CONSUMERS POWER COMPANY
Docket 50-255
Request for Change to the Technical Specifications
License DPR-20

For the reasons hereinafter set forth, it is requested that the Technical Specifications contained in the Provisional Operating License DPR-20, Docket 50-255, issued to Consumers Power Company on October 16, 1972, for the Palisades Plant be changed as described in Section I below:

I. Changes and Discussion of Changes

1. In Section 1.1, delete the definition of "Total Radial Peaking Factor - F_r^T".

   The peak rod F_r^T limits are removed from the Technical Specifications since the LOCA analysis results bound each fuel rod type and the minimum DNBR limiting fuel rod type is an interior fuel rod (see proposed changes 19 and 23 below).

2. In Section 1.1, delete the definition of "Narrow Water Gap Fuel Rod".

   The narrow water gap rod F_r^N limits are removed from the Technical Specifications since the LOCA analysis results bound each fuel rod type and the minimum DNBR limiting fuel rod type is an interior fuel rod (see proposed changes 19 and 23 below).

3. In Section 1.1 page 1-2a delete the definition of "Narrow Water Gap Fuel Rod Peaking Factor - F_r^N".

   The narrow water gap rod F_r^N limits are removed from the Technical Specifications since the LOCA analysis results bound each fuel rod type and the minimum DNBR limiting fuel rod type is an interior fuel rod (see proposed changes 19 and 23 below).

4. In Technical Specification 3.1.1(a) change "1500 gpm" to "2810 gpm". This modifies a proposed change in our March 25, 1988 submittal.

   The shutdown cooling flow rate of 2810 gpm is consistent with the results calculated for the boron dilution event documented in ANF-88-108.

5. In the "Basis" for Technical Specification 3.1.1, from our March 25, 1988 submittal, change the sentence that reads:

   "By imposing a minimum shutdown cooling pump flow rate of 1500 gpm, sufficient time is provided..."

   to:

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"By imposing a minimum shutdown cooling flow rate of 2810 gpm, sufficient time is provided ..."

These shutdown cooling flow rates and shutdown margin requirements are consistent with the result calculated for the boron dilution event documented in ANF-88-108.

6. In the "reference" section of Technical Specification 3.1.1, change reference 6, "ANF-87-150(P), Volume 2, Section 15.4.6.3.2" to "ANF-88-108". This modifies a proposed change in our March 25, 1988 submittal.

   The new reference for the boron dilution analysis of record is ANF-88-108.

7. Change Technical Specification 3.10.1c., from our March 25, 1988 submittal, to read:

   "c. At less than the hot shutdown condition, with at least one primary coolant pump in operation or at least one shutdown cooling pump in operation with a flowrate ≥ 2810 gpm, the boron concentration shall be greater than the cold shutdown boron concentration for normal cooldowns and heatups, i.e., non-emergency conditions.

   During non-emergency conditions, at less than the hot shutdown condition with no operating primary coolant pumps and a primary system recirculating flow rate < 2810 gpm but ≥ 650 gpm, within one hour either:

   1. (a) Established a shutdown margin of ≥ 3.5% and
      (b) Assure two of the three charging pumps are electrically disabled.

   OR

   2. At least every 15 minutes verify that no charging pumps are operating. If one or more charging pumps are determined to be operating in any 15 minute surveillance period, terminate charging pump operation and insure that the shutdown margin requirements are met and maintained.

   During non-emergency conditions, at less than the hot shutdown condition with no operating primary coolant pumps and a primary system recirculating flow rate less than 650 gpm, within 1 hour:

   (a) Initiate surveillance at least every 15 minutes to verify that no charging pumps are operating. If one or more charging pumps are determined to be operating in any 15 minute surveillance period, terminate charging pump operation and insure that the shutdown margin requirements are met and maintained."

The shutdown margin requirement of 3.5% for hot and cold shutdown conditions and/or the restrictions on recirculating flow rate and charging pump operations are consistent with the results of the boron dilution transient, reported in ANF-88-108.

8. In the "Basis" for Technical Specification 3.10, from our March 25, 1988 submittal, insert the following sentence after the sentence that begins: "This requirement applies to normal operating situations and not during emergency conditions..."

   "By imposing a minimum shutdown cooling pump flowrate of 2810 gpm, sufficient time is provided for the operator to terminate a boron dilution under assymetric conditions. For operation with no primary coolant pumps operating and a recirculating flowrate less than 2810 gpm the increased shutdown margin and controls on charging pump operability or alternately the surveillance of the charging pumps will ensure that the acceptance criteria for an inadvertent boron dilution event will not be violated.[(3)]"

   Refer to discussion for change 7 above.

9. In the "Reference" section of Technical Specification 3.10, from our March 25, 1988 submittal, add the following:

   "(3) ANF-88-108"

   ANF-88-108 documents the boron dilution transient analysis results for Palisades Cycle 8.

10. Revise the equation for allowable power level (APL) under the "Basis" for Technical Specification 3.11-2, Excore Power Distribution Monitoring System, to read:

$$APL = \left[\frac{LHR(Z)_{TS}}{LHR(Z)_{Max} \times V(Z) \times 1.02} \right]_{Min} \times \text{Rated Power} \qquad (2)$$

   The changes made to the APL equation are consistent with the removal of Figure 3.23-2 (see proposed Change 16 below).

11. Remove Item (4) under the "Basis" for Technical Specification 3.11.2. Item (4) begins: "$E_p(Z)$ is a factor to account..."

   The removal of Item 4 under "Basis" of Technical Specification 3.11.2 is consistent with the removal of Figure 3.23.2 (see proposed Change 16 below).

12. Change Item "(5)" to "(4)" under the "Basis" for Technical Specification 3.11.2.

   Since Item "(4)" in the current "Basis" for Technical Specification 3.11.2 is recommended to be removed, Item "(5)" becomes Item "(4)" in the proposed Technical Specifications.

13. Change Item "(6)" to "(5)" under the "Basis" for 3.11.2 and revise to the following:

"(5) The quantity in brackets is the minimum value for the entire core at any elevation (excluding the top and bottom 10% of the core) considering limits for peak rods. If the quantity in brackets is greater than one, the APL shall be the rated power level."

14. Add "(2) "ANF-88-107" under the "References" for Technical Specification 3.11.2.

The reference for the LOCA analysis supporting Cycle 8 operation is ANF-88-107.

15. Remove Technical Specification Figure 3.23-1 and replace with the attached Figure 3.23-1.

The LOCA analysis for Palisades Cycle 8 (Reference 2) justifies the use of the linear heat rate (LHR) function given in the attached Figure 3.23-1.

16. Remove Technical Specification Figure 3.23-2.

The LOCA analysis performed for Cycle 8 was performed at a peak assembly discharge burnup of 52.5 GWD/MT. This bounds all assembly exposures less than this value. Therefore, the allowable LHR as a function of burnup is not required for exposures up to 52.5 GWD/MT.

17. Remove Technical Specification Figure 3.23-3.

The LOCA analysis in Reference 2 was performed for a maximum pellet LHR of 15.28 KW/Ft and, therefore, bounds peak pellet LHR limits in the current Technical Specifications (Technical Specification Table 3.23-1) for both narrow gap and interior fuel rods. Therefore, Figure 3.23-3 which provides the LHR function for interior and narrow gap fuel rods can be removed.

18. Replace Table 3.23-1 in the current Technical Specifications with the attached Table 3.23-1.

The Basis for the change is the same as proposed Change 17 above.

19. Replace Table 3.23-2 in the current Technical Specifications with the attached Table 3.23-2.

In support of a low radial leakage core, the radial peaking factor limit supported for Cycle 8 was increased by 3.5% for both 208 and 216 rod assemblies. The peak rod and narrow gap rod F_r limits are removed from Table 3.23-2 since the LOCA analysis results bound each fuel rod type and the minimum DNBR limiting fuel rod type is an interior fuel rod.

20. Change Technical Specification 3.23.1, Linear Heat Rate (LHR) to

read:

"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 $F_A(Z)$ [the function $F_A(Z)$ is shown in Figure 3.23-1]."

The changes to Technical Specification 3.23.1 reflect the removal of Figures 3.23-2 and 3.23-3 and the revision of Table 3.23-1 to remove reference to narrow gap and interior fuel rod LHR limits (see proposed Changes 16, 17 and 18 above).

21. From the "Basis" for Technical Specification 3.23.1, from our March 25, 1988 submittal, remove the "(3)" from the second line of the first paragraph.

    Reference 3 is no longer applicable.

22. Remove References 1, 2, and 3, from the "References" section for Technical Specification 3.23.1, from our March 25, 1988 submittal, and add new reference (1).

    "(1) ANF-88-107"

    The reference for the LOCA analysis supporting Cycle 8 operation is ANF-88-107, which supercedes the references currently listed for Technical Specification 3.23.1.

23. Remove references to F_r^T and F_r^N from Technical Specification 3.23.2, Radial Peaking Factors, from our March 25, 1988 submittal, (three places).

    The radial peaking factor limits for the peak rod F_r^T and the narrow gap rod F_r^N are being removed from Table 3.23-2 (see proposed change 19). Based on monitoring experience, the measured $F_r^{\Delta H}$ for interior fuel rods consistently exhibit the least amount of margin to Technical Specification limits. Therefore, the Technical Specifications limits on radial peaking for narrow gap and peak rod are being removed.

24. Delete reference to "F_r^T " and "F_r^N " from Technical Specification 4.19.2.

    The radial peaking factor limits for the peak rod, F_r^T, and the narrow gap rod, F_r^N, are being removed from Table 3.23-2 (see proposed Change 19).

Analysis of No Significant Hazards Consideration

The implementation of a low radial leakage core in Cycle 8, for the purpose of reducing Reactor Vessel fluence, requires that allowed radial peaking factors for the core be increased by 3.5% to maintain full power capability. The increased peaking factors can be accommodated because an improved Reactor Protection System (RPS) is being installed for Cycle 8. The improved Thermal Margin/Low Pressure (TM/LP) trip with axial monitoring is the primary RPS improvement, which allows additional operating margin without reducing the margin of safety. The Technical Specifications Change Request for the Reactor Protection System (previously submitted) provides additional basis for this proposed change.

The Technical Specifications changes being made include: 1) an increase of 3.5% in the allowed radial peaking factor, 2) the elimination of separate peaking factor limits for "narrow gap" and "peak" rods, 3) elimination of separate LOCA kW/ft limits for "narrow gap" and "peak" rods, 4) elimination of a burnup penalty on LOCA kW/ft limits, 5) modification of the required axial correction of local kW/ft limits, and 6) increase in the local kW/ft limits for the peak rod of a 216 rod fuel assembly.

1. The proposed change will not increase the probability of an accident previously evaluated in the FSAR. The changes to the Technical Specification proposed by this request involve no changes to hardware. The hardware changes for the RPS modification are assumed in the analysis supporting these specification changes. Changes to Linear Heat Rate (LHR) limits result in changes to alarm setpoints for incore detectors in the Primary Data Logger. However, the bases for the alarms remain unchanged. Technical Specification monitoring requirements for LHR limits ensure that the NRC acceptance criteria (10CFR50.46(b)) for Loss-of-Coolant accidents will be met. Monitoring of radial peaking factors ensure that the assumptions used in the analysis for establishing DNB margin, LHR and the thermal margin/low pressure and variable high power trip setpoints remain valid during operation. The changes proposed to the Technical Specifications maintain conservative surveillance requirements and are based upon Advanced Nuclear Fuels Corporation (ANF) methodology approved by the NRC.

Additional requirements relative to shutdown cooling flow rates are established by this Technical Specifications change. These changes were required as a result of updated analysis methodology to treat the conservative use of non-uniform, asymmetric flow mixing related to dilution events and were not the result of any physical changes being made by this change request or the RPS Modification.

All acceptance criteria as defined in ANF 87-150(NP), Volume 2 are met. The acceptance criteria for Operating Events through Infrequent Events require the pressure in the Primary Coolant loop and main steam system remains less than 110% of design value (2750 psia). The Technical Specifications changes resulted in no change in system response to transients required to be analyzed.

2. The proposed change will not significantly increase the consequences of accidents previously analyzed in the FSAR. This is because the higher peaking factors and modified kW/ft limits will allow some fuel rods to operate at higher local powers and thus higher temperatures. All Standard Review Plan Chapter 15 accidents and transients, however, have been reviewed or reanalyzed to determine that acceptance criteria are still met.

   The Standard Review Plan states clearly all those Chapter 15 events which must be analyzed to determine that minimum acceptance criteria are always met. Advanced Nuclear Fuels (ANF) has performed a review or reanalysis of all Chapter 15 events with the assumptions of the modified peaking factors for Cycle 8. This review and analysis is reported in ANF report ANF-88-108 "Palisades Cycle 8: Disposition and Analysis of Standard Review Plan Chapter 15 Events." In this report each event described in the Standard Review Plan was reviewed and dispositioned into one of the following categories: (Refer to Table 2-1 of ANF 88-108)

   a. The event initiator or controlling parameters have been changed from the analysis of record so that the event needs to be reanalyzed for the current licensing action.

   b. The event is bounded by another event which is to be analyzed.

c. The event causes and principle variables which control the results of the event are unchanged from or bounded by the analysis of record; or

d. The event is not in the licensing basis for the plant.

ANF report ANF 88-108 assumes that ANF 87-150(NP), Volumes 1 and 2, are part of the licensing basis for Palisades. These reports were submitted to the NRC as part of the RPS Modification.

The ANF report ANF-88-108 summarizes the disposition of events and provides the results for normal operations, anticipated operational occurrences and postulated accidents reanalyzed for this submittal. The results show that acceptance criteria are met for each event. Table 1 compares the results of ANF 88-108 to ANF 87-150(NP) Volume 2.

The RPS Modification and associated Technical Specifications Change Request were supported by the analyses of ANF 87-150(NP) Volume 1 and 2.

TABLE 1

		ANF 88-108		ANF 87-150(NP) Vol 1	
EVENT		MDNBR	MLHR	MDNBR	MLHR
15.1.3	Increase in Steam Flow	1.46	14.9	1.497	13.2
15.2.1	Loss of External Load	1.71	13.5	1.776	12.7
15.3.1	Loss of Forced Reactor Coolant Flow	1.40	13.1	1.455	12.7
15.3.3	Reactor Coolant Pump Rotor Seizure	1.28	13.1	1.34	(3)
15.4.1	Uncontrolled Control Bank Withdrawal at Subcritical or Low Power	1.01 (1)	(3)	1.036 (1)	14.8
15.4.2	Uncontrolled Control Bank Withdrawal at Power	1.25	14.8	1.304	15.3
15.4.3	Control Rod Misoperation				
	(a) Dropped Rod or Bank	1.25	15.6	1.301	17.4
	(b) Single Rod Withdrawal	1.22	15.1	1.273	17.4
	(c) Core Barrel Failure	1.25	(3)	(4)	(4)
15.4.6	CVCS Malfunction Resulting in Decreased Boron Concen	(SDM adequacy demonstrated)			
15.4.8	Control Rod Ejection	<1.17 (2)	-	<1.17 (2)	-

NOTES:

(1) <2.9% of the Core experiencing DNB vs <2.3% for ANF 87-50(NP), Volume 2
(2) <12.2% of the core experiencing DNB vs 12.2% for ANF 87-150(NP) Volume 2
(3) In some cases actual MLHR values have not been reported but have been verified to be well below the fuel centerline melt criteria of 21 kW/ft and to be not applicable as SAFDL because event is classified as Infrequent Event.
(4) Core barrel failure was bounded by FSAR analysis, thus not analyzed for RPS Mod. Higher peaking factors for Cycle 8 required reanalysis.

For event 15.4.1, Uncontrolled Control Bank withdrawal at subcritical or low power, the minimum DNBR is 1.01, which is below the 1.04 calculated in ANF 87-150(NP), Volume 2 and the 1.17 DNB safety limit for the XNB critical heat flux correlation. The percent of the core experiencing boiling transition was calculated to be less than 2.9% for Cycle 8 as compared to less than 2.3% for the ANF 87-150NP, Volume 2 analysis. Due to conservative assumptions in the fuel failure calculation, the offsite radiological doses remain less than 10% of the 10CFR100 limits. Therefore, applicable acceptance criteria are satisfied.

For event 15.4.8, Control Rod Ejection, the fuel failure evaluation was reanalyzed using Cycle 8 specific post-ejection radial peaking factors. In ANF 87-150(NP), Volume 2, it was determined that 12.2% of the fuel rods in the core will fail due to the penetration of DNB. ANF determined that the radiological consequences of the calculated 12.2% fuel failures are below 10CFR100 does limits and the whole body does is less than 25% of the respective 10CFR100 limit. Due to the conservative assumptions employed in the fuel failure analysis, the amount of fuel that is predicted to fail for Cycle 8 is less than 12.2% and the radiological conclusions reached in the previous report and safety evaluation remain valid. Applicable acceptance criteria are considered to be met.

Conservative assumptions for these fuel failure analyses included a lower DNB value than reported and the assumption that for assemblies with any rods failed the entire assembly was assumed failed.

The LOCA analysis for Palisades Cycle 8 operation is reported in ANF report ANF-88-107. Numerous changes have occurred in the ANF LOCA methodology since the previous licensing calculations were performed for the Palisades Plant. The methodology improvements provided additional margin which was more than sufficient to cover the increased peaking factors and the simplification of kW/ft limits desired for Cycle 8 operation. The new methodology, however, also made it necessary to perform a mini break-spectrum analysis to verify the limiting break size. The results of the analysis verified the 0.6 Double Ended Cold Leg Break (DECLG) as the limiting break size. The analysis demonstrates that the 10CFR50.46(b) criteria are satisfied for the Palisades plant with the new axially dependent power peaking limit curve. The analysis supports a maximum LHR of 15.28 kW/ft up to a relative core height of 0.6 and a LHR of 14.75 kW/ft at a relative core height of 0.8. The analysis supports a total radial peaking factor of 1.92 and a maximum average steam generator tube plugging level of 29.3% with up to 4.5% asymmetry. The peak cladding temperature was calculated to be 1914°F for the BOC profile and 2114°F for the EOC profile both of which are lower than the FSAR analysis results. The analysis supports Cycle 8 operation and is intended to support operation for future cycles with potential higher peaking factors, up to 1.92. The LOCA analysis results and methodology used are supported to peak assembly burnup of 52.5 GWD/MTU versus the old analysis limit of 43.6 GWD/MTU (peak rod).

The containment pressure analysis of record is not impacted by the new LOCA analysis. The 0.6 double ended cold leg break, as used in the containment analysis, remains the limiting break. The new LOCA analysis calculates lower containment pressures than the previous LOCA analysis. Both analysis had input parameters set to provide conservatively low containment pressures as this maximizes the LOCA severity.

3. The possibility of an accident of a new or different type than any previously evaluated in the FSAR will not be created since this change does not involve hardware and is supported by analysis methodology previously approved by the NRC.

4. The proposed change will result in a reduction to the margin of safety as defined in the basis of the Technical Specifications, however, the reduction is not significant. In Table 1, for many anticipated operational occurrences, the DNBR ratio is closer to the limiting value than was previously the case. However, the safety analysis reports described in 2

above show that there remains adequate margin to the acceptance criteria for normal operation, all anticipated operational occurrences and postulated accidents.

The requirements of 10CFR50, Appendix A, Criteria 10, 20, 25, and 29 require that the design and operation of the plant assure that the Specified Acceptable Fuel Design Limits (SAFDLs) not be exceeded during Anticipated Operational Occurrences (AOOs). The SAFDLs applicable to the Palisades Plant are: 1) the fuel shall not experience centerline melt (21 kW/ft), and 2) the minimum XNB DNB ratio shall be greater than or equal to 1.17. Only accident categories of Operational Events and Moderate Frequency Events are included in the definition of AOOs. Results are summarized in Table 1.

SRP events listed in Table 1 which are not AOOs are 15.3.3, 15.4.1, 15.4.3.c and 15.4.8. Significant margin exists between the SAFDLs and the transient analysis results for the AOO events.

While the percent of fuel failure is predicted to be greater than that calculated in ANF 87-150(NP) Volume 2, for the Uncontrolled Control Bank withdrawal at Subcritical or lower power (classified as an Infrequent Accident category) the offsite radiological doses remain less than 10% of the 10CFR100 limits. For the Control Rod Ejection, a Limiting Fault event analysis results were shown to be bounded by the results reported in ANF 87-150(NP) Volume 2.

II. Conclusion

The Palisades Plant Review Committee and the Nuclear Safety Board have reviewed this Technical Specification Change Request and have determined the changes resulting from the Cycle 8 specific changes to involve an unreviewed safety question. However, it is concluded that the proposed changes do not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or

2. Create the possibility of a new or different type of accident than those previously evaluated; or

3. Involve a significant reduction in the margin of safety.

The proposed changes therefore do not involve a significant hazards consideration. A copy of this Technical Specification Change Request has been sent to the State of Michigan Official designated to receive such applications for amendment to the Operating License.

CONSUMERS POWER COMPANY

By David P. Hoffman
David P Hoffman, Vice President
Nuclear Operations

Sworn and subscribed to before me this 1st day of September 1988.

Elaine E Buehrer
Elaine E Buehrer, Notary Public
Jackson County, Michigan
My commission expires October 31, 1989