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satisfy the criterion. Appropriate consideration shall be given to the following factors:

(1) A flux peaking augmentation factor of 1.0,

(2) A measurement calculational uncertainty factor of 1.10,

(3) An engineering uncertainty factor (which includes fuel column shortening due to densification and thermal expansion) of 1.03, and

(4) A thermal power measurement uncertainty factor of 1.02.

b. If the quadrant to core average power tilt exceeds 15%, except for physics tests, then:

(1) The linear heat generation rate shall promptly be demonstrated to be less than that specified in Part a, or

(2) Immediate action shall be initiated to reduce reactor power to 75% or less of rated power.

c. If the power in a quadrant exceeds core average by 10% for a period of 24 hours or if the power in a quadrant exceeds core average by 20% at any time, immediate action shall be initiated to reduce reactor power below 50% until the situation is remedied.

d. If the power in a quadrant exceeds the core average by 15% and if the linear heat generation rate cannot be demonstrated promptly to be within limits, then the overpower trip set point shall be reduced to 80% and the thermal margin low-pressure trip set point (P_{Trip}) shall be increased by 400 psi.

e. If the power in a quadrant exceeds core average by 5% for a period of 30 days, immediate action shall be initiated to reduce reactor power to 75% or less of rated power.

f. The part-length control rods will be completely withdrawn from the core (except for rod exercises and physics tests).

g. The calculated value of F_r^A shall be limited to $\leq$ 1.45 (1.0 + 0.5 (1 - P)), the calculated value of F_r^{T*} shall be limited to $\leq$ 1.77 (1.0 + 0.5 (1 - P)), and the calculated value of $F_r^{\Delta H}$ shall be limited to $\leq$ 1.66 (1.0 + 0.5 (1 - P)), where P is the core thermal power in fraction of core rated thermal power (2530 MWt).

(*For the duration of Cycle-4 for H-fuel only, F_r^T for rods adjacent to the wide water gap shall be limited to 1.90 (1.0 + 0.5 (1 - P)).)

Specification (Contd)

a 10-hour period) at least each two hours thereafter or the reactor power level shall be reduced to less than 50% of rated power (65% of rated power if no dropped or misaligned rods are present). If readings indicate a local power level equal to or greater than the alarm set point, the action specified in 3.11.b shall be taken.

g. F_r^A, F_r^T and $F_r^{\Delta H}$ shall be determined whenever the core power distribution is evaluated. If either F_r^A, F_r^T or $F_r^{\Delta H}$ is found to be in excess of the limit specified in Section 3.10.3(g), within six hours thermal power shall be reduced to less than that required to assure compliance with Section 3.10.3(g).

Basis

A system of 45 in-core flux detector and thermocouple assemblies and a data display, alarm and record functions has been provided. A four level, five level or six level system may be used.(1)(2) The out-of-core nuclear instrumentation calibration includes:

a. Calibration (axial and azimuthal) of the split detectors at initial reactor start-up and during the power escalation program.

b. A comparison check with the in-core instrumentation in the event abnormal readings are observed on the out-of-core detectors during operation.

c. Calibration check during subsequent reactor start-ups.

d. Confirm that readings from the out-of-core split detectors are as expected.

Core power distribution verification includes:

a. Measurement at initial reactor start-up to check that power distribution is consistent with calculations.

b. Subsequent checks during operation to insure that power distribution is consistent with calculations.

c. Indication of power distribution in the event that abnormal situations occur during reactor operation.

If the data logger for the in-core readout is not in operation for more than two hours, power will be reduced to provide margin between the actual peak linear heat generation rates and the limit and the in-core readings will be manually collected at the terminal blocks in the control room utilizing a suitable signal detector. If this is not feasible with the

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ECCS AND THERMAL-HYDRAULIC ANALYSIS FOR THE PALISADES RELOAD H DESIGN

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1.0 INTRODUCTION AND SUMMARY

This report documents LOCA/ECCS and thermal-hydraulic analyses for the Exxon Nuclear Company (ENC) Batch H fuel design for the Palisades nuclear power plant. These analyses support a maximum allowable linear heat generation rate limit of 15.28 Kw/ft for the Palisades H design, which is the same as previously established for the ENC E/G design[1]. The allowable assembly radial peaking factor of 1.45 at full core power (2530 MWt) is also unchanged from the analyses for the E/G design. Increased local peaking for the wide gap edge rods in the Palisades H design has been considered. Thus for the present H-fuel analysis, a wide gap corner rod local peaking factor of 1.31 is incorporated in the analyses versus a wide gap corner rod local peaking of 1.22 in the E/G analyses. Limits on interior rod local peaking remain unchanged.

The mechanical design differences between the Palisades H design and the prior E/G design are shown in Table 1.1. In addition to the mechanical design differences of Table 1.1, the neutronics design of the Palisades H reload fuel is slightly different from the E/G reload fuel (e.g. fuel rod enrichments, poison rods.) The present LOCA/ECCS and thermal-hydraulic analyses account for the mechanical design differences, as well as the increased local peaking for the wide gap rods in the H-fuel design.

Table 1.1 Design Differences between Reloads E/G and H

Design Component	Reload E/G	Reload H
Fuel Pellet		
- Pellet Diameter (in.)	.3505	.35
- Dish Volume (%)	2.0	1.0
Cladding		
- Outside Diameter (in.)	.415	.417
Fuel Rod		
- He Fill Pressure (psia)		

2.0 ECCS ANALYSIS

2.1 LOCA PERFORMANCE SUMMARY

The performance of the Palisades H fuel design in a postulated loss-of-coolant accident (LOCA) has been evaluated. The nominal total peaking F_Q limit of 2.76 that previous ECCS analyses had determined for ENC Batch E/G fuel has been applied to the H-fuel design, and has been found to be acceptable relative to 10 CFR 50.46 criteria. This F_Q limit corresponds to a maximum allowed linear heat generation rate (LHGR) of 15.28 Kw/ft for the Palisades H design. The calculations were performed for the Palisades 0.6 DEG/PD limiting break[2]. The calculations account for the mechanical design differences from E/G fuel noted in Table 1.1, and incorporate an ECCS limiting rod local peaking factor of 1.31. An axial power peak location sensitivity calculation was also performed. This calculation confirmed the continued applicability of the ECCS allowable LHGR as a function of axial power peak location established previously for E/G fuel[1]. The present analysis is sufficient for pellet exposures of approximately 30,000 MWD/MTM for the Palisades H design fuel.

2.2 MODELS AND ASSUMPTIONS

RELAP4-EM/HOT CHANNEL and TOODEE2 heatup analyses were made to determine the performance of Batch H fuel during a LOCA. The HOT CHANNEL and TOODEE2 calculations require the following boundary conditions:

- The upper and lower plenum fluid conditions (pressure and enthalpy) versus time during blowdown.
- Normalized power versus time.
- The EOBY and BOCREC event times.
- Reflood rate and saturation temperature versus time during the refill and reflood time periods.
- ECCS subcooling during reflood.

These boundary conditions are provided by the Palisades, 2530 MWt core power, 0.6 DEG/PD limiting break calculation for ENC Batch E/G fuel[3]. As in the previous break spectrum analysis for Palisades[3], the axial power peak location relative to the bottom of the active core for the reference case in the present analysis is at X/L=0.6. The present axial power peak location sensitivity study also considered the most limiting case of the previous E/G-fuel axial sensitivity study[1] where the maximum LHGR has been reduced by 16% and peaked at X/L=0.8, and the axial power distribution has a 1.1 skewing factor. The key parameters used in the analysis are summarized in Table 2.1. With the exception that the limiting rod local peaking has been increased from 1.22 to 1.31, the peaking factors identified in Table 2.1 are the same as identified in Reference 1. The analysis is in accordance with

ENC's WREM-II PWR ECCS Evaluation Model[2,4,5]. The HOTCHANNEL calculations were performed with ENC's Version 26A of the RELAP4 code as have prior analyses for Palisades. The current MAY79 TOODEE2 code was used to calculate the limiting rod heatup transient.

Fuel parameters in the present analysis correspond to beginning-of-life conditions. The recent Nuclear Regulatory Commission (NRC) Clad Swelling and Rupture Model[6] has been applied in the present TOODEE2 heatup calculations for the Palisades H design. The present analysis includes consideration of uncertainties in the hoop stress at the time of clad rupture in the application of the NRC model. This is done by consideration of rod internal pressure uncertainties in accordance with the models detailed in References 7, 8, 9, and as approved in Reference 10.

2.3 ANALYSIS RESULTS

The results of the final TOODEE2 heatup calculations are given in Table 2.2. The upper bound on fuel rod internal pressure uncertainties has been incorporated into these heatup results. Table 2.2 shows that a margin exists between the calculated PCT and the limiting PCT of 2200°F for both axial peaking locations at X/L=0.6 and X/L=0.8. The corresponding limiting rod clad temperature heatup transients as calculated using TOODEE2 are given in Figures 2.1 and 2.2 respectively.

Table 2.3 shows a comparison of calculated peak clad temperature (PCT) results for E/G fuel in Reference 1 with the present results for the H-fuel design.

In comparing these results, it is noted that the PCT's calculated for E/G fuel were determined using the ENC clad swelling and rupture model, while those for H-fuel used the NRC model[6]. In addition, the H-fuel design has a slightly larger fuel rod diameter than that of the E/G designs. In spite of these design and analysis differences, the calculated PCT's for both designs are approximately equivalent.

Table 2.1 Palisades H-fuel ECCS Analysis Parameters

Reactor Power at 102%, MWt	2580.6
Reactor Pressure, psia	2060.
Heat Release in Fuel	97.5%
Limiting Break	0.6 DEG/PD
Hot Assembly Radial Peaking	1.45
Hot Rod Local Peaking, F_ℓ	1.31
Engineering Factor	1.03
Reference Case	
X/L	0.6
Skewing Factor	1.0
F_Q	2.76
Top Skewed Axial Power Profile Case	
X/L	0.8
Skewing Factor	1.1
F_Q	2.32

Table 2.2 Heatup Analyses Results for the H-Fuel Design

Axial Power Peak Location, X/L	0.6	0.8
Skewing Factor	1.0	1.1
Total Peaking, F_Q	2.76	2.32
Peak Clad Temperature (PCT), ^oF	2067.	2183.
Max. Local Zr/H_2O Reaction, %	6.8	11.1
Hot Rod Burst Time, sec.		
Hot Rod Burst Location, ft.		
Rupture Pressure, psid		
Flow Blockage, %		
Time of PCT, sec.		
PCT Location, ft.		
Max. Zr/H_2O Reaction Location, ft.		
Heatup Rate at Rupture - ^oC/sec.		

Table 2.3 Comparison of Peak Clad Temperature Results for E/G and H-Fuel Designs

Axial Power Peak Location, X/L	Skewing Factor	G-Fuel F_ℓ=1.22	H-fuel F_ℓ=1.31
0.6	1.0	2081	2067
0.8	1.1	2172	2183

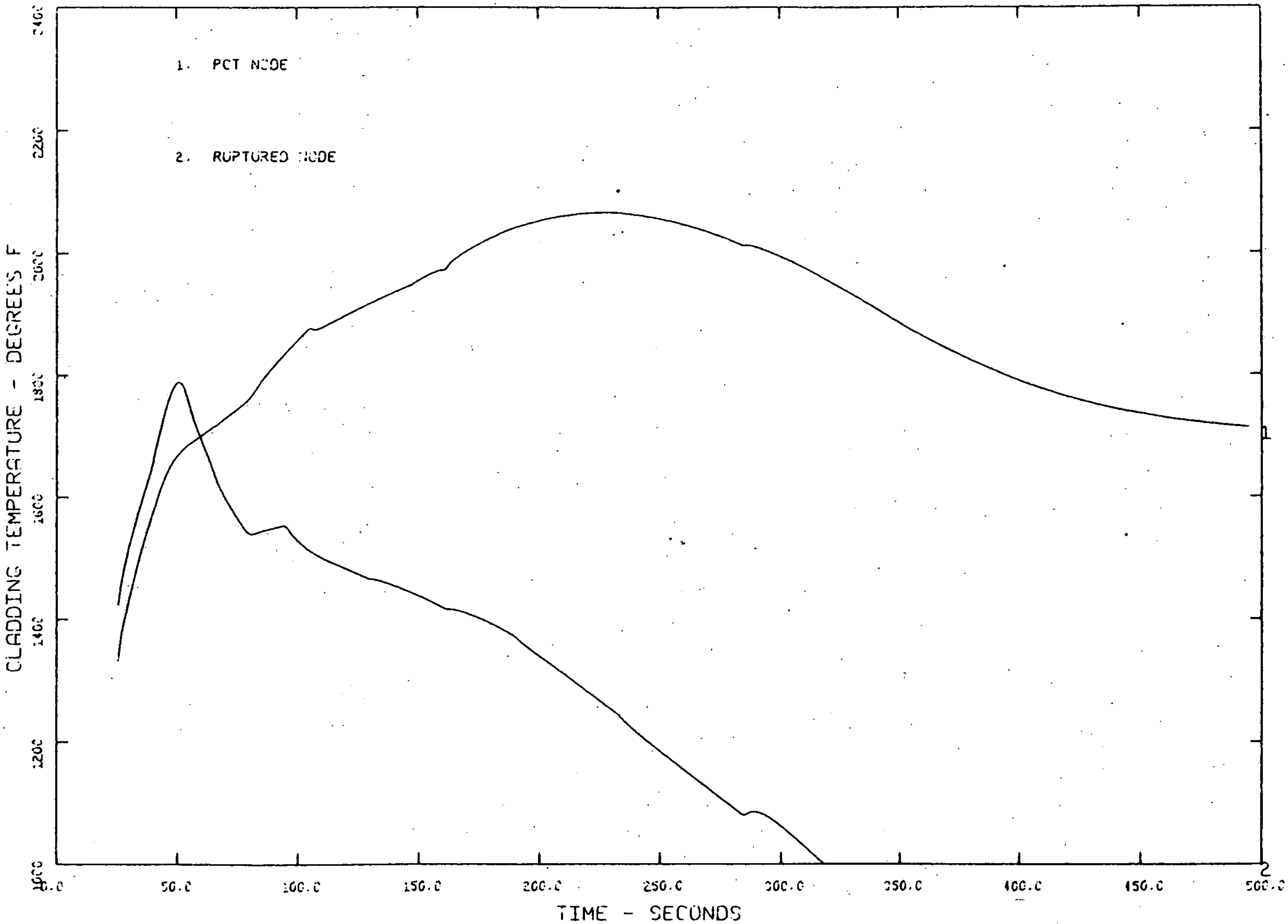

Figure 2.1 Cladding Surface Temperature During Heatup for H-Fuel
(Axial Peak Power Location X/L=0.6, Skewing Factor=1.0)

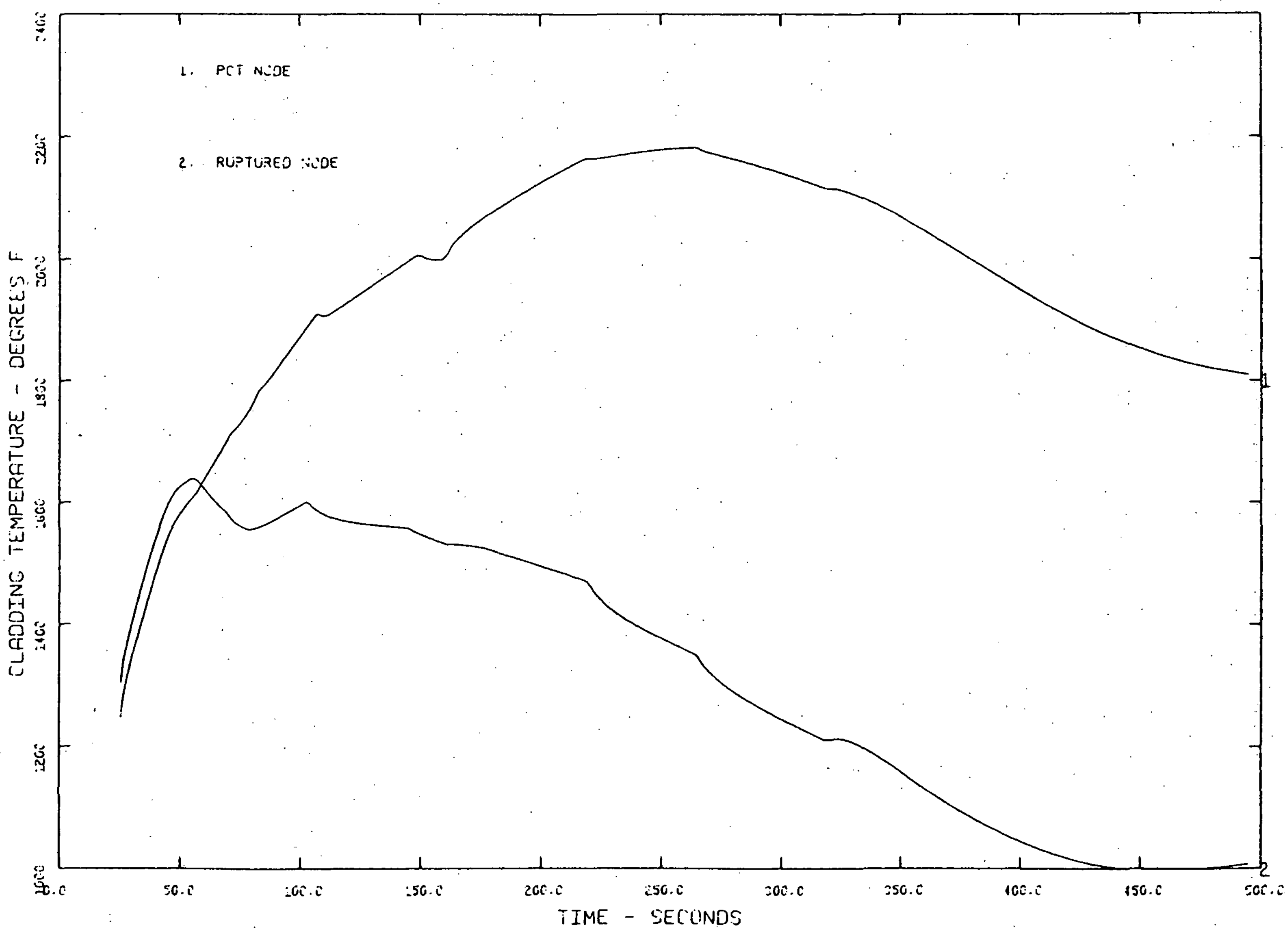

Figure 2.2 Cladding Surface Temperature During Heatup for H-Fuel
(Axial Peak Power Location X/L=0.8, Skewing Factor=1.1)

3.0 THERMAL-HYDRAULIC ANALYSIS

The ENC Reload H fuel is designed to be compatible with the Palisades Reactor core and with the existing fuel. The thermal-hydraulic design criteria for ENC reload fuel at Palisades are:

- The maximum fuel temperature at 115% overpower shall not exceed the fuel melting temperature.
- The minimum DNBR shall be greater than or equal to 1.30 at 115% of rated power based on the W-3 correlation (or an accepted equivalent) plus correction factors which have been accepted by the NRC for the purpose of licensing the fuel design described herein.
- The cladding temperature at nominal operating conditions (based on crud-free surface) shall be less than:

  850°F internal surface

  675°F external surface

  750°F volume average (local)
- The fuel assemblies must be thermally and hydraulically compatible with the existing fuel and the reactor core during the design life of the fuel.

The thermal-hydraulic analysis for the Palisades H fuel was performed in a manner consistent with conditions reported in licensing data provided for operation of the Palisades plant at 2530 MWt[1,11]. The thermal-hydraulic design conditions for this analysis are shown in Table 3.1, which reflects the increased local peaking for the wide gap edge rods. The resulting performance of Palisades H fuel falls within the thermal-hydraulic design criteria. Therefore, the insertion of the Batch H reload fuel into the Palisades reactor is acceptable.

TABLE 3.1

THERMAL-HYDRAULIC DESIGN CONDITIONS AND PEFORMANCE

Reactor Conditions	Design	Nominal
Core power (MWt)	2910	2530
Total reactor flow rate (Mlb/hr)	121.7	121.7
Active core flow rate (Mlb/hr)	114.4	114.4
Coolant inlet temperature (°F)	542.5	537.5
Core pressure (psia)	2010	2060
Power Distribution		
Assembly radial peaking, F_R		1.45
Pin power peaking (for interior rods), $F_R x F_\ell$		1.66
Pin power peaking (for narrow gap edge rods), $F_R x F_\ell$		1.77
Pin power peaking (for wide gap edge rods), $F_R x F_\ell$		1.90
Axial peaking, F_a (at 0.6 of active fuel height)		1.41
Engineering factor, F_e		1.03
Total peaking factor		2.76
Thermal-Hydraulic Performance		
Hot assembly flow factor		0.97
MDNBR		$\geq$1.3
Fuel center temperature (°F)*		4767
Clad outer surface temperature (°F)**		668
Clad inner surface temperature (°F)**		803
Volumetric averaged clad temperature (°F)**		736

* At 115% of rated power.
** At nominal power.

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ECCS AND THERMAL-HYDRAULIC ANALYSES FOR

THE PALISADES RELOAD H DESIGN

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A F F I D A V I T

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I, James N. Morgan, being duly sworn, hereby say and depose:

1. I am Manager, Licensing and Safety Engineering, for Exxon Nuclear Company, Inc., ("ENC") and as such I am authorized to execute this Affidavit.

2. I am familiar with ENC's detailed document control system and policies which govern the protection and control of information.

3. I am familiar with the document XN-NF-80-18(P), entitled "ECCS and Thermal-Hydraulic Analyses for the Palisades Reload H Design," referred to as "Document". Information contained in this Document has been classified by ENC as proprietary in accordance with the control system and policies established by ENC for the control and protection of information.

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13. Based on my experience in the industry, I do not believe that the background data and incentives of ENC's competitors are sufficiently similar to the corresponding background data and incentives of ENC to reasonably expect such competitors would be in a position to duplicate ENC's proprietary information contained in the Document.

THAT the statements made hereinabove are, to the best of my knowledge, information, and belief, truthful and complete.

FURTHER AFFIANT SAYETH NOT.

J. N. Morgan

SWORN TO AND SUBSCRIBED
before me this 24 day of
April, 1980.

Susan L. Backus
NOTARY PUBLIC