XN-NF-77-22

STEADY-STATE THERMAL HYDRAULIC AND NEUTRONICS ANALYSIS OF THE PALISADES REACTOR FOR OPERATION AT 2530 MW_T

JULY 15, 1977

RICHLAND, WA 99352

EXXON NUCLEAR COMPANY, Inc.

Issue Date: 07/15/77 XN-NF-77-22

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1.0 INTRODUCTION

This report presents the evaluation of the thermal hydraulic and neutronics performance of the Palisades core and Cycle 2 Reload fuel types at the core power level of 2530 MW_t. The nominal primary coolant operating conditions selected for this analysis are 2060 psia and 537°F core inlet temperature.

The Exxon Nuclear Company (ENC) fuel assemblies are designed to be compatible with the Palisades reactor core and with Batch D fuel. This was demonstrated in the Thermal Hydraulic Analysis Report submitted in support of the Cycle 2 license application (XN-76-3 [P]) and Supplement 1. The reload fuel assemblies are designed to operate at a core power of 2650 MW_t steady-state and anticipated transient conditions. The mechanical integrity of the fuel at pressure up to 2100 psia, at nominal core inlet temperatures up to 543°F, and at peak fuel rod powers very nearly the same as will be experienced at 2530 MW_t was reported in XN-76-52, submitted in support of operation at 2100 psia. Operation at 2100 psia has been reviewed and approved by the NRC.(1)

The limiting fuel types for the analysis are Batches D and E fuels. Batch D fuel was supplied by Combustion Engineering. Batches E and F fuels were supplied by Exxon Nuclear Company. Batch F fuel is 1.5% enriched and has low power generation and is therefore not limiting. Batch E fuel is representative of future anticipated ENC Palisades reload deisgn. It is anticipated that future reload fuel performance will be enveloped by this analysis.

The Palisades reactor core is currently licensed at 2200 MW_t. Operation at 2530 MW_t with adequate thermal margins is principally achieved by reduction of allowed nuclear peaking factors; the peak linear heat generation rate at stretch power is nearly the same as for the currently licensed 2200 MW_t.

Analysis of the neutronics characteristics of the Cycle 2 core at 2530 MW_t and an exposure of 15 GWD/T are presented in Section 5.0 of this report.

2.0 SUMMARY

The required reactor thermal margins and thermal margin trip limits to maintain MDNBR $\geq$ 1.3 (W-3 correlation) during steady-state and transients for the limiting D and E fuels is presented in this document. The steady-state thermal margin analysis was performed at 115 percent of 2530 MW_t (2910 MW_t) to provide margins for the transients. On the basis of core conditions including power peaking given in Table 2.1, the MDNBR at over-power was calculated to be 1.30. This indicates that 15 percent overpower results in an MDNBR lower than calculated for the most restrictive anticipated plant transient, i.e., 1.35.[2]

The method of analysis employed are consistent with prior ENC submittals for Palisades, and the application of the W-3 DNB correlation is consistent with the methods described in Reference (3). A summary of the steady-state DNB analysis results are presented in Table 2.1. The table shows the nuclear peaking and engineering factors assumed in the analysis for fuel types D and E. This table also compares the limits to those derived in support of Cycle 2 operation at 2200 MW_t.

Steady-state plant thermal margins were determined parametrically as a function of power, core inlet temperature, and primary system pressure. The results of this analysis are given in Section 3. These limits were used to derive set point equations as prescribed in the plant technical specifications. These set point equations are required in performing the plant transient analysis as report in XN-NF-77-18. The results of the transient analysis indicate that adequate thermal margins are achieved to protect an MDNBR $\geq$ 1.3 during anticipated transients (Class 2 transients).

The neutronics characteristics of the Palisades Cycle 2 core have been evaluated. These characteristics are such that routine procedures for maintenance of thermal margins enable safe operation of the core at power levels up to 2530 MWt and Cycle 2 average core exposures up to 15 GWD/MTU.

TABLE 2.1

SUMMARY OF DNB ANALYSIS

Design Reactor Core Conditions	2530 (MW_t)	2200 (MW_t)
Design Overpower, (MW_t)	2910	2684
Nominal Total Core Flow Rate, (10^6 lb_m/hr)	121.7	124
Nominal Active Core Flow Rate, (10^6 lm_m/hr)	114.4	116.8
Primary Pressure (psia)	2060	2100
Core Inlet Temperature (°F)	537	543.
Core Pressure Drop (psia)	13.5 ± 0.5	13.5 ± 0.5
Fuel Bundles in Core	204	204
Average Linear Heating Rate, (kw/ft)	5.3	4.6
Maximum Linear Heating Rate,(kw/ft) nominal	13.8	13.9
*Maximum Linear Heating Rate, (kw/ft) at overpower	15.9	17.0
Fraction of Heat Generated in Fuel	0.975	0.975
MDNBR (at overpower)	1.3	1.3
Nuclear Peaking Factors	2530	2200
Radial	1.45	1.6
Axial	1.4	1.5
Local	1.22	1.21
Engineering	1.03 (E)	1.03 (E)
	1.05 (D)	1.05 (D)
Total	2.55 (E)	3.05
	2.60 (D)	

* The lower maximum linear heating rate at the higher core power is accounted for by the allowance of lower nuclear peaking factors at stretch power.

TABLE 2.1 (continued)

Fuel Bundle Description

	Batch D	Batch E
Rod Diameter	0.4175 in.	0.415 in.
Rod Pitch	0.55 in	0.55 in
Active fuel length	131.4 in	131.8 in.
Number of active rods	216	208
Number of poison rods	0	8
Instrument tubes	1	1
Number of guide bars	8	8
Total positions	225	225
Number of Spacers	10	10
Average heating rate at 2530 MW_t	5.244 kw/ft	5.429 kw/ft
" " " " 2200 MW_t	4.560 kw/ft	4.721 kw/ft

3.0 THERMAL HYDRAULIC PERFORMANCE

The evaluation of the thermal hydraulic design of the Palisades Reactor core and the ENC reload fuel was performed with the XCOBRA-IIIC[4] thermal hydraulic computer code employing the W-3 critical heat flux correlation. A hydraulic model of one-eighth of the Palisades Reactor core (see Figure 3.1) for a typical core loading pattern was used to calculate the flow to each fuel assembly. The effect of a 5% lower plenum flow maldistribution was included in this analysis. A separate subchannel hydraulic model (see Figure 3.2) consisting of an octant of the individual fuel assembly was used to determine the MDNBR of limiting fuel bundles. The design basis axial heat flux profile, the bundle radial power distribution, and the bundle local power distribution used in the hydraulic models were based upon neutronic calculations. The core flow rate was determined as described in paragraph 3.3.

The results of the MDNBR analysis with the design basis axial power distribution and at reactor design overpower (115 percent) show that the MDNBR of the limiting ENC and CE fuel bundles are never less than 1.3 at the design operating conditions. Reactor conditions and assembly characteristics considered in this analysis are presented in Table 2.1.

3.1 HYDRAULIC CHARACTERISTICS

The hydraulic characteristics of the ENC reload fuel assemblies used in the hydraulic models are based on experimental results from hydraulic tests on fuel assembly components that are similar to those specified for the Palisades Reactor fuel assemblies. Hydraulic characterization includes single-phase loss coefficients for:

- Lower tie plate and core support (combined)
- Grid spacers
- Upper tie plate and holddown plate (combined)
- Bare rod friction.

The loss coefficients as measured for both ENC fuel assemblies and Combustion Engineering assemblies were used in the analysis of each fuel assembly type in the XCOBRA-IIIC hydraulic models.

3.2 HYDRAULIC COMPATIBILITY

The hydraulic compatibility of the ENC Reload fuel with the Palisades Reactor is measured in part by the impact upon the core flow distribution of a mixed core loading of existing and reload fuels. The split in flow among fuel assemblies is due to the different hydraulic resistances of the various assemblies as reflected in their individual hydraulic characteristics. The flow rate to the hot channel is strongly dependent on the assembly radial power (higher power generally results in lower flow rates) while the variation in axial power profile among bundle has negligible effects. Thus, the same axial power profile was used for all assemblies while the hot bundle was assigned the design radial power factor. The lower plenum inlet flow was assumed to be nonuniform, and the flow to the core region with the highest power assembly was 5% less than core average inlet flow.

The flow distribution for a typical loading pattern of D and E fuels showed less than one percent variation in bundle mass flows from core average for a nominal power distribution and uniform inlet flow. The combination of the maximum design radial bundle power and five percent less than core average inlet plenum mass flow on highest power (F_R = 1.45) E and D assemblies was used in predicting hot bundle flow factors. These flow rates were then applied when modeling the limiting fuel assembly.

3.3 MINIMUM DEPARTURE FROM NUCLEATE BOILING RATIO (MDNBR)

The MDNBR is a function of limiting bundle power, primary system pressure, coolant inlet temperature, and coolant flow rate. The primary coolant flow rate was determined by a conservative relationship that accounted for the changing coolant density with coolant temperature (including core power) conditions based on measured flow at HZP. The effective core flow rate was reduced by 3% to account for core bypass flow and a further 3% to account for uncertainty in flow measurements. The bundle flow rate is finally reduced in accordance with the applicable hot bundle flow factor as described in Section 3.2. The XCOBRA-IIIC limiting fuel assembly model was evaluated at 115 percent of 2530 MW_t power and design radial to find the MDNBR as a function of coolant inlet temperature and primary system pressure. A conservative value of the core inlet temperature, 5°F above nominal, as well as a conservative value of primary pressure, 50 psia below nominal, were used in the analysis. These conservatisms account for unfavorable impact on the thermal margin of measurement uncertainties and normal operating fluctations of temperature and pressure.

3.4 THERMAL MARGIN

For nominal operations, the limiting values of primary coolant pressure, reactor inlet temperature, and reactor power level are defined for a broad range of each parameter for which the thermal criteria are not exceeded. The limits of operation are designed to assure:

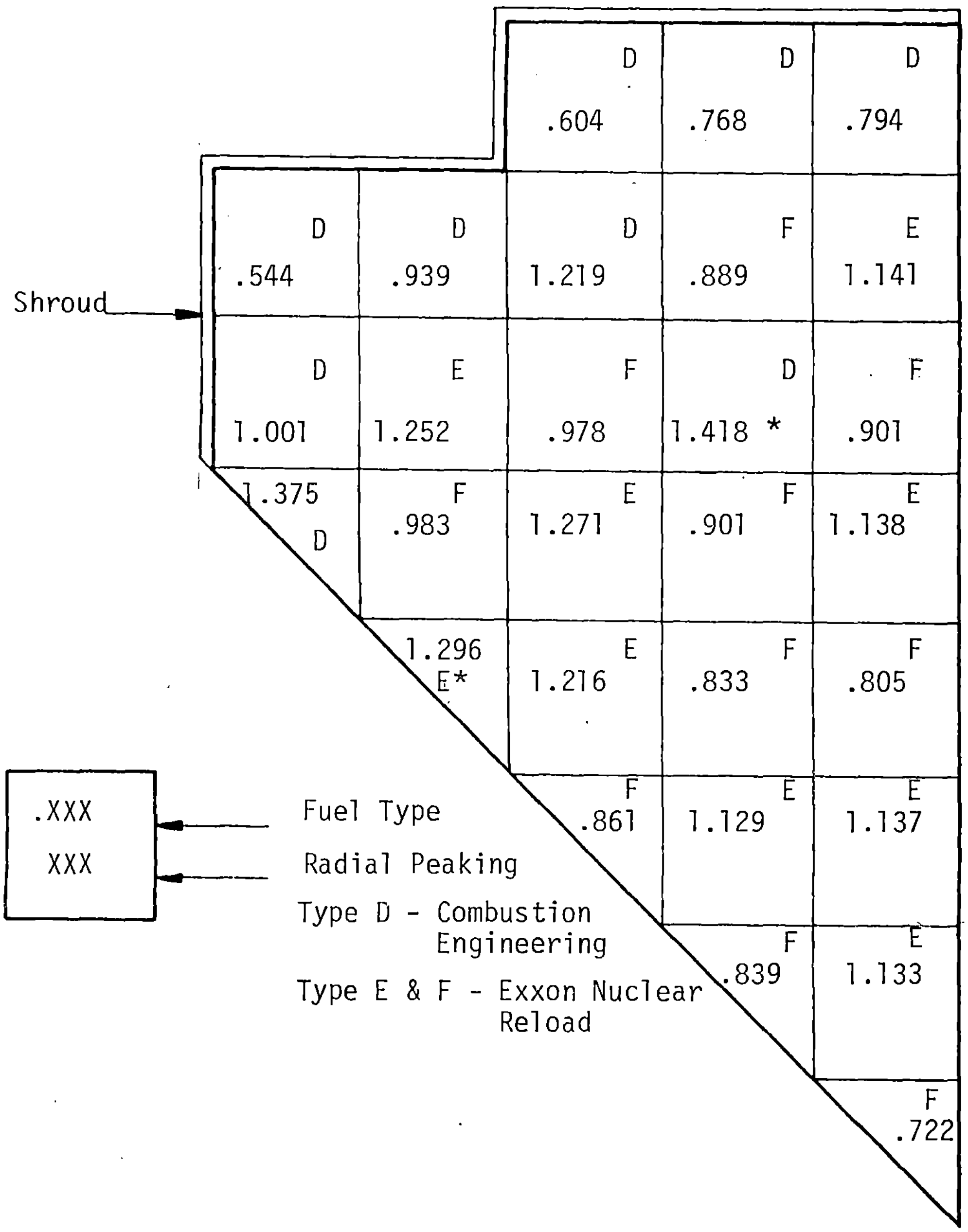
- MDNBR $\geq$ 1.3

- Quality of primary coolant $\leq$ 15 percent at MDNBR location
- Flow stability.

The derivation of the thermal margin curves was accomplished by the utilization of the XCOBRA-IIIC subchannel hydraulic model. The core operating conditions (primary system pressure, core inlet temperature, and core power) were parametrically analyzed so that the combinations of the above primary system parameters that resulted in MDNBR = 1.3 were determined. The assembly mass flow rate was determined based on its dependence on core inlet temerature and power with appropriate adjustments for core bypass, measurement uncertainty, and the hot bundle flow factor. The design nuclear peaking factors were assumed throughout the analysis. The reduction of slope of the curves at low power and low pressure (see Figure 3.3) is due to the occurrence of increased coolant quality limiting the allowed core inlet temperature rather than MDNBR (i.e., W-3 correlation is valid only up to 15% local coolant quality).

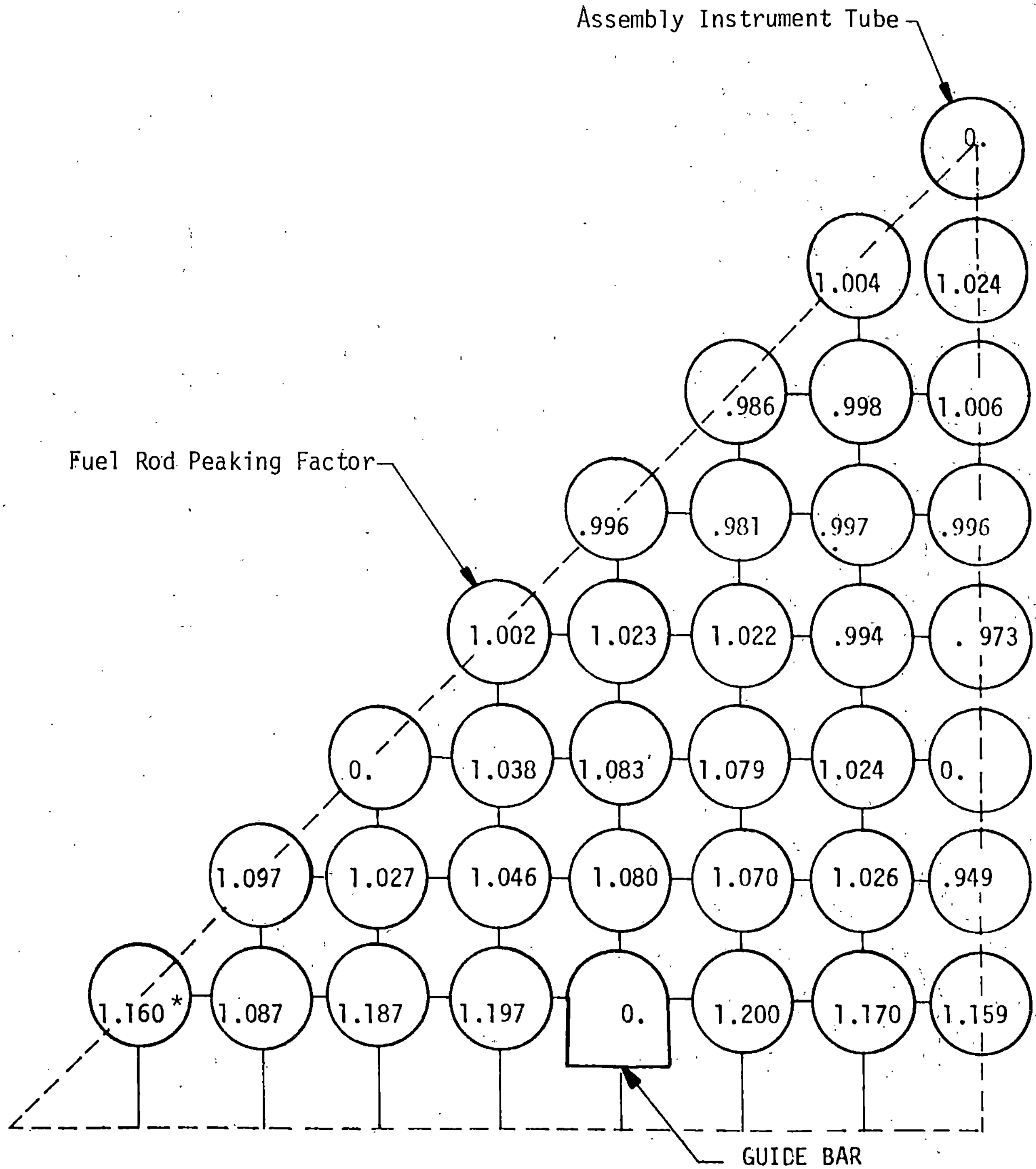
Current low primary coolant flow and low primary system pressure trip points are adequate to prevent the possibility of DNB resulting from local flow oscillations. A parallel channel flow stability analysis was performed. The results of this analysis showed that within the allowed technical specification operating limits no flow instability will occur.

The limit lines shown on Figure 3.3 are provided for the formulation of operating thermal set points in an algorithm compatible with the reactor safety system parameters.



* Raised to design radial peaking (1.45)

FIGURE 3.1 CORE POWER FOR DISTRIBUTION FOR DETERMINATION OF CORE FLOW DISTRIBUTION



* Increased to 1.22 for this analysis.

FIGURE 3.2 PALISADES TYPE E ASSEMBLY SUBCHANNEL MODEL

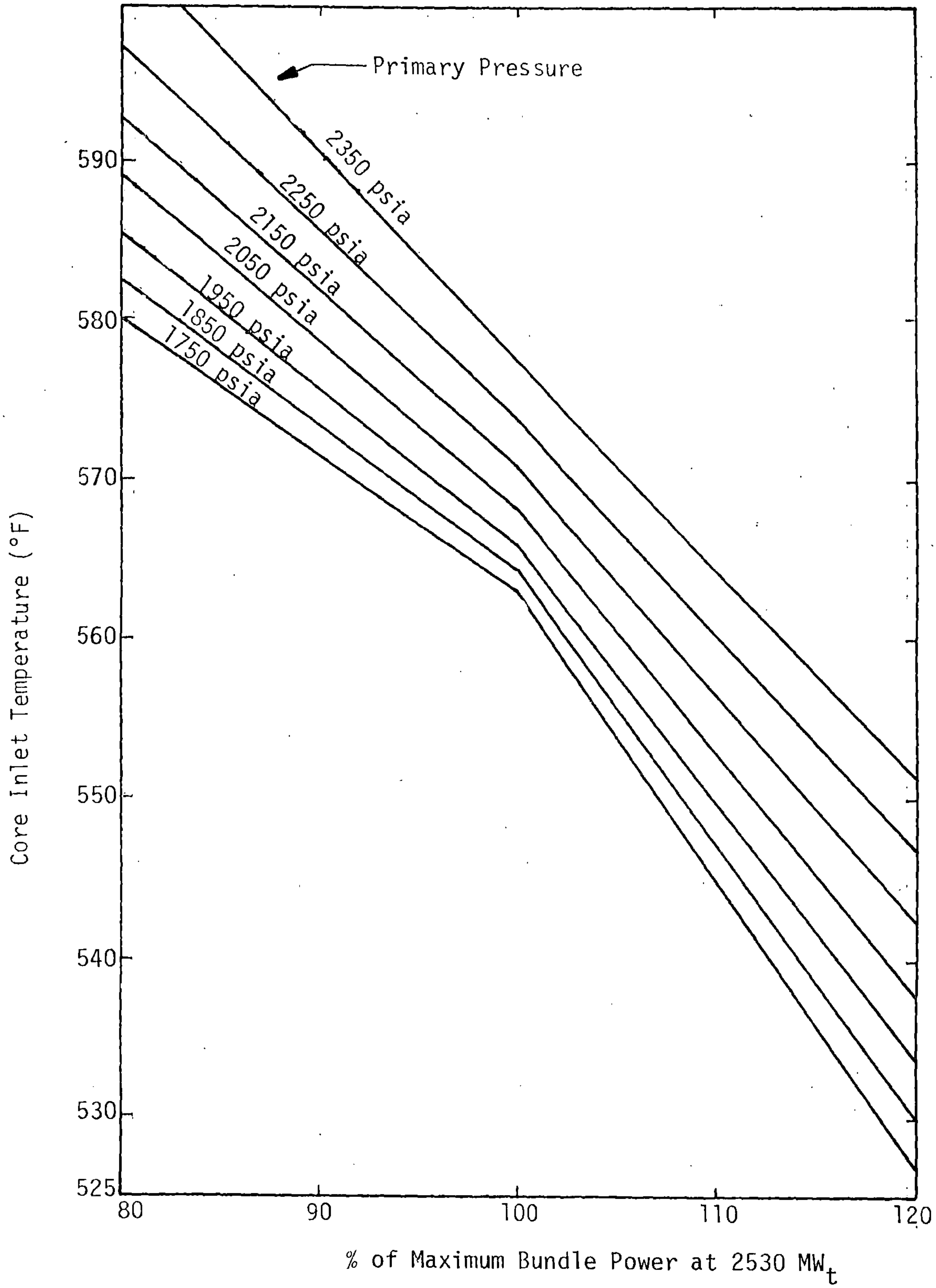


FIGURE 3.3 PALISADES THERMAL MARGIN LIMITING OPERATING CONDITIONS AT DESIGN CORE PEAKING FACTORS (Table 2.1)

4.0 THERMAL DESIGN

4.1 FUEL TEMPERATURE

The GAPEX[5] computer code was used to calculate the pellet-to-clad gap conductance and to calculate the maximum steady-state fuel temperature at design overpower conditions. The fuel temperature calculation considered the effects of the worst fuel pellet and cladding tolerances, fuel densification, pellet cracking, and temperature and density on thermal conductivity. Lyons[6] UO_2 thermal conductivity data was used to define the temperature dependence of the fuel thermal conductivity. Lyon data which is for 95 percent theoretical density fuel was corrected to the reload fuel design density.

A total power peaking factor of 2.55 was used for maximum heat generating rate of 115 percent of 2530 MW_t. Individual peaking factors are listed in Table 4.1.

Steady-state fuel and cladding temperature for the beginning-of-life, low-enrichment fuel were most limiting. All temperatures as shown in Table 4.1 are below design limits.

The GAPEX computer code was used to determine the maximum steady-state poison pellet and poison rod cladding temperatures at overpower. Modifications were necessary to account for:

- The conductivity of B_4C alumina at manufactured density as a function of temperature.

- The decrease in conductivity of B_4C alumina with irradiation.
- The thermal expansion of B_4C alumina.

The linear heat generation rate of the poison rod included factors for core radial peaking, overpower (115 percent), axial peaking, and variations in boron concentration. No densification or cracking effects were considered. Beginning-of-life geometry (largest pellet-to-clad gap) and end-of-life conductivity (maximum conductance) were used to envelope all conditions. Results are listed in Table 4.1.

TABLE 4.1

FUEL, CLADDING AND POISON TEMPERATURES FOR

ENC RELOAD E FUEL RODS WITH A LOCAL PEAKING TO 1.22

Average heat generation rate at 115 percent power	6.24 kw/ft
Peaking Factors:	
Radial	1.45
Axial	1.4
Local	1.22
Engineering	1.03
Total	2.55
Percent Power Deposited in Fuel	0.975
Temperatures:	
Maximum Fuel Centerline	4330°F
Maximum Clad O.D.	662°F
Maximum Clad I.D.	799°F
Maximum Poison Pellet	1100°F

5.0 NEUTRONICS CONSIDERATIONS

The effects of operation at 2530 MWt and of a potential increase in the length of Cycle 2 upon the neutronics characteristics of the Cycle 2 core have been evaluated. Although the Cycle 2 core is somewhat atypical of cores planned for use in the future (i.e., radial power peaking factors for Cycle 2 are somewhat higher than those projected for future cycles), the characteristics of the Cycle 2 core are generally expected to envelope those of future cores. The characteristics of future cores will, of course, be individually analyzed in detail and compared to safety analysis limits at the time of final design of each of the cores.

The analyses reported here relied primarily upon a quarter-core, pin-by-pin PDQ7/HARMONY model for calculation of detailed radial power distributions and differential effects (rod worths, moderator temperature coefficients, etc.) and upon a three-dimensional, quarter-core XTG model for calculation of exposure-dependent core reactivity, axial effects, etc. (See Reference 7 for a basic description of the methodology utilized).

It was assumed that at a Cycle 2 exposure of $\sim$10,500 MWD/MTU the core power will be increased to 2530 MW_t and that core power will be maintained at that level until EOC2. Due to favorable operating experience with the Cycle 2 core and due to a slightly lower rate of reactivity depletion versus exposure than originally expected, Cycle 2 is expected to operate beyond the 12 GWD/MTU EOC2 exposure heretofore addressed. (The core is calculated to be capable of operation at 2530 MW_t to a core exposure of 13,140 MWD/MTU.) To envelope this potential for increased cycle length and to evelope a potential for power coastdown operation beyond this exposure, the neutronics characteristics of the Cycle 2 core were evaluated at 2,530 MW_t and an EOC2 exposure of 15 GWD/MTU.

The neutronics characteristics of the Cycle 2 core at 2,530 MWt and 15 GWD/MTU are compared to those for 2,200 MW_t at 12 GWD/MTU and to the revised safety analysis limits in Table 5.1. The safety-significant neutronics parameters lie well within the safety analysis limits with the exception of radial power peaking which at 15 GWD/MTU is projected to exceed the 2,530 MW_t limit by ∿2%. (The increase from the 1.35 maximum radial peaking factor calculated at 2,200 MW_t and 12 GWD/MTU is primarily the result of increased fuel exposure and increased depletion of the Batch E burnable poison rods). This radial peaking will not, however, be deleterious to core safety since power peaking factors are continuously monitored by in-core detectors and the core power level is adjusted as necessary to maintain margin to thermal limits. (The excess of ∿2% in the calculated radial peaking at 15 GWD/MTU is not expected to actually constrain core operation since core reactivity is expected to be insufficient to maintain 2530 MW beyond 13,140 MWD/MTU).

Control rod worths, reactivity allowances, and calculated shutdown margins at 2,530 MW_t-15 GWD/MTU are compared to values at 2,200 MW_t-12 GWD/MTU in Table 5.2. The gross rod worth is calculated to be slightly higher at 15 GWD/MTU,consistent with the increase calculated between BOC 2 and 12 GWD/MTU, but the maximum stuck rod worth is higher by essentially the same amount and the net rod worth at 15 GWD/MTU is essentially unchanged from that at 12 GWD/MTU. The basic reactivity defect (sum of moderator temperature and Doppler defects) is calculated to be 1.84% $\Delta\rho$ at 15 GWD/MTU (versus 1.29% $\Delta\rho$ at 12 GWD/MTU) and a conservative reactivity allowance of 2.0% $\Delta\rho$ is provided for these effects.

An excess shutdown margin of 0.33% $\Delta\rho$ is conservatively projected from HFP at 15 GWD/MTU. Considering the revised HZP shutdown requirement of 2.0% $\Delta\rho$ and assuming no change in the HZP PDIL rod insertion from that quoted in Reference 8, an excess shutdown margin of 1.41% $\Delta\rho$ is conservatively projected for HZP at 15 GWD/MTU.

It is concluded that the neutronics characteristics of the Palisades Cycle 2 core in combination with routine procedures for maintenance of thermal margins enable safe operation of the core at power levels up to 2,530 MW_t and exposures up to 15 GWD/MTU.

TABLE 5.1

EFFECTS OF INCREASED POWER AND CYCLE LENGTH ON PALISADES

CYCLE 2 CORE NEUTRONICS PARAMETERS

	2200 MWt, 12 GWD/MTU	2530 MWt, 15 GWD/MTU	Safety Analysis Limit
Moderator Temperature Coefficient, $10^{-4}\ \Delta\rho/^{o}F$	-1.42	-2.14	+0.5 to -3.5
Doppler Coefficient, $10^{-5}\Delta\rho/^{o}F$	-1.49	-1.55	-.87 to -1.64
Power Peaking Factors:			
Radial	1.35	1.48*	1.45
Axial	1.10	1.12	1.40
Total (Including Local)	1.74	1.92	$\leq$ 2.55
Max. Ejected Rod Worth, %$\Delta\rho$			
HFP	<0.2	<0.2	<0.6
HZP	<0.90	<0.84	<1.24
Delayed Neutron Fraction	0.0052	0.0049	$\geq$0.0045
Reciprocal Boron Worth, ppm/%$\Delta\rho$	84	85	$\leq$100

*Operation to 15 GWD/MTU would probably require a slight derate due to radial peaking being greater than 1.45. Core would be in power coastdown due to reactivity depletion at this exposure.

TABLE 5.2

PALISADES EOC2 SHUTDOWN MARGIN

(All Entries in Units of %Δρ)

	2200 MWt, 12 GWD/MTU		2530 MWt, 15 GWD/MTU	
	HZP	HFP	HZP	HFP
Total Full Length Rod Worth*	9.15	9.15	9.33	9.33
Stuck Rod Worth	3.52	3.52	3.69	3.69
Total Minus Stuck Rod	5.63	5.63	5.64	5.64
Uncertainty	.56	.56	.56	.56
Net Shutdown Rod Worth	5.07	5.07	5.08	5.08
Doppler Defect	0	1.00	0	1.10
Moderator Temperature Defect	0	.80	0	.90
Moderator Void Defect	0	.10	0	.10
Axial Flux Redistribution	0	.50	0	.50
Required Shutdown Margin	3.40	2.00	2.00	2.00
Total Reactivity Allowances	3.40	4.40	2.00	4.60
Available for Maneuvering	1.67	0.67	3.08	0.48
PDIL Rod Insertion	1.67	0.15	1.67	0.15
Excess Margin	0.00	0.52	1.41	0.33

*EOC Rod Worth = EOC (Calculated) x BOC Measured/BOC Calculated

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