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

December 8, 1981

DISTRIBUTION
Docket
NRC PDR
Local PDR
ORB Reading
NSIC
DCrutchfield
HSmith(2)
TWambach
JLyons
OELD
OI&E (5)
GDeegan (4)
ACRS (10)
BSCharf (10)
RDiggs
SEPB

Docket No. 50-255
LS05-81-12-027

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US NUCLEAR REGULATORY COMMISSION
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Mr. David P. Hoffman
Nuclear Licensing Administrator
Consumers Power Company
1945 W. Parnall Road
Jackson, Michigan 49201

Dear Mr. Hoffman:

SUBJECT: CYCLE 5 RELOAD - PALISADES PLANT

The Commission has issued the enclosed Amendment No. 68 to Provisional Operating License No. DPR-20 for the Palisades Plant. This amendment consists of changes to the Technical Specifications in response to your application dated July 21, 1981, as supplemented August 6, 1981, October 22, 1981 (two letters), November 9, 17, 20, 1981, and December 2, 1981. You also provided information changing the title of figure 3.23-3, submitted October 22, 1981, during a November 18, 1981, telephone discussion between your Mr. B. Johnson and our Mr. T. Wambach.

The amendment approves changes to the provisions of the Appendix A Technical Specifications which specify new limits for radial peaking factors and allowable linear heat rates as well as identifying the use of excore detectors for core power distribution monitoring.

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

Sincerely,

Original signed by
Thomas V. Wambach for/

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Enclosures:
1. Amendment No. 68 to
   License No. DPR-20
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
See next page

F.R. NOTICE
+ AMENDMENT

*SEE PREVIOUS TISSUE FOR CONCURRENCE

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OFFICE	DL: ORB #5*	DL: ORB #5*	OELD	DL: ORB #5	DL: ORB #5	DL: AD/SA	
SURNAME	HSmith:cc	JLyons		TWambach	DCrutchfield	GLainas	
DATE				12/3/81	12/4/81	12/7/81	

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Nuclear Licensing Administrator
Consumers Power Company
1945 W. Parnall Road
Jackson, Michigan 49201

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SUBJECT: CYCLE 5 RELOAD - PALISADES PLANT

The Commission has issued the enclosed Amendment No. to Provisional Operating License No. DPR-20 for the Palisades Plant. This amendment consists of changes to the Technical Specifications in response to your application dated July 21, 1981, as supplemented August 6, 1981, October 22, 1981, (two letters), November 9, 17, 20, 1981, and December 2, 1981. You also provided information changing the title of figure 3-23-3, submitted October 22, 1981, during a November 18, 1981, telephone discussion between your Mr. B. Johnson and our Mr. T. Wambach.

The amendment approves changes to the provisions of the Appendix A Technical Specifications which specify new limits for radial peaking factors and allowable linear heat rates as well as identifying the use of excore detectors for core power distribution monitoring.

Certain modifications to your proposed changes were necessary to meet our criteria. These modifications have been discussed with and agreed to by our staff.

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

Sincerely,

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Enclosures:
1. Amendment No. to
   License No. DPR-20
2. Safety Evaluation
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ORB #5
J Lyons
12/3/81

cc w/enclosures:
See next page

OFFICE			DL: ORB #5	DL:ORB #5	OELD	DL: ORB #5	DL: AD/SA
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

CONSUMERS POWER COMPANY

DOCKET NO. 50-255

PALISADES PLANT

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 68
License No. DPR-20

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

   A. The application for amendment by Consumers Power Company (the licensee) dated July 21, 1981 as supplemented August 6, 1981, October 22, 1981, November 9, 17, 20, 1981 and December 2, 1981 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 3.B of Provisional Operating License No. DPR-20 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B (Environmental Protection Plan), as revised through Amendment No. 68, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas V. Wambach

for Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 8, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 68

PROVISIONAL OPERATING LICENSE NO. DPR-20

DOCKET NO. 50-255

Revise Appendix A Technical Specifications by removing the following pages and by inserting the enclosed pages. The revised pages contain the captioned amendment number and marginal lines indicating the area of change.

Remove	Insert
ii	ii
iii	iii
--	iv*
1-2	1-2
3-38	3-38**
3-58	3-58
3-59	--
3-61	3-61
3-63	3-63
3-64	3-64 (Intentionally Blank)
3-65 - 3-66a	3-65 - 3-66d
3-81a	3-81a
3-87	--
3-87a	--
--	3-103 - 3-113
--	4-81 - 4-84
6-1a	6-1a**

* Included for pagination purposes only.

** These pages are included for the purpose of correcting errors which occurred during the issuance of Amendment No. 62 (page 3-38) and Amendment No. 67 (6-1a).

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Low Power Physics Testing

Testing performed under approved written procedures to determine control rod worths and other core nuclear properties. Reactor power during these tests shall not exceed 2% of rated power, not including decay heat and primary system temperature and pressure shall be in the range of 260°F to 538°F and 415 psia to 2150 psia, respectively. Certain deviations from normal operating practice which are necessary to enable performing some of these tests are permitted in accordance with the specific provisions therefor in these Technical Specifications.

Shutdown Boron Concentrations

Boron concentration sufficient to provide $k_{eff} \leq 0.98$ with all control rods in the core and the highest worth control rod fully withdrawn.

Refueling Boron Concentration

Boron concentration of coolant at least 1720 ppm (corresponding to a shutdown margin of at least 5% $\Delta\rho$ with all control rods withdrawn).

Quadrant Power Tilt

The difference between nuclear power in any core quadrant and the average in all quadrants.

Assembly Radial Peaking Factor - F_r^A

The assembly radial peaking factor is the maximum ratio of individual fuel assembly power to core average assembly power integrated over the total core height, including tilt.

Total Radial Peaking Factor - F_r^T

The total radial peaking factor is the maximum product of the ratio of individual assembly power to core average assembly power times the local peaking factor for that assembly integrated over the total core height, including tilt. Local peaking factor is defined as the maximum ratio of the power in an individual fuel rod to assembly average rod power.

Interior Fuel Rod

Any fuel rod of an assembly that is not on that assembly's periphery.

Total Interior Rod Radial Peaking Factor - $F_r^{\Delta H}$

The maximum product of the ratio of individual assembly power to core average assembly power times the highest interior local peaking factor integrated over the total core height including tilt.

Axial Offset

The difference between the power in the lower half of the core and the upper half of the core divided by the sum of the powers in the lower half and upper half of the core.

Narrow Water Gap Fuel Rod

A fuel rod adjacent to the narrow inter-fuel assembly water gap (a gap not containing a control rod).

Narrow Water Gap Fuel Rod Peaking Factor - F_r^N

The maximum product of the ratio of individual fuel assembly power to core average fuel assembly power times the highest narrow water gap fuel rod local peaking factor integrated over the total core height including tilt.

3.5 STEAM AND FEEDWATER SYSTEMS

Applicability

Applies to the operating status of the steam and feedwater systems.

Objective

To define certain conditions of the steam and feedwater system necessary to assure adequate decay heat removal.

Specifications

3.5.1 The primary coolant shall not be heated above 325°F unless the following conditions are met:

a. Both auxiliary feedwater pumps operable and one fire pump operable.

b. A minimum of 100,000 gallons of water in the condensate storage and primary coolant system makeup tanks combined and a backup source of additional water from Lake Michigan by the operability of one of the fire protection pumps.

c. All valves, interlocks and piping associated with the above components required to function during accident conditions, are operable.

d. The main steam stop valves are operable and capable of closing in five seconds or less under no-flow conditions.

3.5.2 With the primary coolant system at a temperature greater than 325°F, the requirements of 3.5.1 may be modified to permit the following conditions to exist. If the system is not restored to meet the requirements of 3.5.1 within the time period specified below, the reactor shall be placed in the cold shutdown condition within 24 hours.

a. One auxiliary feedwater pump may be inoperable for a period of 72 hours, or

b. The firewater makeup to the auxiliary feedwater pump suction may be inoperable for a period of 72 hours.

3.5.3 If one auxiliary feedwater pump and the firewater makeup supply to the auxiliary feedwater pumps become inoperable, then the plant shall be placed in hot standby within 1 hour, in hot shutdown within the next 6 hours, and in cold shutdown within the following 30 hours.

3.5.4 With both auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible and reduce power within 24 hours to the lowest stable power level consistent with reliable main feedwater system operation.

3.10 CONTROL RODS

Applicability

Applies to operation of control rods and hot channel factors during operation.

Objective

To specify limits of control rod movement to assure an acceptable power distribution during power operation, limit worth of individual rods to values analyzed for accident conditions, maintain adequate shutdown margin after a reactor trip and to specify acceptable power limits for power tilt conditions.

Specifications

3.10.1 Shutdown Margin Requirements

a. With four primary coolant pumps in operation at hot shutdown and above, the shutdown margin shall be 2%.

b. With less than four primary coolant pumps in operation at hot shutdown and above, the shutdown margin shall be 3.75%.

c. At less than the hot shutdown condition, boron concentration shall be shutdown boron concentration.

d. If a control rod cannot be tripped, shutdown margin shall be increased by boration as necessary to compensate for the worth of the withdrawn inoperable rod.

e. The drop time of each control rod shall be no greater than 2.5 seconds from the beginning of rod motion to 90% insertion.

3.10.2 Individual Rod Worth

a. The maximum worth of any one rod in the core at rated power shall be equal to or less than 0.6% in reactivity.

b. The maximum worth of any one rod in the core at zero power shall be equal to or less than 1.2% in reactivity.

3.10.3 Part-Length Control Rods

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

3.10.6 Shutdown Rod Limits

a. All shutdown rods shall be withdrawn before any regulating rods are withdrawn.

b. The shutdown rods shall not be withdrawn until normal water level is established in the pressurizer.

c. The shutdown rods shall not be inserted below their exercise limit until all regulating rods are inserted.

3.10.7 Low Power Physics Testing

Sections 3.10.1.a, 3.10.1.b, 3.10.2.b, 3.10.3 , 3.10.4.b, 3.10.5 and 3.10.6 may be deviated from during low power physics testing and CRDM exercises if necessary to perform a test but only for the time necessary to perform the test.

3.10.8 Center Control Rod Mialignment

The requirements of Specifications 3.10.4.1, 3.10.4.a, and 3.10.5 may be suspended during the performance of physics tests to determine the isothermal temperature coefficient and power coefficient provided that only the center control rod is misaligned and the limits of Specification 3.23 are maintained.

Basis

Sufficient control rods shall be withdrawn at all times to assure that the reactivity decrease from a reactor trip provides adequate shutdown margin. The available worth of withdrawn rods must include the reactivity defect of power and the failure of the withdrawn rod of highest worth to insert. The requirement for a shutdown margin of 2.0% in reactivity with 4-pump operation, and of 3.75% in reactivity with less than 4-pump operation, is consistent with the assumptions used in the analysis of accident conditions (including steam line break) as reported in XN-NF-77-18 and additional analysis.[5] The change in insertion limit with reactor power shown on Figure 3-6 insures that the shutdown margin requirements for 4-pump operation is met at all power levels. The 2.5-second drop time specified for the control rods is the drop time used in the transient analysis.[5]

The maximum individual rod worth of inserted control rods and associated peaking factors have been used to demonstrate reactor safety for the unlikely event of a rod ejection accident as described in Reference 5. The maximum worth of an inserted control rod will not exceed the values of the specification for the regulating group insertion limits of Figure 3-6.

The insertion of part-length rods into the core, except for rod exercises or physics tests, is not permitted since it has been demonstrated on other CE plants that design power distribution envelopes can, under some circumstances, be violated by using part-length rods. Further information may justify their use. Part-length rod insertion is permitted for physics tests, since resulting power distributions are closely monitored under test conditions. Part-length rod insertion for rod exercises (approximately 6 inches) is permitted since this amount of insertion has an insignificant effect on power distribution.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS (Contd)

For a control rod misaligned up to 8 inches from the remainder of the banks, hot channel factors will be well within design limits. If a control rod is misaligned by more than 8 inches, the maximum reactor power will be reduced so that hot channel factors, shutdown margin and ejected rod worth limits are met. If in-core detectors are not available to measure power distribution and rod misalignments > 8 inches exist, then reactor power must not exceed 75% of rated power to insure that hot channel conditions are met.

Continued operation with that rod fully inserted will only be permitted if the hot channel factors, shutdown margin and ejected rod worth limits are satisfied.

In the event a withdrawn control rod cannot be tripped, shutdown margin requirements will be maintained by increasing the boron concentration by an amount equivalent in reactivity to that control rod. The deviations permitted by Specification 3.10.7 are required in order that the control rod worth values used in the reactor physics calculations, the plant safety analysis, and the Technical Specifications can be verified. These deviations will only be in effect for the time period required for the test being performed. The testing interval during which these deviations will be in effect will be kept to a minimum and special operating precautions will be in effect during these deviations in accordance with approved written testing procedures.

Violation of the power dependent insertion limits, when it is necessary to rapidly reduce power to avoid or minimize a situation harmful to plant personnel or equipment, is acceptable due to the brief period of time that such a violation would be expected to exist, and due to the fact that it is unlikely that core operating limits such as thermal margin and shutdown margin would be violated as a result of the rapid rod insertion. Core thermal margin will actually increase as a result of the rapid rod insertion. In addition, the required shutdown margin will most likely not be violated as a result of the rapid rod insertion because present power dependent insertion limits result in shutdown margin in excess of that required by the safety analysis.[5]

References

(1) FSAR, Section 14.
(2) FSAR, Section 3.3.3.
(3) FSAR, Section 7.4.2.2.
(4) FSAR, SEction 7.3.3.6.
(5) XN-NF-77-18.

(INTENTIONALLY BLANK)

3.11 POWER DISTRIBUTION INSTRUMENTATION

3.11.1 INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

The incore detection system shall be operable:

a. With at least 50% of the incore detectors and 2 incores per axial level per core quadrant.

b. With the incore alarming function of the datalogger operable and alarm setpoints entered into the datalogger.

APPLICABILITY

(1) Item a. above is applicable when the incore detection system is used for:

Measuring quadrant power tilt,

Measuring radial peaking factors,

Measuring linear heat rate (LHR), or

Determining target Axial Offset (AO) and excore monitoring allowable power level.

(2) Items a. and b. above are applicable when the incore detection system is used for monitoring LHR with automatic alarms. (Incore Alarm System.)

ACTION 1:

With less than the required number of incore detectors, do not use the system for the measuring and calibration functions under (1) above.

ACTION 2: With the alarming function of the datalogger inoperable, do not use the system for automatic monitoring of LHR (Inoperable Incore Alarm System).

POWER DISTRIBUTION INSTRUMENTATION

3.11.1 INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

ACTION 2: (Contd)

Operation may continue using the excore monitoring system as specified in 3.11.2 or by meeting the requirements of 3.23.1.

Basis

The operability of the incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core. The operability of the incore alarm system depends on the availability of the datalogger as well as the operability of a minimum number of incore detectors. Incore alarm setpoints must be updated periodically based on measured power distributions. The incore detector Channel Check is normally performed by an off-line computer program that correlates readings with one another and with computed power shapes in order to identify inoperable detectors.

POWER DISTRIBUTION INSTRUMENTATION

3.11.2 EXCORE POWER DISTRIBUTION MONITORING SYSTEM

LIMITING CONDITION FOR OPERATION

The excore monitoring system shall be operable with:

a. The target Axial Offset (AO) and the Excore Monitoring Allowable Power Level (APL) determined within the previous 31 days using the incore detectors, and the measured AO not deviated from the target AO by more than 0.05 in the previous 24 hours.

b. The AO measured by the excore detectors calibrated with the AO measured by the incore detectors.

c. The quadrant tilt measured by the excore detectors calibrated with the quadrant tilt measured by the incore detectors.

APPLICABILITY:

(1) Items a., b. and c. above are applicable when the excore detectors are used for monitoring LHR.

(2) Item c. above is applicable when the excore detectors are used for monitoring quadrant tilt.

ACTION 1:

With the excore monitoring system inoperable, do not use the system for monitoring LHR.

ACTION 2:

If the measured quadrant tilt has not been calibrated with the incores, do not use the system for monitoring quadrant tilt.

Basis

The excore power distribution monitoring system consists of Power Range Detector Channels 5 through 8.

The operability of the excore monitoring system ensures that the assumptions employed in the PDC-II analysis[1] for determining AO limits that ensure operation within allowable LHR limits are valid.

3.11.2 EXCORE POWER DISTRIBUTION MONITORING SYSTEM

LIMITING CONDITION FOR OPERATION

Basis (Contd)

Surveillance requirements ensure that the instruments are calibrated to agree with the incore measurements and that the target AO is based on the current operating conditions. Updating the Excore Monitoring APL ensures that the core LHR limits are protected within the ± 0.05 band on AO. The APL considers both LOCA and DNB based LHR limits, and factors are included to account for changes in radial power shape and LHR limits over the calibration interval.

The APL is determined from the following:

$$APL = \left[\frac{LHR(Z)_{TS}}{LHR(Z)_{Max} \times V(Z) \times E_p(Z) \times 1.02}\right]_{Min} \times \text{Rated Power}$$

Where:

(1) $LHR(Z)_{TS}$ is the limiting LHR vs Core Height (from Section 3.23.1),

(2) $LHR(Z)_{Max}$ is the measured peak LHR including uncertainties vs Core Height,

(3) V(Z) is the function (shown in Figure 3.11-1),

(4) $E_p(Z)$ is a factor to account for the reduction of allowed LHR in the peak rod with increased exposure (Figure 3.23.2) such that:

For fuel rod burnups less than 27.0 GWd/MT - $E_p = 1.0$

For fuel rod burnups greater than 27.0 GWd/MT but less than 33.0 GWd/MT - $E_p = 1.0 + 0.0064 \times LHR$

For fuel rod burnups greater than 33.0 GWd/MT - $E_p = 1.0 + 0.0012 \times LHR$

Where LHR is the measured fuel rod average LHR in kW/ft,

POWER DISTRIBUTION INSTRUMENTATION

3.11.2 EXCORE POWER DISTRIBUTION MONITORING SYSTEM

LIMITING CONDITION FOR OPERATION

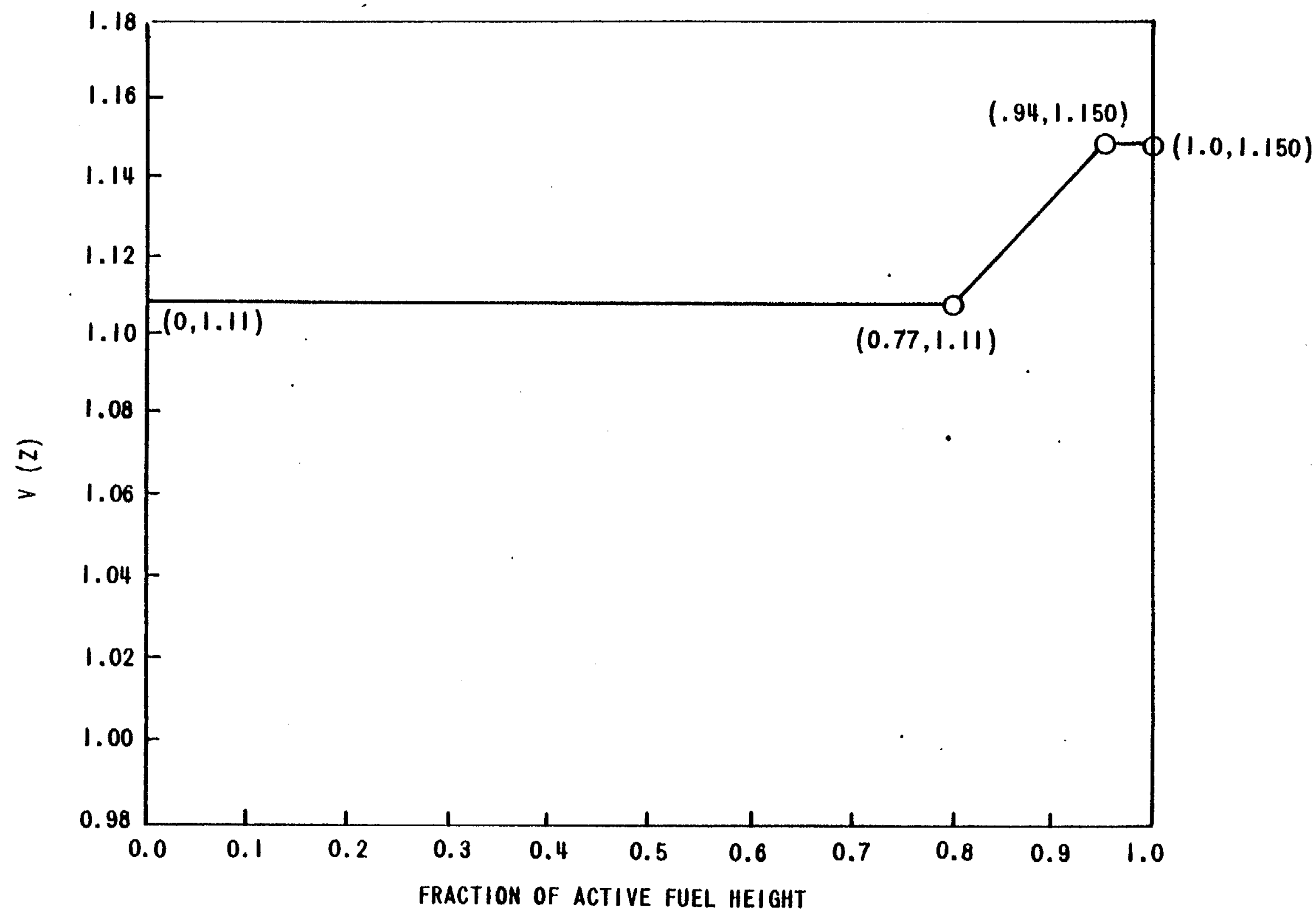
Basis (Contd)

(5) The factor of 1.02 is an allowance for the effects of upburn,

(6) The quantity in brackets is the minimum value for the entire core at any elevation (excluding the top and bottom 10% of core) considering limits for peak rods, interior fuel rods and narrow water gap fuel rods. $E_p(Z)$ is only applied if the minimum value is based on limits for the peak rod. If the quantity in brackets is greater than one, the APL shall be the rated power level.

Reference

(1) XN-NF-80-47



AXIAL VARIATION BOUNDING CONDITION	Palisades Technical Specifications	FIGURE 3.11-1

Table 3.17.4 (Cont'd)

No.	Functional Unit	Minimum Operable Channels	Minimum Degree of Redundancy	Permissible Bypass Conditions
8.	Pressurizer Water Level (LI-0102)	2	1	Not Required in Cold or Refueling Shutdown
9.	Pressurizer Code Safety Relief Valves Position Indication (Acoustic Monitor or Temperature Indication)	1 per valve	None	Not Required below 325°F
10.	Power Operated Relief Valves (Acoustic Monitor or Temperature Indication)	1 per valve	None	Not required when PORV isolation valve is closed and its indication system is operable
11.	PORV Isolation Valves Position Indication	1 per valve	None	Not required when reactor is depressurized and vented through a vent ≥ 1.3 sq. in.
12.	Subcooling Margin Monitor	1	None	Not Required Below 515°F
13.	Auxiliary Feed Flow Rate Indication	1 per Steam Generator	None	Not Required Below 325°F
14.	Auxiliary Feed Pump Auto Initiation Circuitry	1 per (e) Pump	None	Not Required Below 325°F
15.	Excore Detector	1(f)	None	None

(e) With one auxiliary feed pump automatic initiation circuit inoperable, in lieu of the requirement of 3.17.2, provide a second licensed operator in the control room within 2 hours. With both inoperable, in lieu of following the requirements of 3.17.2, start and maintain in operation the turbine driven auxiliary feed pump.

(f) Calculate the Quadrant Power Tilt using the excore readings at least once per 12 hours when the excore detectors deviation alarms are inoperable.

3.23 POWER DISTRIBUTION LIMITS

3.23.1 LINEAR HEAT RATE (LHR)

LIMITING CONDITION FOR OPERATION

The LHR in the peak power fuel rod at the peak power elevation Z shall not exceed the value in Table 3.23-1 times $F_A(Z)$ times $F_B(E)$ [the function $F_A(Z)$ is shown in Figure 3.23-1 and the function $F_B(E)$ where E is the fuel rod burnup is shown in Figure 3.23-2]. The LHR at the peak power elevation in any interior fuel rod or narrow water gap fuel rod shall not exceed the value in Table 3.23-1 times $F_C(Z)$ [the function $F_C(Z)$ is shown in Figure 3.23-3].

APPLICABILITY: Power operation above 50% of rated power.

ACTION 1:

When using the incore alarm system to monitor LHR, and with four or more coincident incore alarms, initiate within 15 minutes corrective action to reduce the LHR to within the limits and restore the incore readings to less than the alarm setpoints within 1 hour or failing this, be at less than 50% of rated power within the following 2 hours.

ACTION 2:

When using the excore monitoring system to monitor LHR and with the AO deviating from the target AO by more than 0.05, discontinue using the excore monitoring system for monitoring LHR. If the incore alarm system is inoperable, within 2 hours be at 85% (or less) of rated thermal power and follow the procedure in ACTION 3 below.

3.23.1 LINEAR HEAT RATE (LHR)

LIMITING CONDITION FOR OPERATION

ACTION 3:

If the incore alarm system is inoperable and the excore monitoring system is not being used, operation at less than or equal to 85% of rated power may continue provided that incore readings are recorded manually. Readings shall be taken on a minimum of 10 individual detectors per quadrant (to include 50% of the total number of detectors in a 10-hour period) within 4 hours and at least every 2 hours thereafter. If readings indicate a local power level equal to or greater than the alarm setpoints, the action specified in ACTION 1 above shall be taken.

Basis

The limitation on LHR ensures that, in the event of a LOCA, the peak temperature of the cladding will not exceed 2200°F.[1] In addition, the limitation on LHR for the highest power fuel rod, narrow water gap fuel rod and interior fuel rod ensures that the minimum DNBR will be maintained above 1.30 during anticipated transients; and, that fuel damage during Condition IV events such as locked rotor will not exceed acceptable limits.[2][3] The inclusion of the axial power distribution term ensures that the operating power distribution is enveloped by the design power distributions.

Either of the two core power distribution monitoring systems (the incore alarm system or the excore monitoring system) provides adequate monitoring of the core power distribution and is capable of verifying that the LHR does not exceed its limits. The incore alarm system performs this

Basis (Contd)

function by continuously monitoring the local power at many points throughout the core and comparing the measurements to predetermined setpoints above which the limit on LHR could be exceeded. The excore monitoring system performs this function by providing comparison of the measured core AO with predetermined AO limits based on incore measurements. An Excore Monitoring Allowable Power Level (APL), which may be less than rated power, is applied when using the excore monitoring system to ensure that the AO limits adequately restrict the LHR to less than the limiting values.[4]

If the incore alarm system and the excore monitoring system are both inoperable, power will be reduced to provide margin between the actual peak LHR and the LHR limits and the incore 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 manpower available, the reactor power will be reduced to a point below which it is improbable that the LHR limits could be exceeded. The time interval of 2 hours and the minimum of 10 detectors per quadrant are sufficient to maintain adequate surveillance of the core power distribution to detect significant changes until the monitoring systems are returned to service.

To ensure that the design margin of safety is maintained, the determination of both the incore alarm setpoints and the APL takes into account a measurement uncertainty factor of 1.10, an engineering

POWER DISTRIBUTION LIMITS

3.23.1 LINEAR HEAT RATES (LHR)

LIMITING CONDITIONS OF OPERATION

Basis (Contd)

uncertainty factor of 1.03, a thermal power measurement uncertainty factor of 1.02 and allowance for quadrant tilt.

References

(1) XN-NF-77-24

(2) XN-NF-77-18

(3) XN-NF-78-16

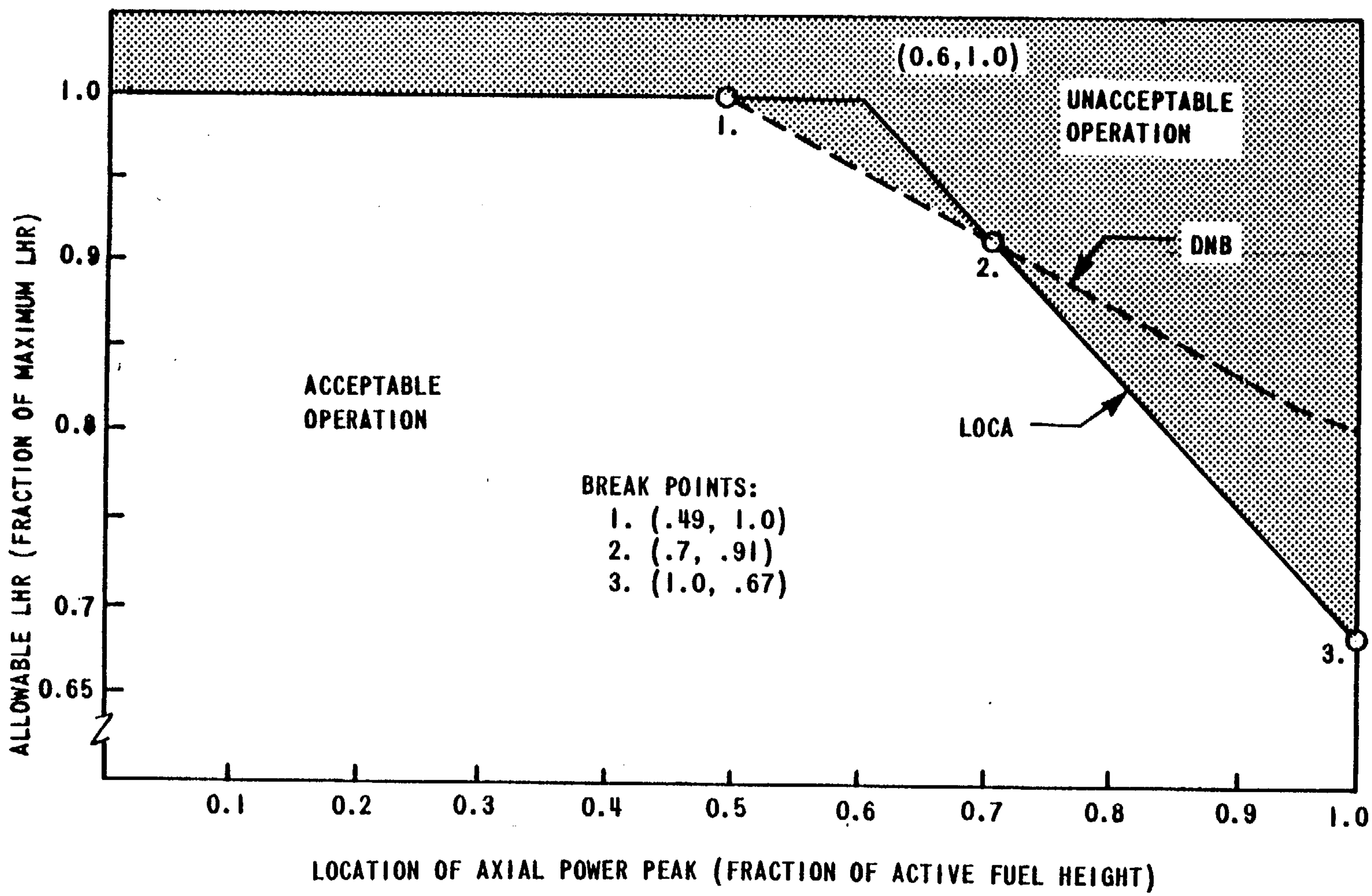
(4) XN-NF-80-47

TABLE 3.23-1
LINEAR HEAT RATE LIMITS

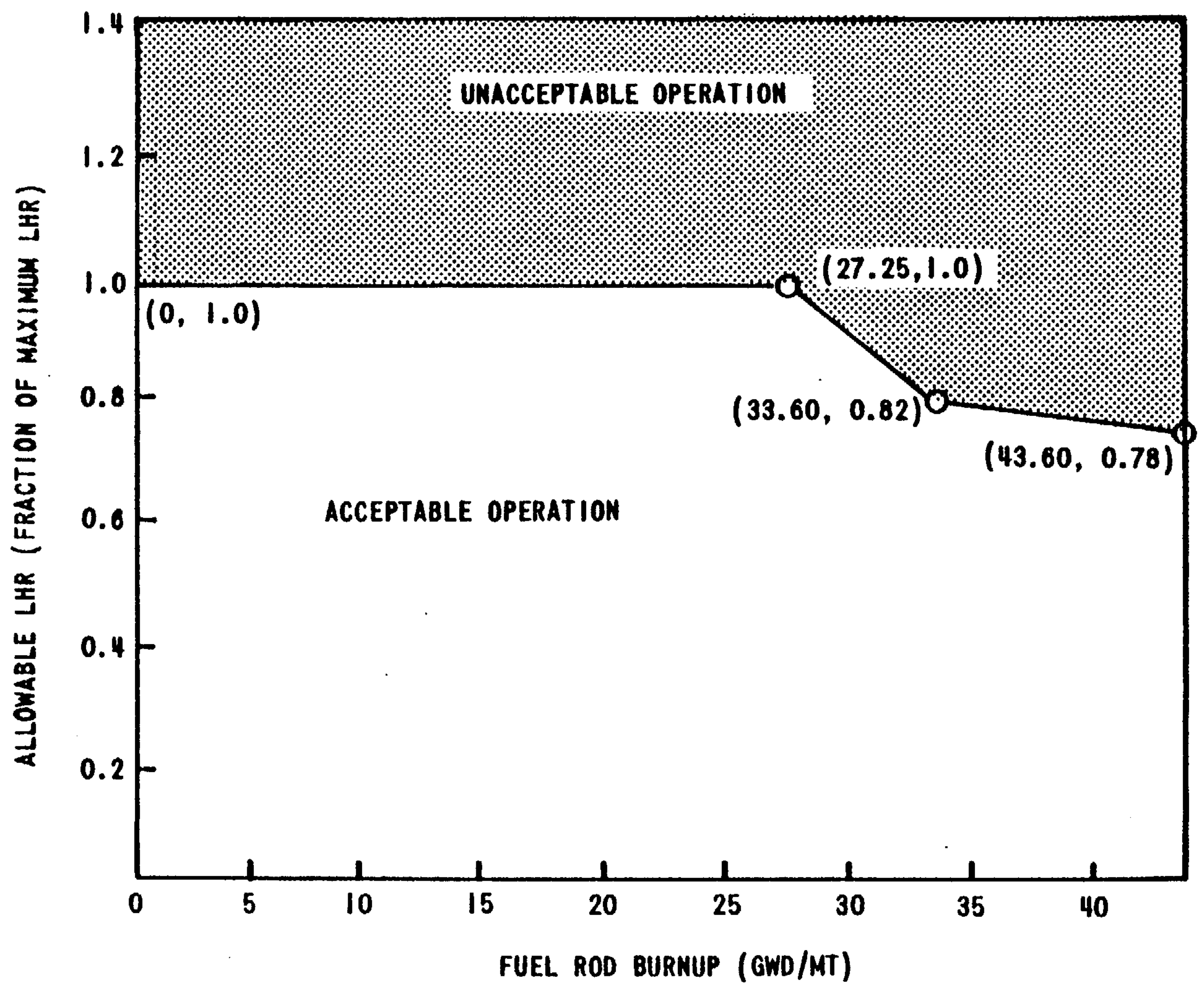
Fuel Rod Type	No of Fuel Rods in Assembly	
	208	216
Peak Rod	15.28	14.72
Narrow Water Gap Rod	15.12	14.47
Interior Rod	14.17	13.89

TABLE 3.23-2
RADIAL PEAKING FACTOR LIMITS, F_L

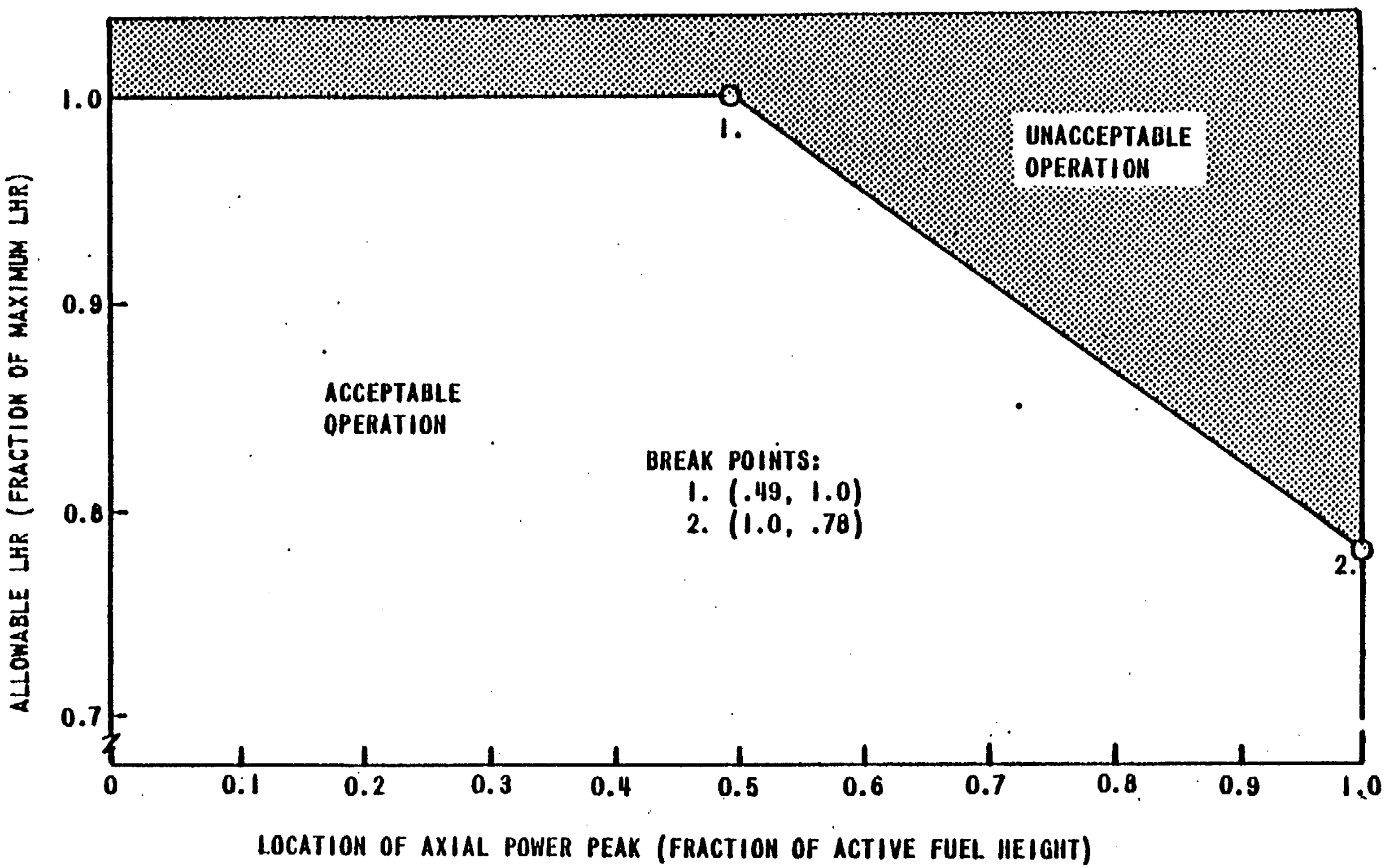
Peaking Factor		No of Fuel Rods in Assembly	
		208	216
Assembly	F_r^A	1.43	1.45
Peak Rod	F_r^T	1.77	1.77
Narrow Gap Rod	F_r^N	1.75	1.74
Interior Rod	$F_r^{\Delta H}$	1.64	1.67



ALLOWABLE LHR AS A FUNCTION OF PEAK POWER LOCATION	Palisades Technical Specifications	FIGURE 3.23-1



ALLOWABLE LHR AS A FUNCTION OF BURNUP	Palisades Technical Specifications	FIGURE 3.23-2



ALLOWABLE LHR AS A FUNCTION OF PEAK POWER LOCATION FOR INTERIOR AND NARROW WATER GAP FUEL RODS	**Palisades Technical Specifications**	FIGURE 3.23-3

3.23.2 RADIAL PEAKING FACTORS

LIMITING CONDITION FOR OPERATION

The radial peaking factors F_r^A, F_r^T, F_r^N and $F_r^{\Delta H}$ shall be less than or equal to the value in Table 3.23-2 times the quantity [1.0 + 0.5(1-P)] where P is the core thermal power in fraction of rated power.

APPLICABILITY: Power operation above 50% of rated power.

ACTION:

With any radial peaking factor exceeding its limit within 6 hours, reduce thermal power to less than the lowest value of:

$$[1 - 2 (\frac{F_r}{F_L} - 1)] \text{ x Rated Power}$$

Where F_r is the measured value of either F_r^A, F_r^T, F_r^N or $F_r^{\Delta H}$ and F_L is the corresponding limit from Table 3.23-2.

Basis

The limitations on F_r^A, F_r^T, $F_r^{\Delta H}$ and F_r^N are provided to ensure that assumptions used in the analysis for establishing DNB margin, LHR and the thermal margin/ low-pressure and high-power trip setpoints remain valid during operation. Data from the incore detectors are used for determining the measured radial peaking factors. The periodic surveillance requirements for determining the measured radial peaking factors provide assurance that they remain within prescribed limits. Determining the measured radial peaking factors after each fuel loading prior to exceeding 50% of rated power provides additional assurance that the core is properly loaded.

3.23.3 QUADRANT POWER TILT - T_q

LIMITING CONDITION FOR OPERATION

The quadrant power tilt (T_q) shall not exceed 5%.

APPLICABILITY: Power operation above 50% of rated power.

ACTION:

1. With the quadrant power tilt determined to exceed 5% but less than or equal to 10%, correct the power tilt within 2 hours or determine within the next 2 hours and at least once every 8 hours thereafter, that the radial peaking factors are within the limits of Section 3.23.2, or reduce power at the normal shutdown rate to less than 85% of rated power.
2. With the quadrant power tilt determined to exceed 10%, correct the quadrant power tilt within 2 hours after exceeding the limit or reduce power to less than 50% of rated power within the next 2 hours.
3. With the quadrant power tilt determined to exceed 15%, be in at least hot standby within 12 hours.

Basis

Limitations on quadrant power tilt are provided to ensure that design safety margins are maintained. Quadrant power tilt is determined from excore detector readings which are calibrated using incore detector measurements.[(1)] Calibration factors are determined from incore measurements by performing a two-dimensional, full-core surface fit of deviations between measured and theoretical incore readings and integrating the fitting function over each core quadrant. Values of LHR and radial peaking factors are increased by the value of quadrant tilt.

POWER DISTRIBUTION LIMITS

3.23.3 QUADRANT POWER TILT - T_q

LIMITING CONDITION FOR OPERATION

References

(1) FSAR, Section 7.4.2.2

4.18 POWER DISTRIBUTION INSTRUMENTATION

4.18.1 INCORE DETECTORS

SURVEILLANCE REQUIREMENTS

4.18.1.1 The incore detection system shall be demonstrated operable:

a. By performance of a Channel Check prior to its use following a core alteration and at least once per 7 days during power operation when required for the functions listed in Section 3.11.1.

b. At least once per refueling by performance of a Channel Calibration which exempts the neutron detectors but includes electronic components.

4.18.1.2 The incore alarm system is demonstrated operable through use of the datalogger program out-of-sequence alarm. The out-of-sequence alarm is demonstrated operable once per refueling by performance of a Channel Check.

POWER DISTRIBUTION INSTRUMENTATION

4.18.2 EXCORE MONITORING SYSTEM

SURVEILLANCE REQUIREMENTS

4.18.2.1 At least every 31 days of power operation:

a. A target AO and excore monitoring allowable power level shall be determined using excore and incore detector readings at steady state near equilibrium conditions.

b. The excore measured AO shall be compared to the incore measured AO. If the difference is greater than 0.02, the excore monitoring system shall be recalibrated.

c. The excore measured Quadrant Power Tilt shall be compared to the incore measured Quadrant Power Tilt. If the difference is greater than 2%, the excore monitoring system shall be recalibrated.

4.19 POWER DISTRIBUTION LIMITS

4.19.1 LINEAR HEAT RATES

SURVEILLANCE REQUIREMENTS

4.19.1.1 When using the incore alarm system to monitor LHR, prior to operation above 50% of rated power and every 7 days of power operation thereafter, incore alarms shall be set based on a measured power distribution.

4.19.1.2 When using the excore monitoring system to monitor LHR:

a. Prior to use, verify that the measured AO has not deviated from the target AO by more than 0.05 in the previous 24 hours.

b. Once per day, verify that the measured Quadrant Power Tilt is less than or equal to 3%.

c. Once per hour, verify that the power is less than or equal to the APL and not more than 10% of rated power greater than the power level used in determining the APL.

d. Once per hour, verify that the measured AO is within 0.05 of the established target AO.

4.19 POWER DISTRIBUTION LIMITS

4.19.2 RADIAL PEAKING FACTORS

SURVEILLANCE REQUIREMENTS

4.19.2.1 The measured radial peaking factors (F_r^A, F_r^T, $F_r^{\Delta H}$ and F_r^N) obtained by using the incore detection system, shall be determined to be less than or equal to the values stated in the LCO at the following intervals:

a. After each fuel loading prior to operation above 50% of rated power, and

b. At least once per week of power operation.

6.3.3 The Shift Technical Advisor (STA) shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the plant staff shall be maintained under the direction of the Nuclear Training Administrator and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR, Part 55.

6.4.2 A training program for the fire brigade shall be maintained under the direction of the Plant Training Coordinator and shall, as practical, meet or exceed the requirements of Section 27 of the NFPA Code.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 68 TO PROVISIONAL OPERATING LICENSE NO. 20

CONSUMERS POWER COMPANY

PALISADES PLANT

DOCKET NO. 50-255

Date: December 8, 1981

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. TO PROVISIONAL OPERATING LICENSE NO. 20

CONSUMERS POWER COMPANY

PALISADES NUCLEAR PLANT

DOCKET NO. 50-255

1.0 INTRODUCTION

By application dated July 21, 1981, as supplemented August 6, 1981, November 9, 1981, November 17, 1981, November 20, 1981, October 22, 1981, and December 2, 1981 and by telephone conversation November 18, 1981 (Reference 6), Consumers Power Company (the licensee) proposed changes to the Technical Specifications (TSs) appended to Provisional Operating License No. OPR-20 for the Palisades Nuclear Plant. The proposal requests extensive changes to the TSs on power distribution limits, control and surveillance requirements as related to the Cycle 5 reload. Included in the submittal are the Cycle 5 Reload Fuel Safety Analysis Report and the Power Distribution Control procedures.

2.0 DISCUSSION

The Cycle 5 Reload application involves fuel types previously considered for Palisades and the extension of the gadolinium lead test assembly program.

The main changes in Cycle 5 are:

(1) Discharge 68 batch G assemblies and add 68 new batch I assemblies to the core.

(2) Adopt the Constant Axial Offset Control strategy.

(3) Continue with the gadolinium bearing fuel demonstration program, with a substantial increase in gadolinium content over that which was used in Cycle 4.

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(4) Incorporate a burnup dependent TS LHR limit for all Cycle 5 fuel and future fuel types.

(5) Modify the TS LHR and radial peaking factor limits to be commensurate with the Cycle 5 core.

(6) Adopt power distribution monitoring with excore detectors as an alternative to incore alarms.

(7) Reanalyze the Steam Line Break and Rod Ejection Events. Both of these Events are analyzed using new analytical methods.

3.0 NOTATION

3.1 Acronyms

The following acronyms and abbreviations which have become "jargon of the trade" in the nuclear industry will be used throughout this report.

BNL = Brookhaven National Laboratory

BOC = Beginning Of Cycle

BP = Burnable Poison

CE = Combustion Engineering

CPC = Consumers Power Company

DBE = Design Basis Event

DNB = Departure from Nucleate Boiling

DNBR = DNB Ratio

ENC = Exxon Nuclear Company

EOC = End Of Cycle

FSAR = Final Safety Analysis Report

HFP = Hot Full Power

HHP = Hot Half Power

HZP = Hot Zero Power

LCO = Limiting Condition for Operation

LHR = Linear Heat Rate

LOCA = Loss Of Coolant Accident

LWR = Light Water Reactor

MDNBR = Minimum DNBR

MWD/MTU = MegaWatt Days per Metric Ton Uranium

NRC = Nuclear Regulatory Commission

PCPOW = PerCent POWer

PCT = Peak Clad Temperature

PDQ7 = Standard nuclear industry diffusion-depletion computer program

PLHR = Peak LHR

PSIA = Pounds per Square Inch Absolute

PSID = Pounds per Square Inch Difference

RCS = Reactor Coolant System

SAFDL = Specified Acceptable Fuel Design Limit [The SAFDLs are NRC criteria (2) and (3) for acceptable consequences of DBEs other than LOCA which appear on page 17 of this report.]

T-H = Thermal-Hydraulic

TS = Technical Specification or Technical Specifications

USNRC = United States NRC

W = Westinghouse

3.2 Notation Specific To This Report

The discussion on this report is facilitated by having some standard notations which will be explained here.

We will indicate a TS limit by enclosing the parameter to be limited in brackets ⦃ ⦄. The same brackets will be used to indicate the limit of a quantity which, itself, does not appear in the TS, but is used to compute the limit of a quantity that does appear as a TS limit (i.e., F_z).

We will use the following symbols for peaking factors:

z = Axial height in core

$\hat{z}$ = Height of axial power peak in core

Z = Height of breakpoint in TS Figures 3.23-1 or 3.23-3

L = Entire active height of core

F_r^A = Corewise assembly radial peaking factor

F_r^P = Corewise pin radial peaking factor

$F_r^{\Delta H}$ = F_r^P for pins interior to the assemblies

F_r^N = F_r^P for pins adjacent the narrow water gap

F_r^W = F_r^P for pins adjacent the wide water gap

F_r^T = F_r^P for all pins

F_ℓ = Assemblywise pin radial peaking factor

F_z = Axial peaking factor

F_Q = Total peaking factor

LHR = Linear Heat Rate

PLHR = Peak LHR

$$\text{Skewing Factor} = \frac{\text{Fraction of Core Power Above } \overset{\updownarrow}{z}}{\text{Fraction of Core Height Above } \overset{\updownarrow}{z}}$$

The TS upper limit of PLHR for the axial peak at $\overset{\updownarrow}{z}/L$ will be written $\{\!\{PLHR(\overset{\updownarrow}{z}/L)\}\!\}$. If $\overset{\updownarrow}{z}/L=70\%$ the TS limit will be written $\{\!\{PLHR(\overset{\updownarrow}{z}/L=70\%)\}\!\}$ or $\{\!\{PLHR(70\%)\}\!\}$. For $\{\!\{PLHR\}\!\}$ constant in the range of $0\leq\overset{\updownarrow}{z}/L\leq 60\%$ we will write $\{\!\{PLHR(\overset{\updownarrow}{z}/L\leq 60\%)\}\!\}$ or $\{\!\{PLHR(\leq 60\%)\}\!\}$. The same type of notation will be used for F_z.

4.0 TS CHANGES

4.1 Motivation For Making TS Changes

In References 1, 2, 3, 4, 5, and 6, Consumers Power Company requested extensive changes to the Palisades Technical Specifications on power distribution limits, control, and surveillance. These involve removing the power distribution limits from Technical Specification Section 3.10 and placing them in a new Section 3.23 adding an excore detector monitoring option to Section 3.11 and adding Sections 4.18 and 4.19 which define surveillance for the requirements of Sections 3.11 and 3.23.

These changes are intended to accomplish the following objectives:

(1) To incorporate a burnup dependent linear heat rate limit for H, I, and future fuel types.

(2) To modify the radial peaking factor limits.

(3) To adopt power distribution monitoring with the excore detectors as an alternative to incore alarms.

(4) To adopt the Standard Technical Specification format for power distribution monitoring and power distribution limits.

4.2 Evaluation of TS Changes

The specific changes requested* and their evaluations are:

A. This change involves adding three definitions to TS 1.0. The wording of these definitions is in conformance with our practices and industry standards and is, therefore, acceptable.

B. This changes the title of TS 3.10 from "Control Rod and Power Distribution Limits" to "Control Rods," and this is acceptable because it is editorial.

C. This change removes the power distribution limits from TS 3.10.3. The power distribution limits will be disclosed under Item I below. The change proposes a new TS 3.10.3 which specifies that the part length control rods will be withdrawn from the core except for control rod exercises and physics tests. The use of part length control rods has been prohibited at Palisades (and a number of other reactors) for several years because they can lead to power distributions which are not desirable. This change merely relocates the provision for prohibition of the part length control rods, and is, therefore, acceptable.

D. This change corrects three cross references in TS 3.10 to be compatible with the rest of the specifications and is, therefore, acceptable.

E. This change removes material on power distribution limits from the basis of TS 3.10. New bases are provided where needed in other TS sections.

F. This change deletes references no longer used in the references of TS 3.10.

G. This change deletes TS Figures 3-9 and 3-10 which will be replaced by TS Figures 3.23-1 and 3.23-2.

---

*The identification of each change by a letter follows the labeling of these changes in Reference 1.

H. This change deletes TS 3.11 entirely and replaces it with specifications for incore and excore instrumentation requirements for monitoring the core power distribution. The specifications for operability of the incore detector system TS 3.11 have been changed to the Standard TS format but otherwise are basically the same as in the present Palisades TS and are, therefore, acceptable.

The applicability of TS 3.11 has been expanded and made more specific than the present specification and now includes use of the incore detector system to determine the target axial offset and excore monitoring allowable power level. These functions will be discussed below.

Proposed TS 3.11.2 defines the operability requirements for the excore detector system. What is proposed allows monitoring of the LHR limits with the excore detectors as an alternative to the present incore monitoring system when operability requirements of the incores cannot be met. The method employed is based on Exxon Nuclear Company's PDC-II as reported in Reference 7. This topical report has been approved by the staff. In PDC-II, the largest peaking factor which can occur in normal operation of the power plant is determined by multiplying predetermined transient components of the peaking factor by the measured steady state peaking factors of the reactor. This is done as a function of axial height. The most limiting ratio of these peaking factors converted to LHR over LHR limit determines an allowed power level which can be permitted using the excore detectors. The active role of the excore detectors in PDC-II is to maintain operation of the reactor within a narrow axial offset band around a target axial offset. This is done because such operation is assumed in prediction of the transient component of the peaking factor.

In the proposed specifications, the target axial offset, which is the offset the reactor assumes naturally when essentially unrodded, and an allowable power level are chosen at least every 31 effective full power days of operation based on incore maps. Also, appropriate uncertainties are accounted for in the determination of allowed power level, including 2 percent for possible upburn (increase) in the radial component of the measured peaking factor between maps. Included as well is a factor for the reduction of the linear heat rate limit between maps by burnup when the limit reduction discussed under I is in effect.

The PDC-II method was originally formulated for Westinghouse reactors, so the analysis described in Reference 8 was provided by the licensee to verify the applicability of PDC-II to Palisades. We have reviewed this document and find that the analysis and model generated for Palisades, including validification of the xenon characteristics of the reactor model against experimental data, suitably verifies the transient peaking factor function.

The Palisades instrument used to measure axial offset does not have all of the indicating and alarm features normally required to use PDC-II. (Neither the target offset nor the allowable offset band width can be varied automatically with power level, and there is no timer to record time out of the target band.) Because of this, the licensee has proposed the very restrictive requirement that the allowable offset band is ±0.05, and it does not vary with power level. Because we consider these restrictions suitably conservative to compensate for the lack of the normal complement of indicating and alarm functions we find the proposed implementation of PDC-II acceptable.

Excore monitoring of the LHR using PDC-II is proposed as an alternate to monitoring with the incores when the datalogger is inoperable. If PDC-II were proposed as the primary or sole means for monitoring, we would require updating of the axial offset indicating and alarm system to be compatible with other reactors using the method. This would also allow less stringent PDC-II specifications.

If neither the incore or excore monitoring systems satisfy operability requirements, the proposed TS retain the existing alternative wherein the reactor power is limited to 85 percent and incore outputs are recorded by hand.

I. This change defines the LHR, radial peaking factor, and quadrant tilt limits for operation of the Palisades reactor. In TS 3.23.1, the currently approved LHR limits are specified, except they are modified at burnups in excess of 27.25 GWd/MT (reduction) factor specified by TS Figure 3.23-2. This factor offsets the adverse effects of fission gas release on predicted clad rupture and flow blockage during the LOCA. The computation of this factor is described in Appendix A of Reference 9 (herein called the Cycle 5 Safety Report). The factor was calculated with approved methods and is, therefore, acceptable. Allowable LHR limits are also modified by TS Figures 3.23-1 and 3.23-3 which limits the PLHR as a function of height of the axial power peak in the core. The development of these figures is described in Reference 10 (herein called the Axial Shape Report). The computations described in the Axial Shape Report were done using approved methods, and therefore, the resultant TS figures are acceptable.

The power level at which the LHR must be monitored is proposed to be 50 percent. The existing specification allowed operation up to 65 percent without incores. The proposed change is thus more conservative and, therefore, acceptable. The action and applicability statements in TS 3.23.1 are in conformance with the incore and excore LHR monitoring methodology discussed under Item H above and are, therefore, acceptable.

In TS 3.23.2, limits for F_r^A, $F_r^{\Delta H}$, F_r^N, and F_r^T are given. These are in conformance with the assumptions used in the Cycle 5 reload analysis and are, therefore, acceptable. The action and surveillance requirements are similar to the existing specifications and are, therefore, acceptable.

In TS 3.23.3, limits for allowable quadrant tilt are proposed. The basic specification that the tilt be maintained under 5 percent is the same as before. Proposed action statements are more conservative than before, which is acceptable, or in the case of large tilts, a requirement is made to be in hot standby within 12 hours, which is in keeping with the Standard TS and, therefore, is acceptable.

J. TS 4.18 and 4.19 are proposed which define surveillance requirements for TS 3.11 and 3.23. We have reviewed these surveillance requirements and find them compatible with the requirements discussed under H above, or the same as surveillance requirements in the Standard TS. The proposed surveillance TS are, therefore, acceptable.

K. This proposed TS adds a requirement for excore detector deviation alarms to the instrumentation LCO table, and specifies manual calculation of the quadrant power tilt once per 12 hours when the alarms are inoperable. Addition to this requirement will improve the operator's ability to detect quadrant power tilts. This change is, therefore, acceptable.

4.3 Findings of Review of TS Changes

We find the proposed TS changes will continue to maintain safety margins established by previous analyses for the Palisades power plant. Specifically, the changes to the LHR monitoring specifications will continue to maintain the LHR below the values used as initial assumptions in the LOCA analysis performed in accordance with 10 CFR 50.46 Appendix K. Additionally, the LHR and radial peaking factor monitoring specified will continue to maintain safety margins on DNBR during steady state, load follow, and anticipated transient operation of the Palisades reactor during Cycle 5.

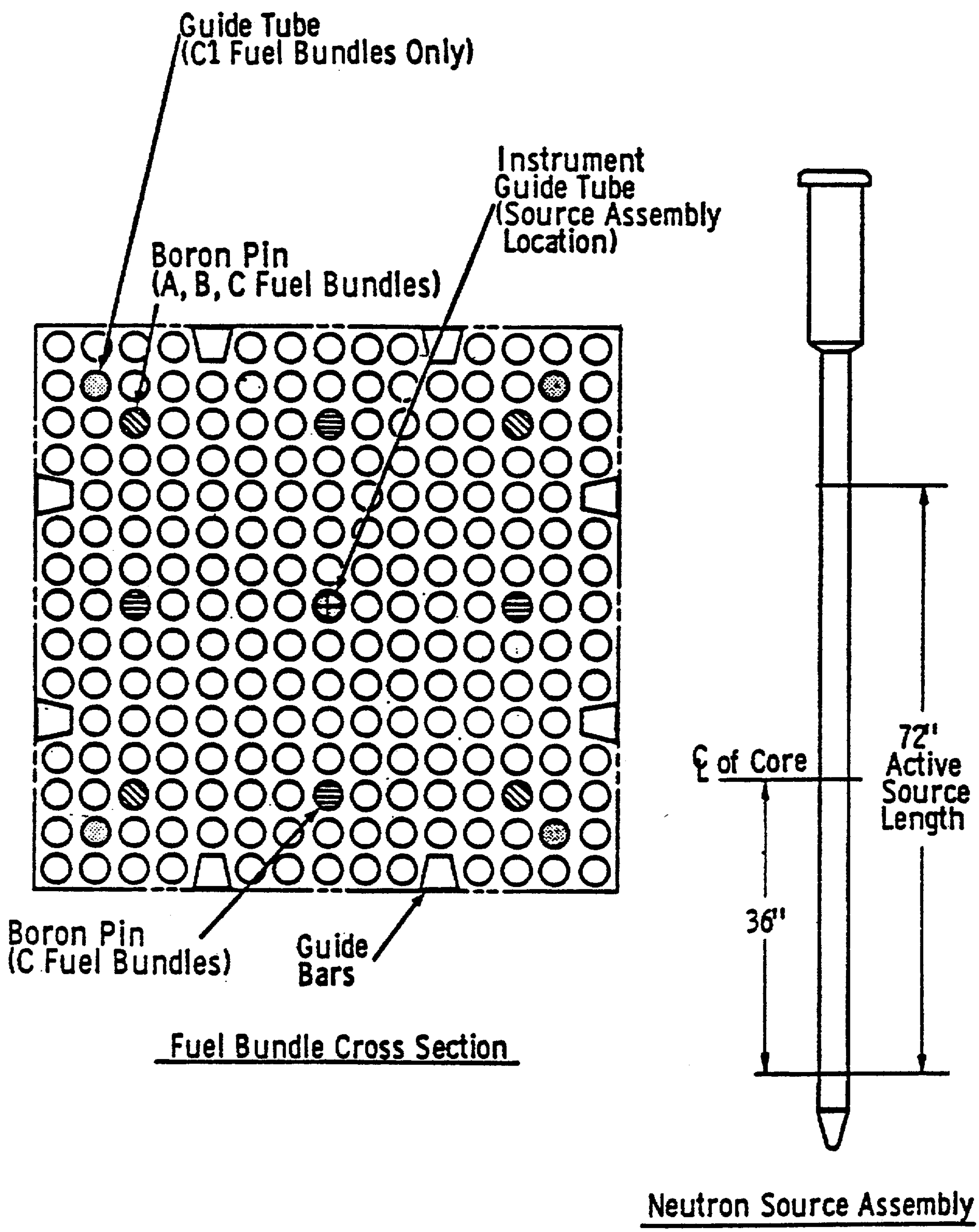
We, therefore, find the proposed TS changes acceptable.

5.0 CYCLE 5 FUEL DESIGN

5.1 Fuel Mechanical Design

All fuel assemblies in the Palisades core have the design shown in Figure 5.1-1. Pertinent design parameters are shown in Table 5.1-1. Batches G, H, and 56 of the batch I assemblies contain 208 fuel pins per assembly. The twelve batch I assemblies containing gadolinium have 216 fuel pins per assembly. These 12 assemblies have a geometry similar to the 208 pin assemblies except that the BP sleeves are replaced with fuel pins. 60 of the batch I assemblies contain fuel pins 2 mils larger in diameter than the rest of the fuel in the

Figure 5.1 Cross Section of Palisades Fuel Assembly
[Reproduced from the FSAR]



Guide Tubes and Source Assemblies Palisades Core	COMBUSTION ENGINEERING, INC. WINDSOR, CONNECTICUT	Figure 3-43

Rev 4/10/69

Table 5.1-1 Fuel design summary
[Reproduced from Cycle 5 Safety Report]

Reload design	G	H	I
Number of assemblies	68	68	68
Initial average enrichment (%)	3.00	3.27	3.25
Pellet density (% TD)	94.0	94.0/94.75*	94.0
Pellet clad gap (in)	0.0075	0.0080	0.0080
Fill gas pressure (psia He)	300	321	321
Wall thickness (in)	.0285	.0295	.0295
Cladding outside diameter (in)	0.417	0.417	0.417/0.419**
Number of assemblies with $B_4C-Al_2O_3$ burnable poison	20	16	8
$B_4C-Al_2O_3$ rods/assembly	8	8	8
Poison loading, gm B10/in	0.0204	0.0204	0.0204
Number of assemblies with Gd_2O_3 burnable poison	8	4	12
Urania/gadolinia rods/assembly	4	8	8
Wt. % Gd_2O_3	1.00	4.0	4.0
BOC 5 batch average exposure (MWD/MT)	21,640	10,090	0

*Gadolinia bearing rods only
**Assemblies fabricated from new fuel pins only

core. As explained in Section 8.3.1 this will give these assemblies better LOCA performance. As can be seen in Table 5.1-1 the enrichment of batch I fuel lies between the enrichments of batches G and H fuel. The external chemical composition of the batch I fuel is identical to that of batches G and H fuel (Q&A 5 of Reference 6), and thus the batch I fuel will not be a source of excessive corrosion, stress corrosion cracking, or crud formation. Other than some reservations on the gadolinium bearing fuel pins expressed in Section 5.5, the batch I fuel is very similar in mechanical design to the batches G and H fuel, which we have already approved, and have withstood service without any deliterious effects. On this basis we approve the mechanical design of the batch I fuel.

5.2 Cladding Creep Collapse

The cladding creep collapse analysis using the approved COLAPX code (Reference 11) for batch H fuel showed that collapse would not occur until reaching an assembly burnup of at least 37,000 MWD/MTU. Since the target lead assembly of Cycle 5 is 35,000 MWD/MTU, creep collapse is not expected to occur.

5.3 Fission Gas Release

Palisades has used the approved Exxon thermal code GAPEXX (Reference 12) with the NRC correction for enhanced fission gas release (Reference 13). This correction increases the cladding temperature, which adds conservatism to the computation. We find this an acceptable method for computing the burnup effects on fission gas release and on the resultant change in thermal performance of the fuel.

5.4 Cycle 4 Fuel Failures

During Cycle 4 operation, Palisades experienced a small number of fuel failures. The subsequent visual inspection of the discharged assemblies (Batches D and E) revealed only a small hole in a fuel rod of Batch E. Although the failure site appeared to be a hydriding failure, Palisades suspected that a manufacturing defect on the cladding outer surface had caused water penetration during the early Cycle 4 operation. Palisades has stated that Exxon, the fuel manufacturer, will examine all eight assemblies of Batch E fuel by periscope and underwater closed-circuit TV for causes of failure. As for CE fuel of Batch D, Palisades has so far no plan for examination since the core of Cycle 5 contains only Exxon fuel. Palisades claimed that the very small number (5 to 10 rods out of total about 41,000 rods) of failed fuel rods during Cycle 4 should have no safety concerns for Cycle 5 even though Palisades could not preclude the possibility of failed fuel rods in other Batches, G and H, which will reside in the core of Cycle 5.

On the basis that (a) Palisades will make a reasonable attempt to find the cause of the fuel rod failures in the near future, (b) the discharged CE and Exxon fuel will not be returned to the core for Cycle 5 operation, thus eliminating the possibility of the failed Exxon fuel residing in the core, and (c) the probability of additional failed fuel rods in Batches G and H of Cycle 5 is small (only one such failure is known to have occurred in Cycle 4), we conclude that the issue of fuel failures during Cycle 4 operation has been adequately addressed.

5.5 Gadolinium Fuel Demonstration Program

5.5.1 Advantageous Properties of Gadolinium as a BP

All commercial power reactors contain BP fuel pins for the purpose of improving the power shape. In the recent past most pressurized water reactors have used boron as the BP. Because of certain advantages of using gadolinium as a BP, Exxon embarked on an experimental gadolinium program in cycles 3 and 4 of the Palisades reactor. In these two cycles the gadolinium behaved as predicted, giving Exxon and Palisades the impetus to go to a cycle 5 core design which would optimize the use of gadolinium as a BP and provide a model for future reload cores.

The reasons that gadolinium is preferable to boron as a BP is that it is a better neutron absorber than boron and it burns out faster than boron. This gives gadolinium the following advantages over boron as a BP:

(1) Gadolinium burns out somewhat faster than the fuel. Because of this, during the early part of their service life the gadolinium bearing assemblies may maintain the same reactivity, or even increase slightly in reactivity. This helps achieve a flatter core power shape throughout the cycle than is possible using boron as the BP.

(2) Since most of the gadolinium has burned out by the end of a cycle, a gadolinium bearing core has more reactivity than a boron bearing core toward the end of a cycle, which makes it possible to stretch the length of a cycle.

(3) Boron is a relatively weak neutron absorber, and whole fuel pins must be replaced by boron pins for the boron to be an effective BP. By comparison gadolinium is a very strong neutron absorber, and a gadolinium bearing pin which contains the usual amount of nuclear fuel plus 4% Gd_2O_3 is an effective BP pin. Thus all the pins in a gadolinium bearing assembly are active fuel pins. This increases the total U235 core loading which helps to extend the cycle.

(4) Because the gadolinium bearing assemblies contain all active fuel pins, as explained in (3) above, the LHR of the active fuel pins can be made smaller without decreasing the assembly power.

5.5.2 Neutronic Computations With Gadolinium Fuel

Comparisons of standard Exxon PDQ7 computations with other computational methods and experimental data are presented in the Cycle 5 Safety Report, Appendix B, and Reference 14. The comparisons in both these reports indicate that Exxon PDQ7 predicts gadolinium bearing assembly powers with a bias of about -3%, without much scatter after the bias is corrected. The Palisades Startup Report will be submitted to the NRC within 90 days after the commencement of Cycle 5, and this report will contain a comparison of the Exxon PDQ7 computations with measured assembly powers at the Cycle 5 startup (Verbal commitment). If the -3% computational bias is seen in the Cycle 5 startup, there may be justification for pressing Exxon to seek out the source of this bias and correct their PDQ7 model.

The ultimate safety of the core power shape is dependent on observing the TS limits and from this point of view a miscalculation of -3% in the reload corewise assembly power calculation has no safety significance.

However, if it is also the case that the INCA constants are being computed incorrectly, then the measured INCA pin powers may well be biased low. This would be a nonconservative situation, and would require correction.

For the moment, data is very scant, and it is difficult to draw any positive conclusions. The best plan would be to wait for the Cycle 5 Startup Report, and if the -3% computational bias is present there as well, we may wish to investigate the matter further.

5.5.3 Other Properties of Gadolinium Fuel

An investigation of the non-neutronic properties of gadolinium fuel is described in Reference 15, and the approval of this report is given in Reference 16. Fuel properties which were considered in this report include (1) melting point, (2) theoretical density, (3) specific heat, (4) thermal diffusivity, (5) thermal conductivity, (6) thermal expansion, (7) densification, (8) fuel swelling, (9) axial gapping, (10) fission gas release, and (11) homogeniety. The investigation of all except the following properties is considered satisfactory: (1) densification, (2) fission gas release, and (3) fuel cladding chemical interaction. We have asked Exxon to prepare an information-gathering program that would acquire the needed information in these areas in a timely fashion.

5.5.4 Approval of Gadolinium Fuel Program

Since the use of gadolinium bearing fuel is still in the experimental stage, there is still a substantial amount of information we wish to gather regarding the use of gadolinium bearing fuel. However, from the experience gained thus far, no safety related issues have surfaced, and on this basis we approve the continuance of the gadolinium fuel demonstration program into Cycle 5.

6.0 BASIS FOR COMPARING CYCLE X PARAMETERS WITH REFERENCE ANALYSIS PARAMETERS: BRIEF REVIEW OF PERTINENT TOPICAL REPORTS

6.1 Original 2530 MWT DBE Analyses

The original DBE analyses performed to support Palisades operation at 2530 MWT in Cycle 2 are described in Reference 17 (herein called the Transient Analysis Report) (analyses described in the Transient Analysis Report will herein be called "reference transient analyses") and in Reference 18 (herein called the 2530 LOCA Report). The peaking factors for these analyses are given in Table 6.1-1.

The NRC criteria for acceptable consequences of DBEs other than LOCA are as follows:

(1) Less than one percent fuel damage during any low probability (Condition IV) event.

(2) A 95% probability at a 95% confidence level that no fuel pin will undergo DNB during normal operation or anticipated (Condition II or III) event. (With the DNBR correlations used by Palisades, this statement is equivalent to MDNBR $\geq$ 1.30.)

(3) The fuel temperature should not exceed the fuel melting temperature during normal operation or anticipated transients.

(4) Peak transient vessel pressure less than 2750 PSIA.

(5) Peak primary to secondary differential pressure less than 1530 PSID during normal operation and anticipated transients.

All transients analyzed in the Transient Analysis Report met the above criteria. The only transient to result in a MDNBR $\leq$ 1.30 was the Locked Rotor, which is a Condition IV Event.

Table 6.1-1 2530 MWT analyses input peaking factors

	Transient analysis report		2530 LOCA report		Axial Shape Report			
	Transient analyses		LOCA analysis		Transient analyses		LOCA analyses	
[F_r^A]	1.45**		1.40**		1.45		1.45	
[F_r^P]	1.77		1.83		1.77		1.77	
[F_z]	1.40		1.40		1.45		1.51	
$\overset{\uparrow}{Z}/L$	60%		60%		60%		60%	
Z/L	--		--		50%		60%	
[F_Q]	2.55		2.64		2.76		2.76	
	208 PA*	216 PA	208 PA	216 PA	208 PA	216 PA	208 PA	216 PA
[PLHR]	14.12	13.60	14.68	14.12	15.28	14.12	15.28	14.12

*208 PA = 208 Pins per Assembly

**These different limits for the transient analyses and LOCA analysis were rectified in the Axial Shape Report.

In the Transient Analysis Report, only those events for which the FSAR analysis or some other previous analysis was not bounding at 2530 MWT were reanalyzed. The reanalyses described in the Transient Analysis Report and the justification for not including the reanalyses of the other previously analyzed events were approved in the NRC Safety Evaluation accompanying Amendment No. 31 to the Palisades Operating License No. DPR-20 (Reference 19). The reason that certain previously analyzed analyses were bounding for Cycle 2 operation at 2530 MWT are listed in Table 4.0-3 of the NRC Safety Evaluation, which is reproduced here as Table 6.1-2. The previous analyses of these events remain bounding for Cycle 5 operation for the same reasons as they did for Cycle 2 operation.

Palisades has taken the position that small break LOCAs are not limiting, and thus do not require analysis. This position was approved by the NRC with the approval of Palisades License Amendment No. 31 (Reference 19). Further strengthening of this position is provided by Reference 20, which demonstrates generically that small break LOCAs are not limiting for Palisades, Fort Calhoun, Millstone, Calvert Cliffs, Saint Lucie, and ANO2 Unit 2 plants.

The large break LOCA analysis met the NRC criteria, which are as follows:

(1) The calculated peak fuel clad temperature does not exceed 2200 DEGF.

(2) The amount of fuel cladding that reacts chemically with water or steam does not exceed 1% of the total amount of zircaloy in the reactor.

(3) The cladding temperature transient is terminated at a time when the core geometry is still amenable to cooling. The hot fuel rod cladding oxidation limits of 17% are not exceeded during or after quenching.

(4) The system long term cooling capability provided for previous fuels remains applicable for ENC fuel.

On these bases the NRC accepted the Transient Analysis Report and the 2530 LOCA Report.

Table 6.1-2 Page reproduced from Safety Evaluation accompanying Amendment 31 to Palisades Operating License No. DPR-20

TABLE 4.0-3

Transients and Accidents not Reanalyzed in Transient Analysis Report

Incident	Reason not reanalyzed
Boron dilution	At startup or refueling the FSAR analysis is still bounding. At power, the incident is bounded by the Rod Withdrawal incident.
Steam generator tube rupture	The FSAR analysis, done at 2650 MWt, is bounding.
Turbine generator overspeed	The FSAR analysis is still valid since it is not affected by the power increase.
Fuel handling accident	A bounding analysis was performed in connection with the spent fuel pool storage expansion approved by us in a license amendment issued on June 30, 1977.
Idle loop startup	Startup of the reactor is not permitted with less than 4 pumps in operation.
Malpositioning of part-length control rod group	Operation of the reactor is permitted only with the part-length control rods completely withdrawn from the core.

6.2 TS LHR Limitation Prior to Cycle 3 and Resolution of This Limitation

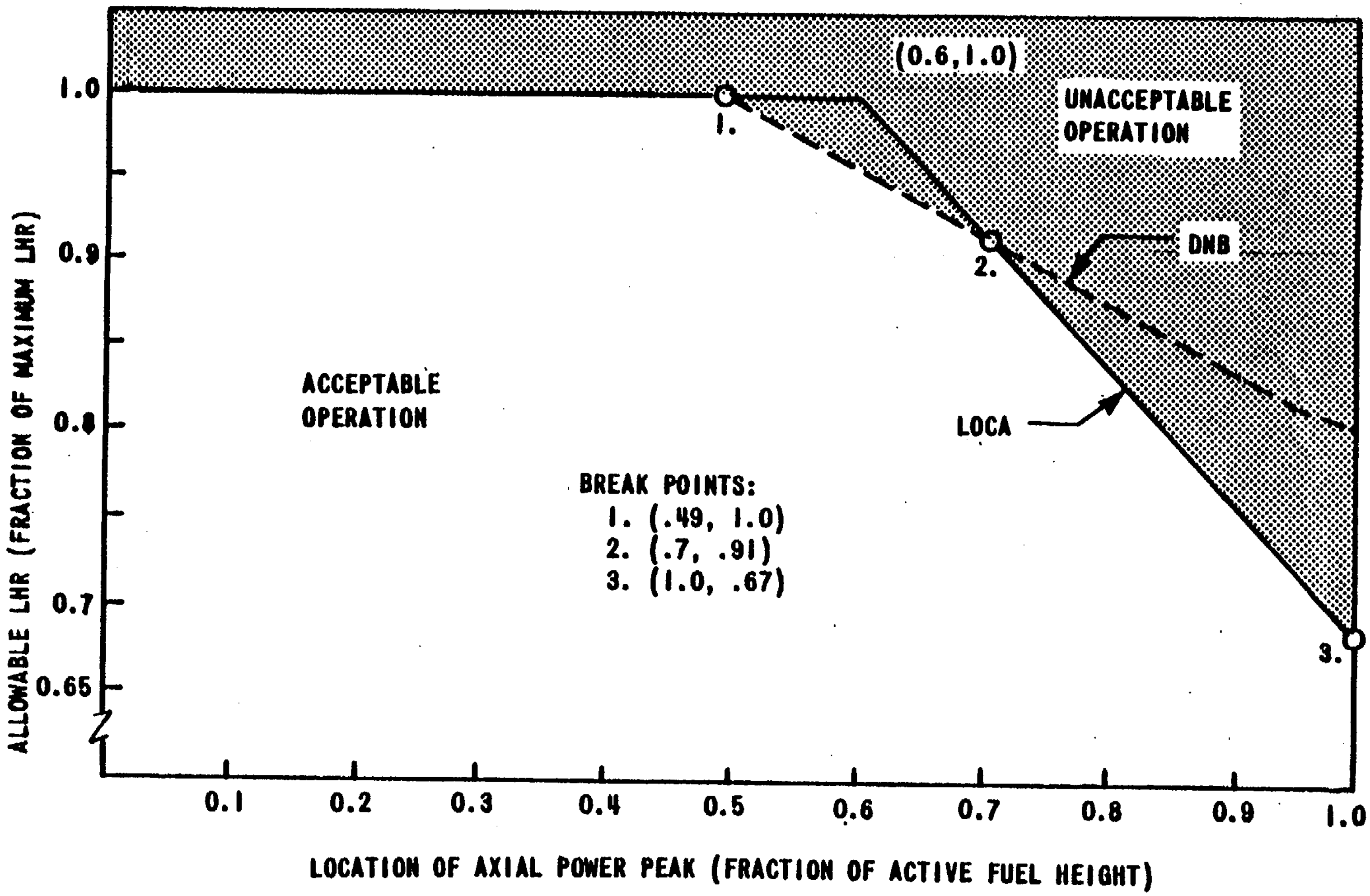
Prior to Cycle 3 the TS contained an LHR limit which was a function of z. The bounding TS curve was extremely restrictive near the top of the core, particularly at EOC when the axial power profile assumed a saddle shape. This restriction was so severe that it forced a power derate near EOC. In order to gain more operating flexibility, during Cycle 3 extensive transient and LOCA reanalyses were performed which are discussed in the Axial Shape Report. This analysis was the foundation of a new TS LHR limit in which PLHR was specified as a function of $\overset{\uparrow}{z}$. That is, with the new TS LHR limit, the LHR was limited at only one axial point. This new TS LHR limit provided much greater operating flexibility than the old TS which specified an LHR limit along the whole length of the core.

TS Figures 3.23-1 and 3.23-3, which are included here, define the new TS PLHR limit.

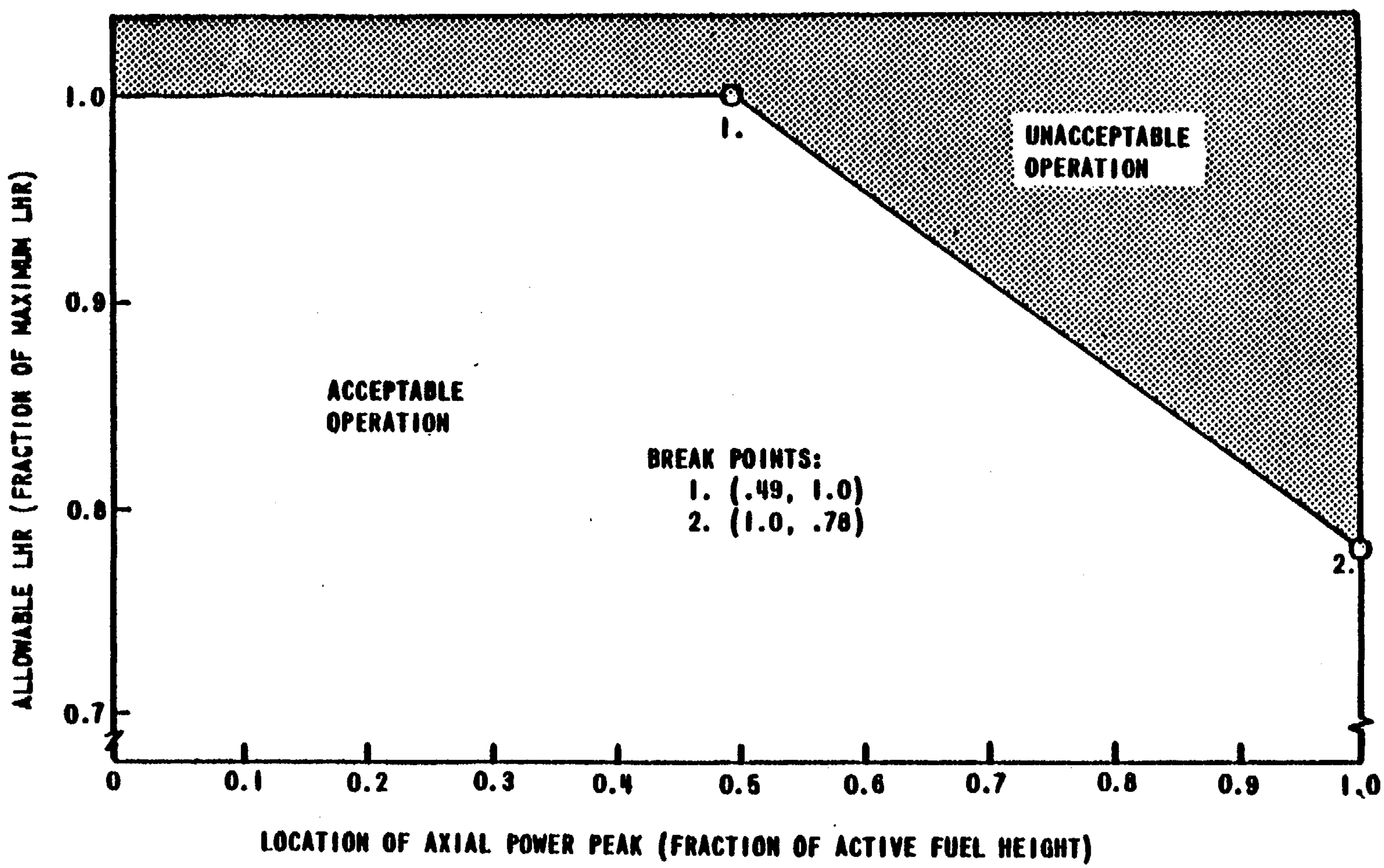
6.3 New TS LHR Limits: TS Figure 3.23-3: Transient Analyses With $\overset{\uparrow}{z}/L > 50\%$, TS Figure 3.23-1: LOCA Analyses With $\overset{\uparrow}{z}/L > 60\%$

From past experience it was well known that if $\{PLHR(\overset{\uparrow}{z}/L)\}$ were computed for $\overset{\uparrow}{z}$ ranging from L to 0, $\{PLHR(\overset{\uparrow}{z}/L)\}$ would increase monotonically as $\overset{\uparrow}{z}$ decreases. As $\overset{\uparrow}{z}$ would decrease from approximately the middle of the core, the increase in $\{PLHR(\overset{\uparrow}{z}/L)\}$ would provide only a small increase in operating flexibility. Thus below some value of $\overset{\uparrow}{z}/L$ the increased operating flexibility provided by a higher $\{PLHR(\overset{\uparrow}{z}/L)\}$ becomes worth less than the computation required to justify this higher $\{PLHR(\overset{\uparrow}{z}/L)\}$. For the transient analyses it was decided to compute $\{PLHR(\overset{\uparrow}{z}/L)\}$ only for $\overset{\uparrow}{z}/L \geq 50\%$, and for LOCA analyses it was decided to compute $\{PLHR(\overset{\uparrow}{z}/L)\}$ only for $\overset{\uparrow}{z}/L \geq 60\%$. Below these $\overset{\uparrow}{z}/L$ values the TS figures dictate a constant value for $\{PLHR\}$ equal to the highest computed value of $\{PLHR\}$. That is, for the transient analysis TS $\{PLHR(\overset{\uparrow}{z}/L < 50\%)\} = \{PLHR(\overset{\uparrow}{z}/L = 50\%)\}$, and for the LOCA TS $\{PLHR(\overset{\uparrow}{z}/L < 60\%)\} = \{PLHR(\overset{\uparrow}{z}/L = 60\%)\}$. This can be seen in TS Figures 3.23-1 and 3.23-3. In the remainder of this section we will call the $\overset{\uparrow}{z}/L$ breakpoint in these TS figures (50% or 60%) Z/L.

In this discussion only the LOCA limit in this figure is pertinent. Ignore the DNB limit.



ALLOWABLE LHR AS A FUNCTION OF PEAK POWER LOCATION	Palisades Technical Specifications	FIGURE 3.23-1



ALLOWABLE LHR AS A FUNCTION OF PEAK POWER LOCATION FOR INTERIOR AND NARROW WATER GAP FUEL RODS	**Palisades Technical Specifications**	FIGURE 3.23-3

In the Axial Shape Report, before analyses began F_r^A was set at 1.45 and F_r^P was set at 1.77. Using these radial peaking factors the transient and LOCA analyses were performed with a number of axial power profiles with $\hat{Z}/L \geq Z/L$. These analyses showed that once $\hat{Z}/L$ was chosen, the value of F_z required to satisfy the appropriate criteria for acceptance of the analysis was relatively insensitive to other details of the power profile, and insofar as the required F_z did depend on other details of the power profile, the effect of these details of the power profile on the required F_z could be reasonably estimated. This made it fairly easy to compute a bounding F_z which is a function of only $\hat{Z}/L$. This provided $\{F_z(\hat{Z}/L)\}$ which was used to compute $\{PLHR(\hat{Z}/L)\}$ = Const * $\{F_r^P\}$ * $\{F_z(\hat{Z}/L)\}$. That portion of TS Figures 3.23-1 and 3.23-3 above Z/L was constructed from the computed values of $\{PLHR(\hat{Z}/L)\}/\{PLHR(Z/L)\}$.

6.4 Reference 2530 MWT Transient Analyses

In Section 2 of Reference 20 it is shown that for the power shapes and coolant conditions used in the Transient Analysis Report, the reactor would have a steady state MDNBR = 1.30 at 115 PCPOW. This means that the transients analyzed in the Transient Analysis Report produce a DNBR degradation no greater than would be produced by running at 15% overpower. The correctness of this statement for power shapes other than those assumed in the Transient Analysis Report is provided by the analyses in the Axial Shape Report in which all power profiles for transient analyses are selected to conform to the criterion MDNBR = 1.30 at 115 PCPOW. The transient analyses in the Transient Analysis Report were reanalyzed using this conglomorate of power shapes, and in all cases the results met the NRC transient analysis acceptance criteria of Section 6.1.

The value $\{PLHR(\hat{Z}/L = 60\%)\}$ = 14.64 KW/Ft and TS Figure 3.23-3 were both generated from analyses in the Axial Shape Report, and any transient reanalysis should encompass a reevaluation of both these items. To date in Palisades reload reports the value of $\{PLHR(\hat{Z}/L = 60\%)\}$ has been reevaluated, but no check of TS Figure 3.23-3 has been performed. For at least a few future reloads the value of $\{PLHR(\hat{Z}/L)\}$ for some $\hat{Z}/L \neq 60\%$ should be evaluated and the ratio between the two $\{PLHR\}$ compared with TS Figure 3.23-3. In Section 6.9 one such example of a check on TS Figure 3.23-1 is described.

6.5 Computation of Assemblywise Pin Radial Peaking Factor, $\ddagger F_r^P \ddagger$, For Cycle X

For every cycle the transient analysis peaking factors (LCOs which appear in the TS) are chosen so that at 15% overpower and "Design" coolant conditions the MDNBR is 1.30. "Design" coolant conditions are the most adverse coolant conditions of RCS flow, pressure, and inlet temperature allowed by the TS during normal operation. The sections of the TS which delineate allowed coolant conditions do not change from cycle to cycle. The design coolant conditions and Cycle 5 peaking factors are given on page 38 of the Cycle 5 Safety Report.

A few words are in order to explain how the Cycle X peaking factors are computed. Normally $\ddagger F_r^A \ddagger = 1.45$ is taken from TS 3.23.2. (However, as explained in Section 6.9, this was not done in the Cycle 5 analyses.) The maximum power assembly for the core in question is modeled in an infinite sea of similar assemblies and the pin by pin power distribution for that assembly is determined by a 2D PDQ. The assembly $\ddagger F_\ell \ddagger$ is computed and $\ddagger F_r^P \ddagger$ is computed from $\ddagger F_r^P \ddagger = \ddagger F_r^A \ddagger * \ddagger F_\ell \ddagger$. The HFP axial power profile is not computed, but rather the power profile from page 12 of the Axial Shape Report is used. This profile has $\overset{\uparrow}{z}/L = 60\%$, $F_z = 1.45$, and a skewing factor of 1.1.

Using the design RCS conditions, the above assembly pin power distribution, the above axial power profile, and a power of 115 PCPOW, the T-H code COBRA (Reference 12) is run to find MDNBR. Invariably MDNBR turns out to be greater than 1.30. The power in the hottest pin is raised until COBRA computes MDNBR = 1.30. The F_ℓ for this new assembly pin power distribution is computed and the new F_ℓ times the original F_r^A is taken as $\ddagger F_r^P \ddagger$.

The Cycle 5 Safety Report failed to address the DNBR reduction from fuel rod bowing. Consequently, we have applied the current NRC-approved interim method (Reference 22) to evaluate the effect of fuel burnup on rod bowing and the corresponding DNBR reduction. With the DNBR reduction from fuel rod contact calculated with the methods of Reference 23, the interim method uses linear interpolation between zero and full contact in determining the rod bow penalty for partial gap closure. The licensee has pointed out that this method is too conservative based on data reported in the open literature (Reference 24) which shows that no DNBR reduction is observed for gap closure less than 50 percent. We have previously approved modification of the interim rod bow methods for Westinghouse fuel assemblies (Reference 25) to take credit for these data. We, therefore, conclude that the data are applicable to Palisades and may be used for the Palisades fuel rod bowing penalty calculation. Based on the current Palisades over pressure trip setpoint of 2255 psia and the hot pin average heat flux of 0.3×10^6 Btu/hr-ft^2, we have constructed Table 6.5-1 showing gap closure and corresponding DNBR reduction (taking credit for the cited data) as a function of fuel burnup. Since the maximum fuel burnup for Palisades Cycle 5 is 35,000 MWD/MTU, the maximum gap closure will be 47.4 percent and no DNBR reduction is required.

Based on this evaluation, we conclude that rod bow compensation is not required for Cycle 5 and results in the Cycle 5 Safety Report are valid. The safety analysis has shown that the Palisades Cycle 5 core satisfies the SAFDL criteria. We, therefore, conclude that the proposed Palisades Cycle 5 operation is acceptable.

6.6 Criteria For Determining Which Transient Events Require Reanalysis For Cycle X

In most transients the core is represented by a point kinetics model. For these cases the course of the transient does not depend on the detailed geometry of the core, but only on the point kinetics reactivity parameters, the rod drop time, and the shutdown margin. If these parameters are no more adverse for Cycle X than for the reference transient analysis, then the reference transient analysis bounds the transient analysis for Cycle X. Transient analyses which must account, at least in some measure, for three dimensional effects are the Dropped Rod Event, the Ejected Rod Event, the Single Rod Withdrawal Event, and the Steam Line Break Event. For these events, the power peaking in the core must be computed, and thus for these events the parameters which affect power peaking, as well as the reactivity parameters, must be shown to be less adverse for Cycle X than for the reference analysis to be bounding. In Section 7 some details on the direction in which various parameters must change in order to make the consequences of various transients more adverse are given.

Table 6.5-1 Rod Bow Penalty

Assembly burnup GWD/t)	Gap closure (%)	DNBR reduction (%)
0	0	0
10	30.0	0
20	38.3	0
30	44.6	0
35	47.4	0
40	50.0	0
50	54.7	1.6
60	59.0	3.0

With the following three changes made the analyses in the Transient Analysis Report can be considered to be the reference analyses for Cycle X:

(1) $\llbracket F_z(\overset{\uparrow}{z}/L = 60\%)\rrbracket = 1.45$.

(2) $\llbracket F_r^P \rrbracket$ must be computed using the axial power profile on page 12 of the Axial Shape Report, rather than the axial power profile on page 13 of the Transient Analysis Report.

(3) $\llbracket PLHR(\overset{\uparrow}{z}/L)\rrbracket$ must be limited as explained in Sections 6.2 and 6.3.

6.7 Incongruity Between Transient Analysis Input Peaking Factors and TS Peaking Factors

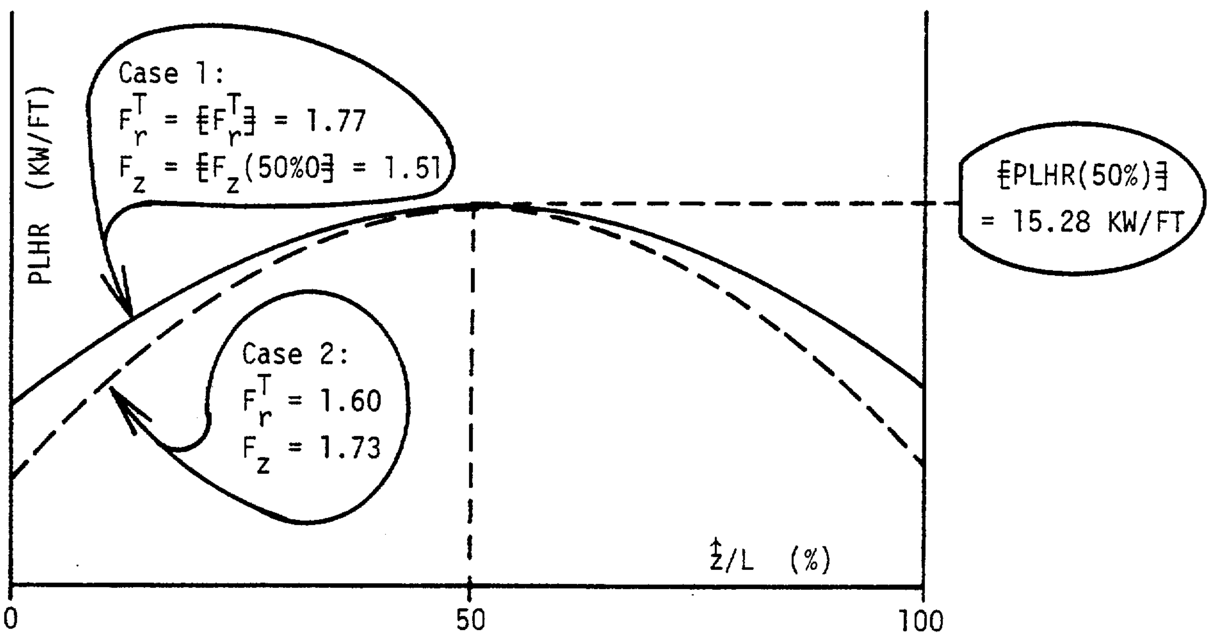
The computation of the transient analysis F_r^P requires as inputs F_r^A, the assemblywise pin power distribution for the hot assembly, and the axial power profile of the hot assembly. As explained in Section 6.5 the pertinent transient analysis parameters which must be controlled for the transient analysis to be valid are F_r^A, F_r^P, F_z and $\overset{\uparrow}{z}$. However, the quantities specified in the TS are $\llbracket F_r^A \rrbracket$, $\llbracket F_r^P \rrbracket$, and $\llbracket PLHR(\overset{\uparrow}{z}/L)\rrbracket$. It is easy to see that it is possible to have a power shape which conforms to the TS criteria, but has a larger F_z than the analysis that led to the TS criteria. One might ask "If the transient analyses were performed with this larger F_z, would not the predicted consequences be more adverse than those predicted by the transient analysis that led to the TS, thus making the TS criteria an inappropriate means for limiting the severity of the event?" The answer is "No," and the reasoning that leads to this conclusion is as follows:

First note that $\llbracket PLHR(\overset{\uparrow}{z}/L)\rrbracket$ is specified in the TS, but $\llbracket F_z(\overset{\uparrow}{z}/L)\rrbracket$ is not. Obviously it is only possible to have $F_z > \llbracket F_z(\overset{\uparrow}{z}/L)\rrbracket$ if $F_r^{P^z}$ is sufficiently less than $\llbracket F_r^P \rrbracket$ that the relationship $F_r^{P^z} * F_z \leq \llbracket F_r^P \rrbracket * \llbracket F_z(\overset{\uparrow}{z}/L)\rrbracket$ is maintained.

Consider two cases:

Case 1: Use reference transient analysis peaking factors for $\overset{\uparrow}{z}/L = 50\%$.

Case 2: Use $Fz > \{F_z(\overset{\uparrow}{z}/L = 50\%)\}$. In this case to make the arithmetic a little more transparent we will assume $PLHR = \{PLHR(\overset{\uparrow}{z}/L = 50\%)\}$ rather than $PLHR < \{PLHR(\overset{\uparrow}{z}/L = 50\%)\}$.



In the remainder of this section we will use the symbol $\overset{\downarrow}{z}$ to represent the height of MDNBR.

$\overset{\downarrow}{z}$ must lie above $\overset{\uparrow}{z}$, and usually lies only slightly above $\overset{\uparrow}{z}$. For practically any attainable axial power shapes (and with certainty for non-skewed power shapes) we have

$$\frac{\Delta H \text{ (Case 1, 0 to } \overset{\uparrow}{z})}{\Delta H \text{ (Case 2, 0 to } \overset{\uparrow}{z})} > 1$$

It then follows that

$$\text{Critical Heat Flux (Case 1, } \hat{z}) < \text{Critical Heat Flux (Case 2, } \hat{z})$$

Also

$$\text{Actual Heat Flux (Case 1, } \hat{z}) \cong \text{Actual Heat Flux (Case 2, } \hat{z})$$

Thus

$$\frac{\text{Critical Heat Flux (Case 1, } \hat{z})}{\text{Actual Heat Flux (Case 1, } \hat{z})} < \frac{\text{Critical Heat Flux (Case 2, } \hat{z})}{\text{Actual Heat Flux (Case 2, } \hat{z})}$$

$$\text{MDNBR (Case 1)} < \text{MDNBR (Case 2)}$$

Thus if a core has a power distribution which has F_r^P and F_z traded off against each other relative to the reference transient analysis values so that $F_r^P < [F_r^P]$, $F_z > [F_z]$, and $F_r^P * F_z \leq [F_r^P] * [F_z]$, then this core will enjoy a higher MDNBR than the reference analysis predicts.

6.8 Anomalous Values of F_r^A in the Cycle 5 Safety Report

There is an anomaly on page 38 of the Cycle 5 Safety Report which deserves clarification. Here there are three values of F_r^A: 1.43, 1.46, and 1.45.

This anomaly arose because of a misunderstanding of definitions.

Originally this calculation was done with all F_r^As equal to 1.45, which was the TS value.

The definition of F_r^A is F_r^A = (Hottest assembly power)/(Average assembly power)

For the T-H calculation, somehow Exxon misconstrued the definition of F_r^A to be

F_r^A = (Highest Assembly LHR)/(Core Average LHR), where

$$\text{(Assembly LHR)} = \frac{\text{(Total Assembly Power)}}{\text{(Number of feet of active fuel pins in assembly)}}$$

and

$$\text{(Core LHR)} = \frac{\text{(Total Core Power)}}{\text{(Number of feet of active fuel pins in core)}} .$$

The Cycle 5 TS have been written to compensate for this misunderstanding by making the TS F_r^A = 1.43, the lowest F_r^A which appears on page 38.

This misunderstanding has been clarified to all parties involved, and will not recur in future reloads. (Q&A 4 of Reference 3)

6.9 Reference LOCA Analysis

The LOCA analysis in the Axial Shape Report has been the reference LOCA analysis for Cycles 4 and 5. The value $\{PLHR(\hat{z}/L \leq 60\%)\}$ = 15.28 KW/FT and TS Figure 3.23-1 were both generated from analyses in the Axial Shape Report and any LOCA reanalysis must encompass a reevaluation of both of these items. Unlike the transient analysis case, for the LOCA analysis there is no simple TS adjustment (such as demanding MDNBR = 1.30 at 15% overpower) which will bring Cycle X in line with the reference LOCA analysis. But rather, all the Cycle X LOCA analysis input parameters must be less adverse than the reference LOCA analysis input parameters for the reference LOCA analysis to be bounding.

TS Figure 3.23-1 was derived assuming $\{F_r^A\}$ = 1.45 and $\{F_r^P\}$ = 1.77, and thus far in this discussion the validity of TS Figure 3.23-1 for other radial peaking factors has not been considered. Ideally TS Figure 3.23-1 would be valid for any set of radial peaking factors and, for that matter, for any LOCA input parameters different from those used in the reference LOCA analysis. If this is the case, then if a LOCA reanalysis becomes necessary, the only

LOCA computation required would be for $\{PLHR(\overset{\uparrow}{z}/L = 60\%)\}$ and all other $\{PLHR(\overset{\uparrow}{z}/L \neq 60\%)\}$ would be given by TS Figure 3.23-1.

To date Palisades has presented one example which shows the validity of TS Figure 3.23-1 for inputs other than those used in the reference LOCA analysis. In Reference 26 (herein called the Corner Pin Report) LOCA analyses were performed with $\{F_r^P\} = 1.90$, rather than the reference LOCA analysis value of $\{F_r^P\} = 1.77$. In the Corner Pin Report, $\{PLHR(\overset{\uparrow}{z}/L = 60\%)\}$ and $\{PLHR(\overset{\uparrow}{z}/L = 80\%)\}$ were computed, and the ratio between these two [PLHR] matched TS Figure 3.23-1 perfectly.

Rather surprisingly, not only did the ratio between the two [PLHR] in the Corner Pin Report match reference LOCA analysis ratio, but the individual [PLHR] in the Corner Pin Report were identical to the $\{PLHR\}$ in the reference LOCA analysis.

The "perfect matches" discussed in the last two paragraphs are perfect matches insofar as the [PLHR] are concerned. There are slight differences in the fuel designs, the analytical methods used, and the PCTs reached in the two cases. The "perfect matches" must be interpreted with the understanding that these differences exist.

The above example hardly constitutes proof that TS Figure 3.23-1 is valid for any set of LOCA analysis inputs, and any future LOCA reanalyses should be done with at least two $\overset{\uparrow}{z}$ values to verify that TS Figure 3.23-1 remains valid. However the excellent results obtained in this example give good reason to expect that other examples will verify that TS Figure 3.23-1 is valid for a wide range of LOCA input values.

6.10 Fuel Exposure Sensitivity

Fuel with high exposure develops high gas pressure within the fuel pins. In the LOCA analysis for high exposure fuel the local 17% oxydation limit restricts the (gas pressure, LHR) combination to values which prevent clad rupture in event of LOCA. The mechanism by which the high exposure fuel would suffer the

17% local clad oxydation is massive clad ballooning prior to rupture causing extensive steam flow blockage which would result in high temperatures and high oxydation rates above the region of flow blockage. An analysis of the allowed F_Q as a function of exposure is described in Appendix A of the Cycle 5 Safety Report. All other fuel in the core is mechanically similar to the batch H fuel, and hence the batch H fuel analysis applies to the batch G and batch I fuel as well.

The results of this analysis are incorporated into the LHR limit of TS 3.23.1 via the dependence on Figure 3.23-2. This TS applies to all fuel in the core, and will apply to the fuel in future reloads if it is mechanically similar to the batch H fuel.

During Cycle 5 batches H and I fuel will not receive enough exposure to be limited by TS Figure 3.23-2. The batch G fuel is sufficiently depleted that its LHR will most likely fall below the limit imposed by TS Figure 3.23-2, and probably no special measures will be required to limit the power in the batch G fuel assemblies.

6.11 LOCA Analysis Inputs

The validity of the reference LOCA analysis depends on only the following parameters, and any core which is bounded by these parameters is bounded by the reference LOCA analysis.

(1) The LOCA PLHR limit of TS 3.23.1, which includes the dependence on TS Figures 3.23-1 and 3.23-2. These figures have already been discussed.

(2) The limits in TS 3.23.2 of HFP $F_r^A \leq 1.45$ and HFP $F_r^T \leq 1.77$, with appropriately higher values permitted at less than 100 PCPOW.

(3) A maximum core power of 102% of 1530 MWT.

(4) RCS inlet temperature and pressure must be within the bounds allowed by TS 3.1.1.g.

(5) A minimum number of active SG tubes which is as follows:

SG#1 6112 active tubes SG#2 6757 active tubes

The required number of active tubes in the two SGs may be reversed.

(6) A fuel assembly geometry that does not lead to greater flow resistance or less heat transfer than assumed in the reference LOCA analysis.

6.12 Cycle 4 Batch H Fuel Corner Pin LOCA Analysis

Because of high radial peaking of the corner pins in the batch H fuel which are adjacent to wide water gaps, it was necessary to implement separate TS criteria for the power peaking in these pins in order to be able to reach 2530 MWT in Cycle 4. The analysis to support the corner pin peaking criteria is described in the Corner Pin Report, which was previously discussed in Section 6.9. Since the beginning of Cycle 4 these corner pins have burned down sufficiently that in Cycle 5 they can meet the power peaking criteria of the TS based on the reference LOCA analysis in the Axial Shape Report, and the analysis of the Corner Pin Report and the TS change required in Cycle 4 are not applicable to Cycle 5, and have been deleted.

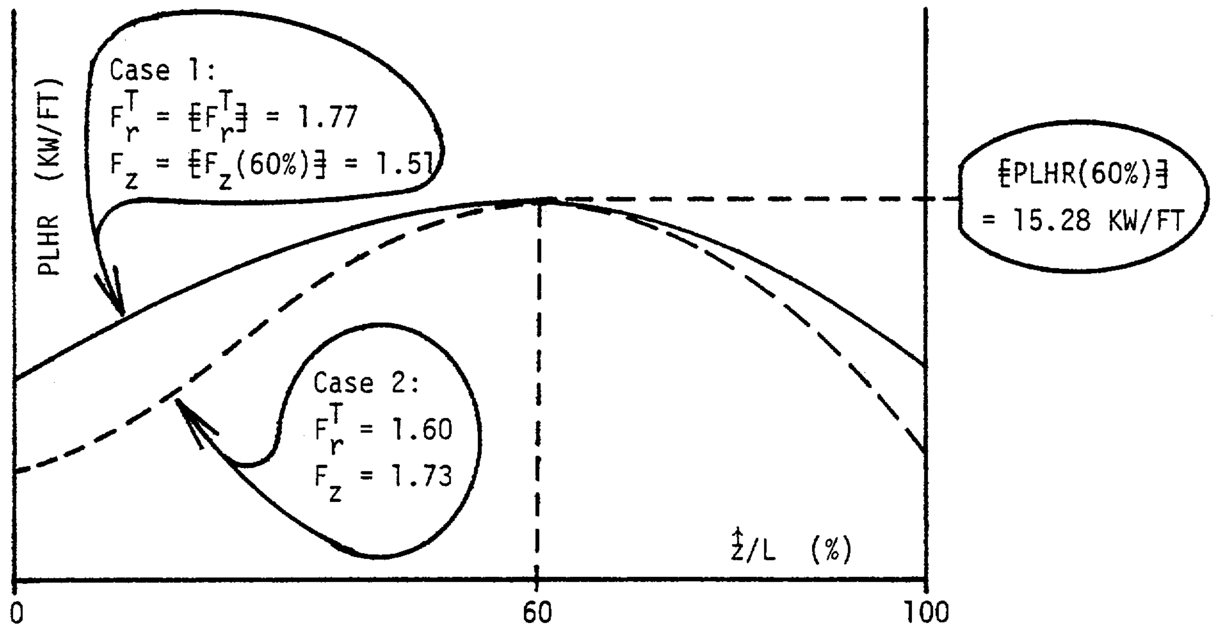
6.13 Incongruity Between LOCA Analysis Peaking Factors and TS Peaking Factors

This type of problem was discussed in Section 6.7 for transient analyses, and the line of reasoning for both transient analyses and LOCA analyses is identical up to the point where the figure is drawn in Section 6.7. We will therefore pick up the line of reasoning at that point.

Consider two cases:

Case 1: Use reference LOCA power peaking factors for $\hat{z}/L = 60\%$.

Case 2: Use $F_z > [F_z(60\%)]$. In this case to make the arithmetic a little more transparent we will assume PLHR = ⟦PLHR(60%)⟧ rather than PLHR < [PLHR(60%)].



In the following argument, bear in mind that in the LOCA analysis the blowdown computation is an assembly computation.

Case 1	Case 2
Local pin power distribution is highly peaked in the vicinity of the hot pin.	Local pin power distribution is fairly homogeneous throughout assembly.
A few pins near the peak pin are close to ⟦PLHR⟧ at $\hat{z}$.	Most pins in the assembly are close to ⟦PLHR⟧ at $\hat{z}$.
At start of blowdown a few pins in the assembly experience DNB near $\hat{z}$.	At start of blowdown most pins in the assembly experience DNB near $\hat{z}$.

The break size is chosen so that the flow is up and down away from $\hat{z}$. Thus the amount of time that the region near $\hat{z}$ is covered with water is small, and the above effect has little impact on the amount of heat removed in the region near $\hat{z}$.

Case 1

Well below $\overset{\uparrow}{z}$ a few pins are fairly hot and experience DNB. This reduces the rate of heat transfer away from these pins.

Case 2

Well below $\overset{\uparrow}{z}$ all pins are fairly cool. Relatively little DNB occurs and heat transfer is good.

Because of the above effect, after blowdown the Case 1 assembly has more stored energy than the Case 2 assembly. Insofar as the core power shape is separable, most or all of the assemblies in the Case 2 core will have a higher F_z than the Case 1 core, and the above argument holds for the whole core as well as the hot assembly.

Because the lower part of the Case 1 core has more stored heat after reflood than the Case 2 core, reflood occurs slower in the Case 1 core and the Case 1 core reaches a higher PCT. [PCT always occurs during reflood just before the point of PCT is quenched, and this point lies a little above $\overset{\uparrow}{z}$.]

Thus, if a core has a power distribution which has F_r^T and F_z traded off against each other relative to the reference LOCA analysis values so that $F_r^T < \llbracket F_r^T \rrbracket$, $F_z > \llbracket F_z \rrbracket$, and $F_r^T * F_z \leq \llbracket F_r^T \rrbracket * \llbracket F_z \rrbracket$, then this core will enjoy a lower PCT than the reference LOCA analysis predicts.

7.0 ANALYSIS OF DBEs OTHER THAN LOCA

From Sections 6.4 and 6.5 the first step in Cycle X DBE analysis is to pick the TS peaking factors $\llbracket F_r^A \rrbracket$, $\llbracket F_r^{\Delta H} \rrbracket$, $\llbracket F_r^N \rrbracket$, $\llbracket F_r^W \rrbracket$, and $\llbracket PLHR(\overset{\uparrow}{z}/L) \rrbracket$ so that at 115 PCPOW and design coolant conditions MDNBR = 1.30. This guarantees that at 100 PCPOW the same DNBR margin exists that existed for the reference DBE analyses at the beginning of the DBE. [It turned out that for Cycle 5 the LOCA $\llbracket F_r^T \rrbracket$ was more limiting (less than) $\llbracket F_r^W \rrbracket$, so that it was not necessary to have a separate TS for $\llbracket F_r^W \rrbracket$.]

Having chosen the peaking factors in this way, each Cycle X DBE will be bounded by the reference DBE analysis if all the Cycle X DBE inputs are bounded by the reference DBE analysis inputs. A list of the important inputs for DBE analyses, along with their reference analysis value and Cycle 5 value, are given in Table 7.0-1.

In Sections 7.1 thru 7.21-22 each DBE is examined to determine if the reference analysis is bounding for Cycle 5. Only one DBE, the Steam Line Break Event requires reanalysis. This reanalysis is reviewed in Section 7.17-18.

Exxon has developed a new Rod Ejection analysis methodology which is described in Reference 27. Even though the reference analysis inputs bounded the Cycle 5 inputs for this event, Exxon reanalyzed this event so that Palisades would be assured of having the best reference analysis for each DBE that Exxon is able to provide. (Q&A 4 of Reference 3)

In the examination to determine if the reference analysis inputs bound the Cycle 5 inputs several specific reasons why the reference analysis input is bounding occur for a number of DBEs. In Table 7.0-2 these reasons are listed as Remark 1 thru Remark 7. In the individual DBE discussion each remark that is applicable to that DBE is referenced.

Table 7.0-1 Important DBE Analysis Inputs

	DBE Number	DBE	
DBEs Analyzed in Transient Analysis Report	1	Uncontrolled Rod Withdrawal	BOC-HFP
	2	Uncontrolled Rod Withdrawal	BOC-HHP
	3	Uncontrolled Rod Withdrawal	EOC-HFP
	4	Uncontrolled Rod Withdrawal	EOC-HHP
	5	Control Rod Drop	BOC-HFP
	6	Control Rod Drop	EOC-HFP
	7	Four Pump Coastdown	BOC-HFP
	8	Locked Rotor	BOC-HFP
	9	Reduction in Feedwater Enthalpy	BOC-HFP
	10	Reduction in Feedwater Enthalpy	EOC-HFP
	11	Increased Feedwater Flow	EOC-HHP
	12	Excessive Load	EOC-HFP
	13	Excessive Load	EOC-HZP
	14	Loss of Load (DNBR Limited)	BOC-HFP
	15	Loss of Load (Pressure Limited)	BOC-HFP
	16	Loss of Feedwater	BOC-HFP
	17	Steam Line Break	EOC-HFP
	18	Steam Line Break	EOC-HZP
	19	Single Rod Withdrawal	BOC-HFP
	20	Single Rod Withdrawal	EOC-HFP
	21	Rod Ejection	BOC-HZP
	22	Rod Ejection	EOC-HFP
	23	Cycle 5 Values	

Table 7.0-1 Important DBE Analysis Inputs (Continued)

DBE Number	Doppler Temperature Coefficient (DTC) (pcm/degf)						Moderator Temperature Coefficient (MTC) (pcm/degf)	
	BOC			EOC				
	HFP	HHP	HZP	HFP	HHP	HZP		
	Nominal Transnt Analysis Report Value is -1.09	Use HFP Value in Transnt Analysis Report	Nominal Transnt Analysis Report Value is -1.50	Nominal Transnt Analysis Report Value is -1.38	Use HFP Value in Transnt Analysis Report	Nominal Transnt Analysis Report Value is -1.88	BOC	EOC
1	-0.87						5.0	
2		-0.87					5.0	
3				-1.66				-35.0
4					-1.66			-35.0
5	-0.87						5.0	
6				-1.66				-35.0
7	-0.87						5.0	
8	-0.87						5.0	
9	-0.87						5.0	
10				-1.66				-35.0
11					-1.66			-35.0
12				-1.66				-35.0
13						-2.26		-35.0
14	-0.87						5.0	
15	-0.87						5.0	
16	-0.87						5.0	
17				*				**
18						*		**
19	-0.87						5.0	
20				-1.66				-35.0
21			-1.20				5.0	
22						-2.26		-35.0
23	-1.29	< -1.29	-1.55	-1.49	< -1.49	-1.73	-4.5	-25.6

* Curve on page 123 of Transient Analysis Report is applicable to both reference analysis and cycle 5 analysis.

** Curve on page 122 of Transient Analysis Report is applicable to both reference analysis and cycle 5 analysis.

Table 7.0-1 Important DBE Analysis Inputs (Continued)

DBE Number	Shutdown Margin (%)						Rod Drop Time (sec)	
	In the Transient Analysis Report the Shutdown Margin varies from case to case depending on the values of MTC and DTC used for that case. In all HFP cases the Shutdown Margin is less than 2.0%, and in all HZP cases the Shutdown Margin is 2.0%.							
	BOC			EOC				
	HFP	HHP	HZP	HFP	HHP	HZP		
1	< 2.0						3.0	
2		< 2.0					3.0	
3				< 2.0			3.0	
4					< 2.0		3.0	
5	< 2.0						3.0	
6				< 2.0			3.0	
7	< 2.0						3.0	
8	< 2.0						3.0	
9	< 2.0						3.0	
10				< 2.0			3.0	
11					< 2.0		3.0	
12				< 2.0			3.0	
13						2.0	3.0	
14	< 2.0						3.0	
15	< 2.0						3.0	
16	< 2.0						3.0	
17				< 2.0			3.0	
18						2.0	3.0	
19	< 2.0						3.0	
20				< 2.0			3.0	
21			2.0				3.0	
22				< 2.0			3.0	
23	2.40	∿ 2.50	2.60	2.33	∿ 2.44	2.56	2.5	

Table 7.0-1 Important DBE Analysis Inputs (Continued)

DBE Number	Beta (%) [Delayed Neutron Fraction]		RCS Pressure (psia)	RCS Flow (%)	RCS Inlet Temp (degf)			
	BOC	EOC						
1	0.75		2010	100	542.5			
2	0.75		2010	100	542.5			
3		0.45	2010	100	542.5			
4		0.45	2010	100	542.5			
5	0.75		2010	100	542.5			
6		0.45	2010	100	542.5			
7	0.75		2010	100	542.5			
8	0.75		2010	100	542.5			
9	0.75		2010	100	542.5			
10		0.45	2010	100	542.5			
11		0.45	2010	100	542.5			
12		0.45	2010	100	542.5			
13		0.45	2010	100	542.5			
14	0.75		2010	100	542.5			
15	0.75		2110	100	542.5			
16	0.75		2010	100	542.5			
17		0.45	2010	100	542.5			
18		0.45	2010	100	542.5			
19	0.75		2010	100	542.5			
20		0.45	2010	100	542.5			
21	0.60		2010	100	542.5			
22		0.45	2010	100	542.5			
23	0.61	0.52	2060±50	101.6	537.5±5.0			

Table 7.0-1 Important DBE Analysis Inputs (Continued)

DBE Number	($\Delta\rho$, F_r^A) Values for Control Rod Drop Event							
	BOC - HFP				EOC - HFP			
	Maximum F_r^A		Minimum $\Delta\rho$		Maximum F_r^A		Minimum $\Delta\rho$	
	$\Delta\rho$ (%)	F_r^A	$\Delta\rho$ (%)	F_r^A	$\Delta\rho$ (%)	F_r^A	$\Delta\rho$ (%)	F_r^A
1								
2								
3								
4								
5	-0.12	1.66	-0.04	1.60				
6					-0.12	1.64	-0.04	1.60
7								
8								
9								
10								
11								
12								
13								
14								
15								
16								
17								
18								
19								
20								
21								
22								
23	-0.121	1.505						

Table 7.0-1 Important DBE Analysis Inputs (Continued)

DBE Number	F_r^A for Single Rod With-drawal	Steam Line Break Peaking Factors and MCHFR at Stuck Rod						
		EOC - HFP			EOC - HFP			
		F_r^A	F_Q	MCHFR	F_r^A	F_Q	MCHFR	
1								
2								
3								
4								
5								
6								
7								
8								
9								
10								
11								
12								
13								
14								
15								
16								
17		8.87	18.2	1.30				
18					8.09	16.0	1.41	
19	1.6							
20	1.6							
21								
22								
23	1.4	6.22	22.4	1.35	7.17	19.5	1.40	

Table 7.0-1 (Continued) Important DBE Analysis Inputs

As stated in the text, the Rod Ejection Event analysis performed in the Transient Analysis Report would be bounding for Cycle 5, and hence reanalysis would not be required for Cycle 5. However, Exxon has developed a new Rod Ejection method, and they wish to perform a new Rod Ejection analysis for all the PWRs they supply, which will become the reference analysis for future cycles.

The format of this table up to this point would be inconvenient for listing the Rod Ejection data, and hence we are changing the format for this last page of Table 7.0-1.

Important Inputs and Outputs for the Rod Ejection Event

	BOC - HFP		BOC - HZP		EOC - HFP		EOC - HZP	
	Ref Anal	Cyc 5	Ref Anal	Cyc 5	Ref Anal	Cyc 5	Ref Anal	Cyc 5
F_Q After Ejection	---	2.76	13.48	13.4	6.77	3.02	---	12.1
Ejected Rod Worth (%)	---	0.15	1.24	1.02	0.60	0.20	---	0.94
Doppler Coeficnt (pcm/degf)	---	-1.29	-1.20	-1.55	-1.10	-1.49	---	-1.73
Beta Fraction (%)	---	0.61	0.60	0.61	0.45	0.52	---	0.52
Energy Deposition (cal/gm)	---	164	247	143	200	173	---	126

Table 7.0-2 Remarks applicable to DBEs

(Just referred to as "Remarks" in DBE descriptions)

1	Since it is not clear whether maximum or minimum reactivity feedbacks make the results of this analysis more adverse, the reference analysis was performed at both BOC and EOC.
2	The rod drop time is relatively unimportant in this event because the transient is so slow that DNBR cannot decrease very much in the time it takes the rods to drop.
3	The shutdown worth in this event is relatively unimportant because the MDNBR occurs when the rods are just starting to enter the core.
4	The delayed neutron beta fraction in this event is relatively unimportant to the course of the pre-trip transient because the transient is slow and other reactivity effects play the dominant role in determining the course in the pre-trip part of the transient.
5	Due to the delayed neutron beta fraction, the power level during the trip suffers a prompt drop corresponding to a rapid negative reactivity insertion of (shutdown margin minus beta) and then slowly decays by an amount corresponding to a negative reactivity insertion equal to beta. The prompt drop is not enough to bring the power to zero.
6	Remark 4 plus: The MDNBR has occurred by the time the prompt drop is over, and the beta fraction plays no significant role in determining MDNBR.
7	The shutdown worth and rod drop time do not affect the severity of this event because MDNBR occurs before the reactor trip.

7.1-4 Uncontrolled Rod Withdrawal

See remark 1. Based on this we conclude that the reference analysis is bounding for Cycle 5.

Also applicable: Remarks 2, 3, 4, 6

7.5-6 Control Rod Drop

Normally in the analysis of the Control Rod Drop, the ($\Delta\rho$, F_r^A) pairs which would result from dropping each individual control rod into the core are evaluated. Then the transient is analyzed using each ($\Delta\rho$, F_r^A) pair that appears to be a likely candidate for producing the most adverse results from this event. The results of the transient are made worse by lowering $\Delta\rho$ or raising F_r^A. In the reference analysis only two ($\Delta\rho$, F_r^A) pairs were analyzed, the pair with the lowest $\Delta\rho$ and the pair with the highest Fr.

Since the reference analysis was performed, the effects of the following simplifying assumptions made in analytical model have been observed (Q&A 33 of Reference 3): The model assumes a constant F_r^A in the core throughout the course of the transient, which is the F_r^A that applies after the rod is dropped. Actually the core F_r^A goes from its initial steady state value to its maximum value in about the same time period that the reactor goes from its initial steady state power to the power corresponding to the negative reactivity insertion of the dropped rod.

After running a number of Rod Drop Events with this analytical model, it was observed that the effect of this simplifying assumption is that in the analysis the thermal conditions of power, pressure, temperature, and flow are the most adverse at the beginning of the transient before the rod has had a chance to drop. From this three important conclusions can be drawn:

(1) The analysis is always conservative.

(2) The value of $\Delta\rho$ has no influence on the computed value of MDNBR, and thus it is only necessary to examine the ($\Delta\rho$, F_r^A) combination with the largest value of F_r^A.

(3) The MTC, DTC, and Beta Fraction have no influence on the computed MDNBR.

For Cycle 5 at BOC the ($\Delta\rho$, F_r^A) pair with the largest F_r^A is (-0.121%, 1.505). For this pair the reference analysis, with a ($\Delta\rho$, F_r^A) pair of (-0.12%, 1.66) is bounding by a wide margin.

The Cycle 5 operating power distribution is flatter at EOC than at BOC. The F_r^A after the rod drop is fairly well approximated by (F_r^A before rod drop)* (azimuthal tilt caused by dropped rod). There is no reason to suppose the tilt caused by the dropped rod will be significantly higher at EOC than BOC. Therefore the dropped rod F_r^A is expected to be larger at BOC than at EOC. This was born out in the reference analysis. For this reason the BOC Rod Drop Event is expected to have more adverse consequences than the EOC Rod Drop Event, and the ($\Delta\rho$, F_r^A) pairs were computed only for BOC conditions.

Based on all the above considerations, we conclude that the reference analysis of the Rod Drop Event is bounding for Cycle 5.

<u>7.7-8</u> <u>Loss of Flow Events</u> (Four Pump Coastdown and Locked Rotor)

The severity of both these events are controlled by the same parameters, so we can discuss both of them together.

The primary parameters affecting the severity of the Loss of Flow Events is the time it takes the flow sensor to send out a trip signal and the rod drop time. The flow sensor response is identical in the reference analysis and in Cycle 5 and the rod drop time is 0.5 sec less in Cycle 5 than in the reference analysis. Based on this we conclude that the reference analysis of the Loss of Flow Events is bounding for Cycle 5.

The degradation of DNBR during the Loss of Flow Events is primarily due to the slowdown of the coolant flow and the heatup of the clad due to this slowdown. The reactor trip occurs before any change in power has a chance to significantly change the fuel temperature, and thus the MTC, DTC, and Beta Fraction have little effect on the severity of Loss of Flow Events.

Also applicable: Remark 3.

7.9-10 Reduction in Feedwater Enthalpy

See Remark 1. From this we conclude the reference analysis of this event is bounding for Cycle 5.

In the analysis the steady state DNBR is 1.75 and the MDNBR is 1.75, so this is not an event with much safety significance.

Also applicable: Remarks 4, 6, 7.

7.11 Increased Feedwater Flow

The primary parameter affecting the severity of this event is the negative MTC which causes the reactor power to increase during cooldown. Thus the event is analyzed only at EOC when the MTC is most negative. The reference analysis EOC MTC is more negative than the Cycle 5 EOC MTC, and we conclude the reference analysis of the Increased Feedwater Flow Event bounds Cycle 5.

The DTC is slightly less negative in Cycle 5 than in the reference analysis, so the reference analysis DTC does more to help retard the power increase than the Cycle 5 DTC. However the difference in the MTC reactivity insertion is more than twice the difference in the DTC reactivity insertion, so the MTC effect discussed in the first paragraph dominates the course of the transient.

In the analysis the steady state DNBR is 3.37 and the MDNBR is 3.00, so this is not an event of much safety significance.

7.12-13 Excessive Load

Like DBE 11, these are cooldown transients, and the reasons the reference analysis of these events bound Cycle 5 are the same as those iterated under DBE 11.

In the HFP case the steady state DNBR is 1.75 and the MDNBR is 1.74. For the HZP case the MDNBR is 3.60, so these DBEs do not have much safety significance.

7.14-15 Loss of Load

The primary factor affecting the severity of this transient is the MTC. In the reference analysis the BOC MTC is positive, which causes the power to increase as the core heats up from the Loss of Load. For Cycle 5 the BOC MTC is negative which causes the core power to decrease as the core heats up. Based on this we conclude that the reference analysis of the Loss of Load Events bounds Cycle 5.

Also applicable: Remarks 2, 3, 6.

7.16 Loss of Feedwater

Like DBEs 14 and 15 this is a heatup event and the reference analysis bounds Cycle 5 for the same reasons that this was the case for DBEs 14 and 15.

7.17-18 Steam Line Break

From Table 7.0-1 it can be seen that for both the HFP and HZP cases the stuck rod F_Q is higher for Cycle 5 than it is in the reference analysis. Because of this it is necessary to reanalyze the Steam Line Break Events. The description of this reanalysis is given in Reference 3.

The transient time behavior in the Steam Line Break Event depends only on point kinetics parameters, and the Cycle 5 values are bounded by the reference analysis values. Therefore it was not necessary to reevaluate the transient time behavior for the Cycle 5 reanalysis.

What does require reevaluation for Cycle 5 is the peaking factors about the stuck control rod and the T-H analysis at the point of MCHFR.

The methodology in the Transient Analysis Report was used with the following exceptions:

(1) A reduction in radial peaking was achieved by taking into account the fact that a portion of core power at the time of thermal margin limiting conditions is due to decay heat.

(2) In the reference analysis, a Modified Barnett Correlation (Reference 28) applied in a conservative subchannel basis was used to compute MCHFR. In the Cycle 5 analysis the same basic reference was used, but applied on an assembly cross sectional basis consistent with Reference 28, and consistent with the original work by Barnett (Reference 29). Palisades estimates that removing this conservatism results in a 25% to 30% increase in MCHFR.

(3) The MCHFR correlation has been modified for application to nonuniform axial heat flux profiles.

The results of the reference analyses and the Cycle 5 analyses are as follows:

Conditions at Point of MCHFR		EOL-HEP		EOL-HZP	
		Reference analysis	Cycle 5 analysis	Reference analysis	Cycle 5 analysis
	F_r^A	8.87	6.22	8.09	7.17
	F_Q	18.2	22.4	16.0	19.5
	MCHFR	1.30	1.35	1.41	1.40

We find the three changes in the analytical methodology to be reasonable and therefore acceptable. Since the Cycle 5 analyses both gave a MCHFR which would result in less than 1% fuel damage we find the results of these analyses acceptable. On this basis we approve the Cycle 5 Steam Line Break Event reanalysis.

7.19-20 Single Rod Withdrawal

See Remark 1. Also from Table 7.0-1 the power peaking at the location of the withdrawn rod is greater in the reference analysis than in Cycle 5. Based on this we conclude that the reference analysis of the Single Rod Withdrawal Event bounds Cycle 5.

Also applicable: Remarks 2, 3, 4, 6.

7.21-22 Rod Ejection

For the Rod Ejection Events the reference analysis inputs bound the Cycle 5 values. However Exxon has a new Rod Ejection analysis method (Reference 27) which they want to apply to all the plants they refuel, and thus they have reanalyzed the Rod Ejection Event for Palisades using the new method. The review of Reference 27 has progressed to the point where we will allow its use on interim basis until the final review is complete. The NRC criteria for consequences of the Rod Ejection Event are:

(1) The energy deposition in the fuel be $\leq$ 280 cal/gm.

(2) The peak system pressure be less than the design pressure ($\leq$ 2750 PSIA).

As can be seen in Table 3.0-1, for all cases studied the energy deposition is well within the 280 cal/gm limit.

In the reference analysis the BOC-HZP case produces the greatest pressure surge. For this case the transient is over in 4.92 seconds, and in this time the core generates 10,950 MW-sec which results in a pressure surge of 200 PSI.

Since the Cycle 5 analysis has a more negative doppler coefficient, both the peak power and the duration of the transient are less than they are in the reference analysis. Thus the Cycle 5 transient must produce a pressure surge of less than 200 psi. Since nominal system pressure is 2060 ± 50 psia the peak pressure reached is well below our criterion of 2750 PSIA.

On these bases we find the Cycle 5 Rod Ejection Analysis and its predicted consequences acceptable.

8.0 CYCLE 5 LOCA ANALYSIS

8.1 Cycle 5 LOCA Analysis Input Parameters Compared With Reference LOCA Analysis Input Parameters

The Cycle X LOCA analysis input parameters which must be bounded by the reference LOCA analysis input parameters for the reference LOCA analysis to be bounding for Cycle X were given in Section 6.11. Parameter 1, the ⟦PLHR⟧ specified in TS 3.23.1 has been changed to use the Cycle 3 value of 15.28 KW/FT value for the 208 pin assemblies and 14.71 KW/FT = (208/216)*15.28 KW/FT for the 216 pin assemblies. (In Cycle 3 all assemblies contained 208 pins.) There are a number of arguments that can be cited which show that calculating the ⟦PLHR⟧ for the 216 pin assemblied in this way is a conservative procedure.

Parameters 1, 2 (⟦F_r^T⟧ value), 3, 4. These parameters are automatically met because they are TS values which have not changed since the reference LOCA analysis was performed in Cycle 3.

Parameter 2: ⟦F_r^A⟧ value. As explained in Section 7.3.13.2, for Cycle 5 ⟦F_r^A⟧ = 1.43 whereas the reference LOCA analysis value was 1.45. This TS change makes the Cycle 5 core LOCA behavior conservative relative to the reference LOCA analysis.

Parameter 5: Number of active SG tubes. According to Q&A 41 of Reference 3 the number of active tubes in the reference LOCA analysis is 502 less than existed in 1977. Since 1977 only about 80 additional SG tubes have been plugged, so currently we have good margin with respect to the number of active SG tubes.

Parameter 6: Fuel geometry. As explained in Section 8.3.1 a slight change in the batch I fuel geometry should give it better LOCA performance than the fuel geometry assumed in the LOCA analysis.

Cycle 5 Neutronics. Parameters 1 and 2 deal with TS peaking factors. While the TS values of these peaking factors have not changed in Cycle 5, in Section 8.2 a number of reasons are given to show that the Cycle 5 peaking factors will fall below the TS values. If a LOCA were to occur during Cycle 5, these lower peaking factors would make the actual LOCA consequences less severe than predicted in the reference LOCA analysis. However, no credit can be taken for the lower computed peaking factors because the TS has not been changed to reflect the lower peaking factors. The criteria for asserting that the Cycle X peaking factors are bounded by the reference LOCA analysis peaking factors is that the Cycle 5 measured peaking factors, not the computed peaking factors, comply with the TS.

8.2 Features of Batch I Fuel Which In Cycle 5 Facilitates Meeting the Power Peaking TS Criteria Based on the Reference LOCA Analysis

8.2.1 Two Types of Batch I Fuel

60 of the batch I fuel assemblies are made from new fuel pins which are 2 mils larger in outside diameter than the previous Palisades fuel. The remaining 8 batch I fuel assemblies are constructed from spare fuel pins left over from batches E, G, and H.

8.2.2 Pinwise Power Flattening Within Batch I Assemblies

All of the batch I assemblies have 4 low enrichment pins on the corner locations to reduce the assemblywise pin power peaking observed in Cycle 4 in the batch H fuel.

The measures taken in Cycle 4 to compensate for this batch H fuel pin power peaking were described in Section 6.12.

The power shape analysis for Cycle 5 indicates that in Cycle 5 at 2530 MWT it will be possible to meet the power peaking TS criteria based on the reference LOCA analysis.

8.2.3 Pinwise Power Flattening Within Spare Rod Batch I Assemblies

The calculated maximum value of F_ℓ for the spare rod batch I assemblies is 1.205, which is well below the 1.22 required to meet TS 3.23.2 when $F_r^A = 1.45$.

8.2.4 Calculated LOCA Power Peaking Factors and TS LOCA Power Peaking Factors

The following peaking factors are presented on page 4 of the Cycle 5 Safety Report.

	Maximum Cycle 5 computed peaking factors	TS limits
F_Q	2.35	<2.76
F_r^A	1.34	<1.43
F_r^T	1.75	<1.77

The features of the batch I fuel described in this section and in Sections 8.2.2 and 8.2.3 do not guarantee that the Cycle 5 power shape is conservative relative to the criteria of the reference LOCA analysis, but rather these features improve the likelihood that during Cycle 5 operation the measured core power shape will meet or fall below the power peaking TS criteria based on the reference LOCA analysis.

8.3 Conservatism in Cycle 5 Over Reference LOCA Analysis

8.3.1 Larger Clad Diameter Pins in Batch I Fuel

As previously stated, the Batch I fuel made from new fuel pins has a 2 mil larger clad outside diameter than fuels used previously at Palisades. This should have a neglegible effect on the steam flow resistance, but because of the larger heat transfer area, it will result in more water heat transfer during blowdown and more steam cooling during reflood. [Admittedly, since the clad diameter is only increased by 2 mils, neither of these effects will be very great.] Thus the batch I fuel, which will be the limiting fuel with respect to LOCA in Cycle 5, would reach a lower PCT in event of LOCA than the reference analysis predicts. This makes the Cycle 5 core LOCA behavior conservative relative to the reference LOCA analysis.

8.3.2 Lower TS $[F_r^A]$ in Cycle 5

As explained in Section 6.8 due to a misunderstanding between different analysis groups at Exxon, analyses were done with three values of $[F_r^A]$: 1.43, 1.46, and 1.45. To compensate for this incongruity, the lowest value, i.e., $[F_r^A] = 1.43$ was used in the Cycle 5 TS. This TS change makes the Cycle 5 core LOCA behavior conservative relative to the reference LOCA analysis.

9.0 STARTUP PHYSICS TESTING PROGRAM FOR CYCLE 5

For Cycle 5 Palisades intends to use the Cycle 4 Startup Physics Testing Program, except that they will drop the Moderator Temperature Coefficient measurement at power and the Power Coefficient measurement at power (Q&A 1 of Reference 3). The reason they are dropping these two measurements is that they are rather inaccurate, and they can calculate these parameters more accurately than they can measure them.

They will still be performing the zero power Moderator Temperature Coefficient measurements which is an accurate measurement.

The startup physics test program as proposed by Palisades for Cycle 5 includes all the tests in our current position and we find the test program as proposed acceptable.

10.0 ENVIRONMENTAL CONSIDERATION

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

11.0 CONCLUSION

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

12.0 REFERENCES

1. Letter, D. P. Hoffman (CPC) to Dennis M. Crutchfield (NRC), July 21, 1981. Subject: Palisades Cycle 5 TS Changes.

2. Letter, D. P. Hoffman (CPC) to Dennis M. Crutchfield (NRC), August 6, 1981. Subject: Correction to TS changes in Reference 1.

3. Letter, B. D. Johnson (CPC) to Dennis M. Crutchfield (NRC), November 17, 1981. Subject: Formal Questions and Answers plus new Steam Line Break Analysis.

4. B. D. Johnson (CPC) to Dennis M. Crutchfield (NRC), December 2, 1981. Subject: Revision to Reference 3 giving method for computing LHR.

5. Letter, D. P. Hoffman (CPC) to Dennis M. Crutchfield (NRC), November 20, 1981. Subject: Revised TS 3.23.1 which specifies allowable LHR.

6. Verbal Commitment: B. D. Johnson (CPC) to Thomas Wambach (NRC), November 18, 1981. Subject: New title for TS Figure 3.23-3.

7. XN-NF-77-57, "Exxon Nuclear Power Distribution Control for Pressurized Water Reactors, Phase 2," January 1978.

8. XN-NF-80-47, "Palisades Power Distribution Control Procedures," October 1980.

9. XN-NF-81-34, "Palisades Cycle 5 Reload Fuel Safety Analysis Report," May 1981. [Herein called the Cycle 5 Safety Report]

10. XN-NF-78-18, "Analysis of Axial Power Distribution Limits for the Palisades Nuclear Reactor at 2530 MWT," June 1, 1978. [Herein called the Axial Shape Report]

11. XN-72-23, "Cladding Collapse Calculational Procedure," ENC, November 1, 1972.

12. XN-73-25, "GAPEXX: A Computer Program for Predicting Pellet-to-Clad Heat Transfer Coefficients," ENC, August 1973.

13. NUREG-0418, "Fission Gas Release from Fuel at High Burnup," R. O. Meyer, C. E. Beyer, and J. C. Vogelwede, USNRC, March 1978.

14. Letter, B. D. Webb (CPC) to Dr. Lawrence Eisenhart (BNL), October 6, 1981. Subject: Comparison Between CASMO and ENC's PDQ7 Computations with Palisades Batch I fuel.

15. XN-NF-79-56(P), "Gadolinia Fuel Properties for LWR Fuel Safety Evaluation," ENC, June 9, 1980.

16. Letter, L. S. Rubenstein (NRC) to Robert L. Tedesco (NRC), October 2, 1981. Subject: Safety Evaluation of Exxon Topical Report on Gadolinium Fuel Properties.

17. XN-NF-77-18, "Plant Transient Analysis of the Palisades Reactor for Operation at 2530 MWT," ENC, July 1977. [Herein called the Transient Analysis Report]

18. XN-NF-77-24, "LOCA Analysis for Palisades at 2530 MWT Using the ENC WREM-II PWR ECCS Evaluation Model," ENC, July 1977. [Herein called the 2530 LOCA Report]

19. Letter: A. Schwencer (NRC) to David Bixel (CPC), November 1977. Subject: Amendment No. 31 to Palisades Operating License No. DPR-20.

20. CENPD-137, "Calculative Methods for the CE Small Break LOCA Evaluation Model," August 1974, Supplement 1P, 1977.

21. XN-NF-77-22, "Steady State Thermal Hydraulic and Neutronics of the Palisades Reactor at 2530 MWT," ENC, July 15, 1977.

22. Memo: D. F. Ross and D. G. Eisenhut (NRC) to D. B. Vassallo and K. R. Goller, February 16, 1977. Subject: Revised Interim Safety Evaluation Report on the Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors.

23. Letter: C. Eicheldinger (W) to V. Stello (NRC), August 17, 1976, NS-CE-1170.

24. E. S. Markowski, et al., "Effect of Rod Bowing on CHF in PWR Fuel Assemblies," ASME Paper 77-HT-91.

25. Memo: R. L. Tedesco (NRC) to D. B. Vassallo, March 28, 1979. Subject: Evaluation of Westinghouse Report, "Effect on CHF of a Partially Bowed Heated Rod in a Cold Wall Thimble Cell Geometry."

26. XN-NF-80-18, "ECCS and Thermal Hydraulic Analyses for the Palisades Reload H Design," ENC, April 1980. [Herein called the Corner Pin Report]

27. XN-NF-78-44, "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," R. J. Burnside et al., ENC, February 1978.

28. E. Daniel Hughes, "A Correlation of Rod Bundle Critical Heat Flux for Water in the Pressure Range 150 to 725 psia," IN-1412, July 1970.

29. P. G. Barnett, "A Correlation of Burnout Date for Uniformly Heated Annuli and its use for Predicting Burnout in Uniformly Heated Rod Bundles," AEEW-R463, 1966.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-255

CONSUMERS POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO PROVISIONAL OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 68 to Provisional Operating License No. DPR-20, issued to Consumers Power Company (the licensee), which revised the Technical Specifications for operation of the Palisades Plant (the facility) located in Covert Township, Van Buren County, Michigan. The amendment is effective as of its date of issuance.

The amendment approves changes to the Appendix A Technical Specifications which specify new limits for radial peaking factors and allowable linear heat rates as well as identify excore detectors for use in core power distribution monitoring.

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

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

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For further details with respect to this action, see (1) the application for amendment dated July 21, 1981 and supplements thereto dated August 6, 1981, October 22, 1981, November 9, 17, 20, 1981 and December 2, 1981, (2) Amendment No. 68 to License No. DPR-20, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room 1717 H Street, N. W., Washington, D. C. 20555, and at the Kalamazoo Public Library, 315 South Rose Street, Kalamazoo, Michigan 49006. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C., 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 8th day of December, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas V. Wambach

Thomas V. Wambach, Acting Chief
Operating Reactors Branch #5
Division of Licensing