures outlined above are repeated at other pressures to obtain the set of TM/LP safety limit lines. The reference to Figure 5.1 is in error and should refer to a set of TM/LP safety limit lines obtained in accordance with the steps outlined above.

(d) The TM/LP safety limit lines are fit with a function of the form

$$P = \alpha\, PF(B)B + \beta\, T_{in} + \gamma_1 \qquad (1)$$

The fit is determined to conservatively envelope the TM/LP safety limit lines (see Figure 24.2). Equation (1) is <u>not</u> the TM/LP trip equation. The final TM/LP trip (TM/LP LSSS) equation is obtained from (1) by writing

$$P_{var} = \alpha\, PF(B)B + \beta\, T_{in} + \gamma \qquad (2)$$

where

$$\gamma = \gamma_1 + b \qquad (3)$$

and b is a positive constant which converts the TM/LP safety limit function to the TM/LP trip function and provides protection against penetration of the appropriate SAFDL during anticipated transients. The values of α, PF(B), and β are carried over to equation (2) without change. The procedure used to obtain α, β, γ and PF(B) is detailed in the following.

The value of β is calculated from two TM/LP safety limit lines as follows:

(1) Select a pair of contiguous safety limit lines. For demonstration, select the 2100 psia and 2250 psia isobars. Then the value of $P_{var_1} - P_{var_2}$ in Step (10) is given by:

$$P_{var_1} - P_{var_2} = 2250 - 2100 = 150 \text{ psia}$$

(2) Determine the value of T_{in} allowed by each safety limit line at 100% power (Figure 24.1). For the demonstration case,

$$T_{in_1} = 567.3°F$$

$$T_{in_2} = 560.3°F$$

and $T_{in_1} - T_{in_2} = 567.3 - 560.3 = 7°F$

(3) A value of β is then computed according to the formula in Step (10):

$$\beta = \frac{P_{var_1} - P_{var_2}}{T_{in_1} - T_{in_2}} = \frac{150 \text{ psia}}{7°F} = 21.43$$

Steps (1) through (3) above are repeated for each pair of contiguous safety lines in Figure 24.1. The value of β is finally fixed equal to the arithmetic average of the values thus obtained.

The value of α is calculated as follows:

(1) Select a pair of contiguous safety limit lines. For demonstration, select 2100 psia and 2250 psia isobars.

(2) Calculate the value of β associated with those two safety limit lines, as described above (β = 21.43).

(3) Calculate the slope of each safety limit line in the straight line portion:

(a) for the 2100 psia isobar,

$$\text{slope} = \frac{\Delta T}{\Delta B} = -\frac{560.3 - 557.4}{112 - 100} = -.2417$$

(b) for the 2250 psia isobar,

$$\text{slope} = \frac{\Delta T}{\Delta B} = -\frac{567.3 - 564.}{112 - 100} = -.2750$$

(4) Compute a value of α as the product of the appropriate β (Step 2 above) and the arithmetic mean of the absolute values of $\frac{\Delta T}{\Delta B}$ found in Step (3) above:

$$\alpha = 21.43\ [.2417 + .2750](1/2) = 5.54$$

(5) Repeat Steps (1) through (4) above for each pair of contiguous safety limit lines in Figure 24.1. The final value of α is fixed equal to the arithmetic average of the values thus obtained.

The value of γ_1 is chosen to yield a function which conservatively envelopes all the safety limit lines. This is accomplished as follows:

(1) Select a safety limit line. For demonstration, choose the 1750 psia isobar.

(2) Determine from Figure 24.1 the allowed inlet temperature at the pressure selected in (1) and 100% power. For the demonstration case,

$$T_{in} = 545°F$$

(3) Write equation (1) at the conditions

$$T_{in} = 545°F \qquad \alpha = 5.54$$

$$B = 100 \qquad \beta = 22.48$$

$$P = 1750 \text{ psia}$$

and solve for γ_1 to get:

$$\gamma_1 = P - \alpha B\, PF(B) - \beta T_{in}$$

$$= 1750 - 554 - 22.48(545)$$

$$= -11{,}055.6$$

The function thus developed is a closed form representative of the TM/LP safety limit lines. The TM/LP trip equation differs from that function only in the value of the additive constant. The value of γ in equation (2) is selected to protect the SAFDL on DNB during anticipated transients and to allow a reasonable administrative operating band. To obtain the value of γ, equation (2) is rearranged and evaluated at conditions consistent with those requirements, as shown below.

$$\gamma = P_{var} - \alpha\, PF(B)\, B - \beta T_{in}$$

$$= 2075 - 5.54 \times 100.5 - 22.48 * 548$$

$$= -10{,}801.$$

where,

2075 is the least rated pressure allowed at full power operation

100.5 is the maximum operating power allowed by the LCO for DNB monitoring

548°F is the rated coolant inlet temperature at full source operation plus a 3°F offset to allow administrative control of reactor operating variables.

The TM/LP trip function is

$$P_{var} = 5.54\ PF(B) * B + 22.48 * T_{in} - 10{,}801.$$

The value of b in equation (3) may be computed as 255 psi. About 55 psi of that are allocated as uncertainties, leaving 200 psi margin between reactor trip and penetration of the SAFDL on DNB as represented by the TM/LP safety limit lines. That 200 psi is equivalent to a 36 percent overpower margin.

The curve fitting described above might have been accomplished via least squares analysis. The present method has at least two advantages over that alternative: 1) simplicity and ease of application, and 2) direct derivation from the physical meaning of the parameters.

(e) The function PF(B) was selected by trial and error to provide a safety limit function* which conservatively envelopes the TM/LP safety limit lines. Variations in axial shape defined by APD were modeled explicitly in the calculations which established the safety limit lines and are therefore implicit in PF(B). No explicit relationship between this function and allowed axial flux shapes is necessary within the context of ENC methodology.

(f) A complete discussion of procedures used to verify the adequacy of the TM/LP trip function is contained in XN-NF-79-79 and XN-NF-79-79, Supplement 1, "Fort Calhoun Cycle 6 Reload Plant Transient Analysis Report."

(g) The response to Question 15 discusses the application of appropriate peaking factors used to construct the TM/LP trip function.

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* See equation (1), response to Question 24(d).

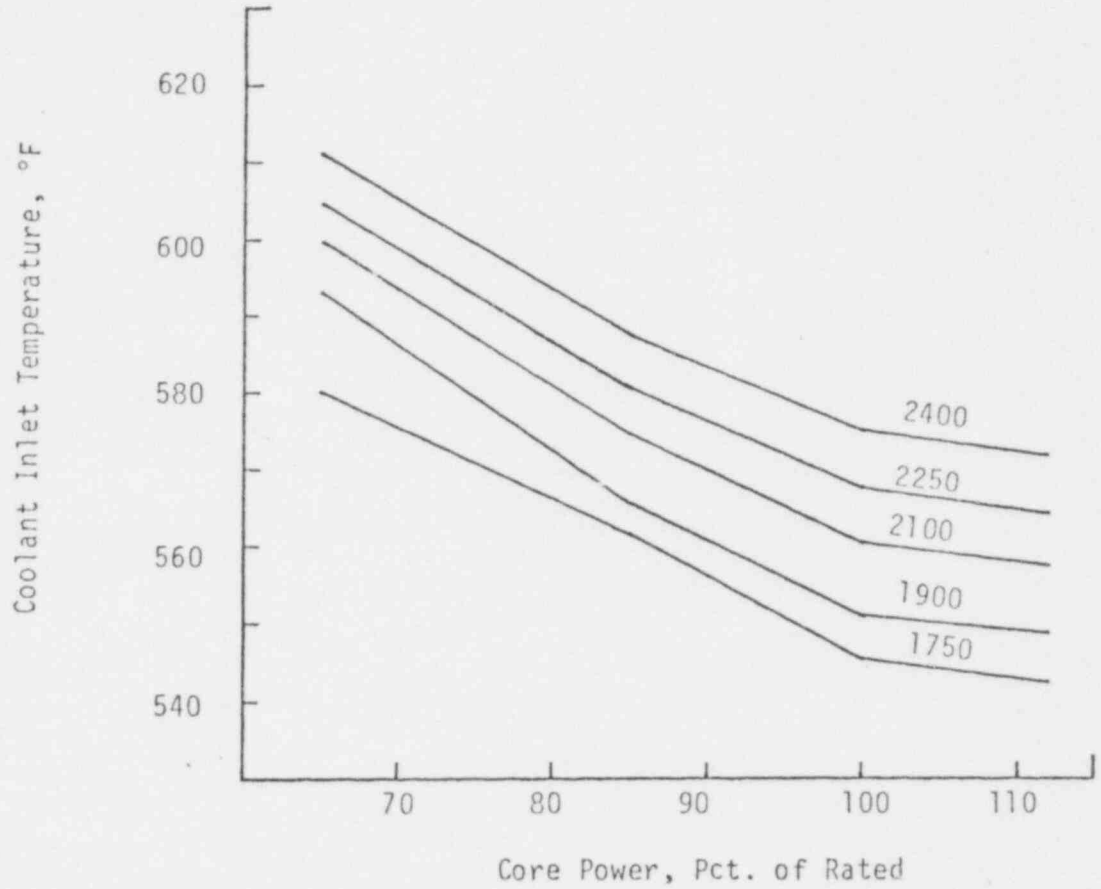

FIGURE 24.1 TM/LP Safety Limit Lines for 1500 MW, 4-pump Operation

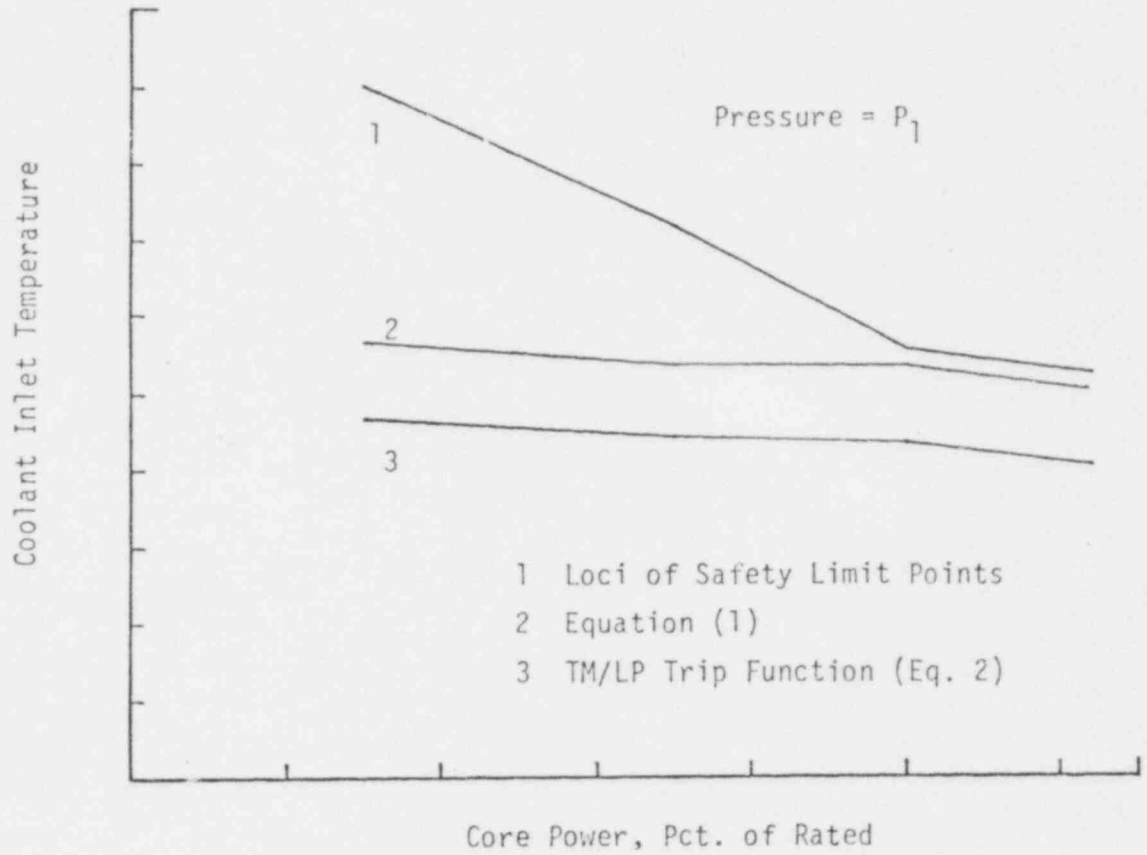

FIGURE 24.2 Comparison of Safety Limits (1) with (2) A Closed Form Function which Conservatively Envelopes them and (3) The TM/LP Trip Function

QUESTION 25

Discuss in detail how the design differences of different types of fuel bundles in a reload core are included in the set point calculations. In particular, differences in fuel bundles manufactured by two different fuel vendors (e.g., Exxon and Combustion Engineering for the Ft. Calhoun Cycle 6 reload) should be addressed. How does Exxon model fuel bundles of another manufacturer? Examples of possible areas of possible insufficient information are

(1) fuel densification characteristics

(2) fuel bundle flow distribution characteristics

(3) fuel rod internal fill gas pressure

Response

The differences in fuel design between the ENC and existing fuels were accounted for in the determination of the APD, TM/LP and LCO for DNB monitoring for Cycle 6.

Local power augmentation as a result of axial densification must be accounted for in the determination of fuel pellet peaking as used in the determination of the APD barn. The peaking augmentation for the ENC fuel was determined and compared against the values currently stipulated for the existing Ft. Calhoun fuel. The larger peaking augmentation as a function of axial height between the ENC and existing fuel was chosen as the appropriate value in determining the Cycle 6 APD barn.

The thermal hydraulic models used to calculate the TM/LP and LCO for DNB monitoring explicitly modeled the hydraulic performance of both fuel types in order to determine the appropriate limiting subchannel flow. The determination of the limiting assembly flow, and subsequent MDNBR, is accomplished in two steps: (1) Core flow distribution to determine the limiting assembly flow rate, and (2) Limiting assembly calculation for evaluation of the core thermal margin (MDNBR).

The core flow distribution calculation directly models the thermal and hydraulic performance of each fuel type as appropriate single hydraulic channels. The thermal performance is evaluated using ENC neutronics methods to determine the core and assembly peaking distribution while the hydraulic performance is determined using the results of pressure drop testing performed by ENC for both fuel types. The results of the calculations indicate that both limiting fuel assemblies will experience no less than 5% of core average assembly flow rates with the ENC assembly having slightly lesser flow. Thus, the limiting ENC fuel assembly was selected for TM/LP and LCO calculations.

The limiting assembly calculations model the limjting ENC fuel assembly into appropriate subchannels with the assembly flow rate as determined above. The calculation is consistent with the methodology used for the core flow distribution calculations. This calculation results in determining the limiting subchannel flow rate used in the ensuing MDNBR calculation (see response to Question 23).

QUESTION 26

Discuss protection provided by set points to asymmetric transients (such as those affecting only one steam generator).

Reference

The loss of load to one steam generator was analyzed using the same plant transient simulation model for the analysis of other plant transients (XN-NF-79-79). In this analysis, all proposed Cycle 6 LSSS trip functions were explicitly modeled. The results of the analysis were documented and transmitted as supplementary information in support of the Cycle 6 Licensing Application (Letter Report from W. C. Jones to R. Reid dated December 4, 1979). The results of the analysis indicate that adequate core protection is provided for Cycle 6 with the proposed set points to preclude penetration of the appropriate SAFDL for Cycle 6.

Question

What equations are used to calculate the DNBR used in the XCOBRA code?

Response

The equation of the W-3 correlation along with the equations for the non-uniform axial heat flux factor and the unheated boundary factor are given in XN-75-45. The procedure for using the W-3 correlation and correction factors is also given.

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