Consumers Power
POWERING MICHIGAN'S PROGRESS

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DOCKET 50-255 - LICENSE DPR-20 - PALISADES PLANT-
TECHNICAL SPECIFICATIONS CHANGE REQUEST -
REACTOR PROTECTION SYSTEM

On December 23, 1987, Consumers Power Company submitted a draft of this Technical Specifications Change Request to initiate NRC review. A subsequent meeting was held with the NRC staff on February 17, 1988. In that meeting, we informed the NRC of our schedule to submit this change request and asked that NRC review, to support the modifications of the Reactor Protection System, be scheduled for a July approval. Our request is based on the potential lead time for any further information that may be required to support this change request, and the interface we have with modifications for our proposed Inadequate Core Cooling Instrumentation project. To allow for the approval of this change request to precede the actual modifications, we request the amendment become effective at the start of Cycle 8.

In addition to the Technical Specifications changes relating to the Reactor Protection System modifications, changes are being proposed relating to Generic Letter 86-13 and removal of the steam generator transient differential pressure limit. Generic Letter 86-13, identified potential inconsistencies between Technical Specifications and FSAR analysis related to Primary Coolant Pump operation and PCS boron concentration. Our review resulted in the proposed changes which add restrictions on operation of the plant to be consistent with the Palisades FSAR safety analysis for main steam line break, rod ejection and rod withdrawal incidents. Also, the steam generator maximum transient differential pressure limit of 1530 psi is being removed and replaced with an operating differential pressure limit of 1380 psi that is consistent with NRC approved steam generator tube plugging criteria.

Limited Dist

A001 1/1

Change: PDR Ltr 1 Encl NP
LPDR 1 NP
NSIC 1 NP

OC1287-0051A-NL02-NL04

See Attached 4/7/88 Ltr Release Right of Copy Material

During the Palisades refueling outage, presently scheduled to begin in September 1988, modifications will be made to improve the capabilities of the Reactor Protection System (RPS). These modifications include the addition of a Variable High Power Trip, the addition of an Axial Shape Index Alarm, the implementation of an improved Thermal Margin/Low Pressure Trip, the addition of an alarm that monitors the maximum cold leg temperature and the modification of the High Rate Trip Bypass hardware. The modified RPS will provide comprehensive protection of the fuel for all Standard Review Plan, Chapter 15, events. The modifications are summarized below.

Variable High Power Trip (VHPT)

The high power trip is being replaced by a variable high power trip. The VHPT will trip the plant when the reactor power increases less than or equal to 10% above the current power level. During power ascensions, the trip set point is increased by pressing a reset button. During power descents the trip set point automatically decreases. The minimum trip set point for the VHPT is less than or equal to 30% of rated power and the maximum set point is less than or equal to 106.5% of rated power.

The VHPT will provide early detection and termination of reactivity insertion transients starting at reduced power levels. Therefore, operator action to mitigate a slow boron dilution transient will not be necessary. Without this RPS modification, anticipated fuel assembly design changes and additional steam generator tube plugging would reduce available thermal margins. The VHPT will enhance core protection and help assure required safety margins are maintained while retaining the current 2,530 MWt rated power level.

Axial Shape Index (ASI) Alarm

The axial shape index (ASI) function is derived from the power range safety drawer excore indications for upper and lower neutron flux power. The new Thermal Margin calculators determine the ASI (lower-upper/lower+upper) and makes corrections for detector geometry. The corrected ASI is used in the TM/LP trip calculations discussed below. Also, positive and negative set points are generated as a function of measured core power. A control room panel alarm will be actuated if the corrected ASI is not bounded by the generated pair of set points.

The RPS modification will monitor the axial power distribution allowing operator action to prevent power operations outside the licensing basis analysis. The enhanced core protection is required for anticipated fuel assembly design changes.

Thermal Margin/Low Pressure (TM/LP) Trip

The current TM/LP trip analog calculators are being replaced by programmable digital calculators. The TM/LP trip is enhanced by using the maximum of the neutron flux power and the delta-T power as input parameters to directly calculate core power independent of core temperature. Also, the new TM/LP trip function will be corrected by the measured axial shape index.

Each component for the RPS modification reduced the uncertainty associated with the TM/LP trip. Therefore, additional operating margin was gained while maintaining the required thermal margins. The additional margin allowed reduced primary coolant flow rates while maintaining rated core power at 2,530 MWt.

T-Inlet Maximum Alarm

The purpose of this modification is to alert the operator of an impending limiting operating condition in order that appropriate action may be taken.

High Rate Trip Bypass

The power range nuclear/instrument drawer will be modified to allow an adjusted nuclear power signal to be sent to the lower power (16.5%) bypass bistable. This will allow proper high rate trip bypass function above 15% power without violating the Technical Specification minimum value.

Equipment to be removed or disabled (ie, function replaced by the Thermal Margin Calculators)

A. Existing thermal power (ΔT power) calculators manufactured by Bell and Howell. These are categorized as "associated circuits." They provide no input to the Reactor Protective Systems. They only provide input to the deviation meter (thermal power vs nuclear power), and an alarm to the operator.

B. Existing TM/LP calculators mounted in the rear of the main control boards. These calculators are comprised of GE/MAC function modules (summers, limiters, function generators). Their function will be replaced by the new Thermal Margin Calculators.

C. The existing high power trip signals to the RPS auxiliary trip units. The auxiliary trip units will be commanded by the new Thermal Margin Calculators (VHPT).

D. The power ratio calculator mounted at the top of panel C-27. The Thermal Margin Calculators perform a 24-hour and a seven day historical trending of ten parameters that will replace this function.

E. The X1/X10 pushbuttons. These are no longer required as a result of the described modifications.

In 1982, Consumers Power Company initiated a Thermal Margin Improvement Program for the Palisades Plant. During this evaluation, we determined that some nonconservative assumptions were used in 1977 while performing the Control Rod Withdrawal Transient Analysis. Therefore, a new analysis was performed by Advanced Nuclear Fuels (ANF) Corp (formerly Exxon Nuclear Company) and documented in XN-NF-83-57, Rod Withdrawal Transient Reanalysis for the

Palisades Nuclear Reactor. This reanalysis applied the additional thermal margin provided by the XNB DNB correlation, the calculated higher primary coolant flow at full power and improved pressurizer model in the revised PTSPWR2 computer code. Amendment No 82 to the Palisades Technical Specifications incorporated the results of this reanalysis.

Several other licensing issues were identified in 1983 which required the additional thermal margin, if a plant derate was to be avoided. These issues included Pressurized Thermal Shock, additional steam generator tube plugging, RTD time delay uncertainty, Asymmetric LOCA loads (Task A-2) and identification of reserve thermal margin to resolve future unknown licensing concerns. Two of the issues, Task A-2 and RTD delay time have been resolved as discussed below. Resolution of the pressurized thermal shock and steam generator tube plugging and reserve thermal margin issues resulted in modification of the Reactor Protection System hardware. Task A-2 was concluded by performing a primary coolant piping Leak-Before-Break Analysis sponsored by the CE Owners Group. RTD delay time is no longer an important issue since maintenance to the RTD thermowells has resulted in regaining fast RTD response times. Also, Consumers Power measured the response time of several primary loop RTDs and obtained acceptable results. Additional safety analysis using the current reactor protection system hardware was determined to provide little or no additional operating margin.

In addition to the above unresolved issues (pressurized thermal shock, steam generator tube plugging and reserve thermal margin) the Palisades RPS has other shortcomings which led to the decision to modify the system. To ensure protection of the fuel during transients starting at 50% power level, excessively conservative assumptions must be made in establishing the trip setpoints. The present trip setpoints are determined at part power conditions which restrict 100% full power operations. Also, the core power level is determined by measuring the hot leg and cold leg temperatures. Therefore, the indicated temperatures are affected by the time constant of the RTD which must be accounted for by making additional conservative assumption in establishing the trip setpoints.

The proposed modifications were reviewed by the current fuel vendor, ANF. After several discussions, the modifications were established ANF is very familiar with the standard Combustion Engineering analog reactor protection system having also completed a reload analysis for the St. Lucie Plant. Essentially the same analytical procedures were used by ANF to develop the new TM/LP trip constants for Palisades that were used for the St. Lucie Plant.

The setpoint analysis performed by ANF used the deterministic methodology (XN-NF-507, Rev 1, "ENC Setpoint Methodology for CE Reactors," July, 1980), rather than the Statistical methodology (XN-NF-507, Supplements 1 and 2, "ENC Setpoint Methodology for CE Reactors - Statistical Setpoint Methodology," September and November, 1982) to account for uncertainties. A generic approval was received on the statistical setpoint methodology and the deterministic methodology has been shown to be conservative with respect to the approved statistical methodology. The methodology to calculate the Limited Safety System Settings (LSSS) and Limiting Conditions for Operation (LCO) before uncertainties are applied to the setpoints are the same in the two methods.

The deterministic setpoint methodology has been submitted as a topical report but never received a generic SER.

The ANF "deterministic" setpoint methodology for CE reactors was submitted and reviewed in 1980 by the NRC. This methodology was subsequently used to develop the setpoints for the Ft. Calhoun reactor for Cycles 6 and 7. The deterministic methodology is currently being used as the basic methodology to determine the setpoints for the Palisades reactor. Revisions to the approach described are due to the differences between the Palisades system and a more standard CE system, and are described in the safety analysis reports.

The ANF setpoint methodology for CE reactors provides the bases for determining setpoints for both the deterministic and the statistical setpoint analysis methodologies. The difference in the two methods is the way the uncertainties are applied to the basic LSSS and LCO calculations. Historically, uncertainties have been included in setpoint analyses in a "deterministic" fashion; that is, each uncertain parameter was always selected to be at its more unfavorable limit. Each of the uncertainty variables were then stacked, which resulted in a very conservative setpoint. In the setpoint analyses employing the statistical uncertainty methodology, the uncertainties are accounted for by statistical convolution. This statistical approach gives a more realistic estimate of the actual limit and results in a less restrictive setpoint. The statistical approach to account for uncertainties in the determination of the setpoints for CE reactors was reviewed and approved by the NRC in their letter of December 16, 1985. A comparison of allowable power setpoints between the deterministic and statistical methodologies is shown in the NRC approved setpoint document. As shown by the comparison, the deterministic methodology results in more conservative setpoints than the statistical methodology.

Advanced Nuclear Fuels Corp (ANF) proprietary report No. ANF-87-150(P), Volume 2, with an affidavit executed by H E Williamson of ANF attesting to the report's proprietary nature, is attached. Volume 2 documents the results of the Standard Review Plan (SRP), Chapter 15, event analyses performed in support of Palisades operation with up to 29.3% steam generator tube plugging and the modified reactor protection system (RPS) which includes a variable high power trip and an improved thermal margin low pressure trip with axial shape monitoring.

Advanced Nuclear Fuels Corp proprietary report No. ANF-87-150(P), Volume 1, Disposition of SRP Chapter 15 events, Low Flow Trip Setpoint and Thermal Margin Analysis for Three Primary Coolant Pump Operation were attached to our December 23, 1987, submittal and are not being resubmitted herein.

The Gamma Metrics Company (GM) documents and reports listed and summarized below and attached hereto are also considered proprietary. In accordance with 10CFR2.790(b), affidavits executed by Clinton L Lingren of GM, attesting to the documents' proprietary nature, are also attached. The affidavit by Mr Lingren lists the documents below and, in addition, lists the GM Instruction Manual, Document No 070, Volumes I, II, III and IV. This Instruction Manual is not being provided with this submittal as it is not considered necessary for this amendment request by Consumers Power Company.

We are limiting the distributions of these proprietary documents to the NRC Document Control, Palisades Project Manager, and Region III.

1. General Purpose Class IE Qualified Microcomputer Hardware Specification (Document No 056)

   This document describes the hardware specifications for the Thermal Margin Monitor to be installed as a part of this modification. The description involves the designed environmental parameters and the input and output capabilities of the monitor.

2. Gamma Metrics Software Requirements Specification (Document No 055)

   This document provides the design specifications for each function of the Thermal Margin Monitor including associated inputs and outputs, the calculations which comprise those functions and the allowable values for adjustable constants used in the calculations. It also discusses the validation and verification requirements of IEEE/ANSI 74.3.2, 1982.

3. Gamma Metrics Software Design Description (Document No 89)

   This document provides a detailed description of the software subroutines the monitor utilizes to carry out each function.

4. Gamma Metrics Software Quality Assurance and Development Plan (Document No 068)

   This document establishes the procedures which provide assurance that the software conforms to established technical requirements and guides the software development process.

5. Gamma Metrics Validation Plan (Document No 088)

   This document established the procedures to be utilized in performing the ANSI/IEEE standard ANS 7-4.3.2 1982 validation of the monitor software.

6. Gamma Metrics Verification Plan (Document No 067)

   This document establishes the procedures for review and audit of the monitor software specifications and design documentations. It also provides procedures for the testing and analysis of the software.

7. Gamma Metrics RCS Series Thermal Margin Monitor Qualification Test Plan (Document No 066)

   This document describes the environmental test requirements for the monitor including seismic, aging, temperature, humidity, radiation, power supply voltage extremes, and electromagnetic interference.

Additionally, attached is Consumers Power Company's Specification J-54, Thermal Margin/Low-Pressure Modification. This document is the procurement specification which provides the general requirements and the codes and standards required for the thermal margin calculators.

A check for $150.00 as required by 10CFR170.12 was provided by letter on January 7, 1988; accordingly, no application fee is submitted herewith.

Additional information on the modifications to the Reactor Protection system to support this change request will be submitted following receipt.

Kenneth W Berry

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Director, Nuclear Licensing

CC Administrator, Region III, NRC
NRC Resident Inspector - Palisades

Attachment

CONSUMERS POWER COMPANY

DOCKET 50-255 - LICENSE DPR-20 - PALISADES PLANT

REQUEST FOR CHANGES TO TECHNICAL SPECIFICATIONS

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

A. Chapter 1 Specifications

In 1.1, Change "Axial Offset" to "Axial Offset or Axial Shape Index"

The thermal margin analysis performed by Advanced Nuclear Fuel, Inc (ANF) to support the revised TM/LP trip and the revised inlet temperature LCO incorporates the Axial Shape Index (ASI) terminology. This is an administrative change.

B. Chapter 2 Specifications

1. In 2.1, Safety Limits - Reactor Core

A. Applicability - replace the current statements with the following:

"This specification applies when the reactor is in hot standby condition and power operation condition."

B. Specification - replace the current specification with the following:

"The MDNBR of the reactor core shall be maintained greater than or equal to 1.17."

The TM/LP trip equations were added to Specification 2.3. This revision also removes Figures 2-1, 2-2 and 2-3 from the Technical Specifications. This change is consistent with the CE Standard Technical Specifications.

2. In 2.1, Basis, Replace all references to the W-3 DNB correlation and its safety limit of 1.3 with the XNB DNB correlation with a limit of 1.17. Delete the remainder of the paragraph start at "The curves for Figures 2-1, 2-2 and 2-3 represent..." Also, a new sentence was added to the last paragraph and References 1, 2 and 3 were changed.

Reference to the W-3 correlation has been deleted because the majority of the transients in Chapter 14 of the Palisades FSAR were analyzed or are bounded by transients analyzed using the

XNB DNB correlation. The primary exception is the steam line break event. Reference to two-pump and three-pump operations and reference to Figures 2-1, 2-2 and 2-3 were deleted to be consistent with changes made to Section 2.1, Specifications.

The sentence added to the end of the Basis justifies using the XNB DNB correlation for Palisades thermal margin analysis. This statement was transferred from the fourth paragraph of the 2.3, Basis. Reference 1 was changed to the appropriate document containing the development of the XNB DNB correlation and Reference 2 was changed to XNB DNB application document.

3. In 2.2, Basis, Delete the phrase "the pressurizer power-operated relief valves at 2400 psia and . . . ." Change the References to updated FSAR references as shown.

The PORVs are inoperable during power operation.

4. In 2.3, Specifications, add the TM/LP trip equations. (See page changes).

This change was made to improve the format of the Palisades Technical Specifications. This change is consistent with the CE Standard Technical Specifications.

5. In Table 2.3.1, Reactor Protection System Trip Setting Limits, delete column pertaining to 2-pump operations, replace Item 1, "High Power" with "Variable High Power" and provide appropriate set points for 4-pump and 3-pump operations. Revise set point for Item 2, "Primary Coolant Flow" for 3-pump operations, revise set points for Item 4, "Thermal Margin/Low Pressure" for 3-pump and 4-pump operations. Revise Notes 1, 2, 3 and 4.

The column giving set points for 2-pump operations was deleted since this mode of operation is being deleted from the Technical Specifications. The new RPS hardware incorporates a Variable High Power trip which limits power operations less than or equal to 10% above the indicated power level to provide protection for such events such as the boron dilution transient. The appropriate set points for both 4-pump and 3-pump operations have been given. The 3-pump thermal margin analysis also justified lowering the low flow trip from 71% to 60% of the 4-pump flow. The referenced figure for Item 4, "Thermal Margin/Low Pressure" was deleted and "P_T" was replaced with "P_{trip}" since the TM/LP trip limits have been incorporated in Section 2.3, Specification. For 3-pump operations, "High Power Level Trip" was replaced with "Variable High Power Trip" due to the hardware changes. The 3-pump thermal margin analysis justified increasing the high power trip from 39% to 49%.

The former note 1 was deleted since the times ten selector switch is to be removed during this plant modification. The new note 1 clarifies the trip setpoint for power operation at less than 20% power. Note 2 was changed to be consistent with FSAR Section 7.2.5.2, Trip Bypass Logic. The first sentence of Note 3 was deleted since the hot leg temperature is not used in the TM/LP trip equation (see Page 2-4 of the Specifications) and the units for the cold leg temperature is given in Specification 3.1.1g. Everything following the first "1750 psia" in Note 3 was deleted since the safety analysis is based upon this assumption. Reference to 2-pump operation was deleted from Note 4 since power operations with less than 3 PCPs is being deleted from the Technical Specifications.

6. In 2.3, Basis, Item 1, change the heading from "High Power" to "Variable High Power," and replace the text in the first paragraph with a new paragraph describing how the VHPT functions and delete the third and fourth paragraphs. Reference 1 was updated and Reference 2 was deleted (see page changes).

   The RPS modification includes a VHPT which replaces the existing High Power trip. The discussion pertaining to different trip set points for 3-pump operation was deleted because they are adequately described in the basis section for the Low Primary Coolant Flow Trip. The fourth paragraph was deleted since the X10 switch will be removed by the plant modification. Reference 1 was changed to the new basis document and Reference 2 was deleted due to changes made to the FSAR.

7. In 2.3, Basis, Item 2, delete all references to 2-pump operation and segregate discussions pertaining to 3-pump operations into one paragraph at the end of the subsection. The notation referring to Reference 3 was changed to Reference 4 and Reference 3 was moved to the first sentence of the section. The notation referring to Reference 5 was moved to the second paragraph. Also, References 4 and 5 were updated (see page changes).

   The references to 2-pump operations were deleted for the previously stated reasons. Segregation of the discussion on 3-pump operations was done for clarification. The historical flow rate data for different pump combinations was deleted from the FSAR, therefore, the notation referring to Reference 5 was moved. Reference 4 and Reference 5 were changed to the new basis documents. The notation referring to Reference 3 was moved to a more appropriate sentence.

8. In 2.3, Basis, Item 3, delete two sentences beginning with "The power-operated" . . . through "operations of the safety valves." Reference 11 was updated.

The LTOP system is disarmed when the PCS average temperature is greater than 430°F. Reference 11 was changed to the new basis document.

9. In 2.3, Basis, Item 4, essentially the entire basis description for the TM/LP trip was rewritten and reference to 2-pump operation was deleted. Reference 7, 13, 14 and 15 were deleted and Reference 12 was updated (see page changes).

   This section was rewritten because the existing analog hardware has been replaced by a microprocessor and the form of the trip function was altered significantly. Reference to 2-pump operation was deleted since this mode of power operation is no longer allowed by the Technical Specifications.

   FSAR Section 3.3.6 has been deleted, therefore Reference 7 was deleted. Reference 12 was changed to the new basis document. Reference 13 was replaced by Reference 12 and Reference 14 is no longer needed since the event was reanalyzed in ANF-87-150(P), Volume 2, Section 15.4. Reference 15 was moved to Section 2.1.

10. In 2.3, Basis, Item 5, change "provide a 15 minute margin before auxiliary feedwater is required" to "allow a safe and orderly plant shutdown and to prevent steam generator dryout assuming minimum auxiliary feedwater capacity." Also, Reference 9 was updated.

    Steam generator water level is not an explicit acceptance criterion. However, the analysis shows that sufficient steam generator water level is maintained to ensure an adequate heat sink until the primary coolant system temperature and pressure are reduced below the initiation threshold of the shutdown cooling system.

    Reference 9 was changed to the new basis document.

11. In 2.3, Basis, Item 7, change to read "... containment high-pressure trip is provided to assure that the reactor is shut down before the initiation of the safety injection system and containment spray.[10]" Delete the last sentence and change reference 10 to "Updated FSAR, Section 7.2.3.9."

    The containment high-pressure reactor trip has been set to initiate prior to SIS and no longer is an identical trip as previously described.

12. In 2.3, Basis, Item 8, include "primary coolant flow" in the second sentence. Also, replace "is" with "and low flow trip are" in the fourth sentence.

Changes make the basis statement consistent with FSAR, Section 7.2.5.2, Trip Bypass Logic, subsection Zero Power Mode Bypass.

C. Chapter 3

1. In 3.1.1, Operable Components, Item a, insert the words "with a flow rate greater than or equal to 1500 gpm" following "shutdown cooling pump" and add "and the plant is operating in cold shutdown or above, except during an emergency loss of coolant flow situation. Under these circumstances, the boron concentration may be increased with no primary coolant pumps or shutdown cooling pumps running." to the end of the paragraph.

   The phrases were added to ensure that the plant operations are bounded by assumptions used in the safety analysis pertaining to boron dilution events.

2. In 3.1.1, Operable Components, Item b, change "above 5% of rated power" to "above hot shutdown" and delete the parenthetical exception.

   This change is made to prevent continual reactor operation except when all the reactor coolant pumps are operating. The present plant safety analysis supports plant operation when all four coolant pumps are operating. The present specification allows continual operation up to 5% power with no requirements with regard to primary coolant pump operation. Thus as a result of this change, the reactor cannot be made critical unless four pumps are operating. The parenthetical was deleted since the requirements are adequately specified in Section 2.3.

3. In 3.1.1, Operable Components, Item b and c, move the third paragraph of Item c to Item b after making the following change: 1) delete the first sentence, 2) replace "Following loss of a pump" with "Before removing a pump from service," 3) replace "or more pumps" with "pump," 4) replace "the pumps to" with "the pump to," 5) in the last two sentences change "hot standby" to "hot shutdown" (two places) and 6) add "and power operations with less than three pumps is not permitted" the last sentence.

   This paragraph was moved from Item c to Item b to place all of the pump operability requirements in one specification. The first sentence was deleted because it would have been redundant. The second change was made because the reactor power must be reduced below the 3 pump allowed power level to select the 3 pump trip set points before a pump can be removed from service. Reference to 1 or 2 pump operation was deleted because required safety analyses have not been performed.

The present specification requires hot standby which allows unlimited operation of the reactor up to 2% power. This change requires the plant to be in hot shutdown within 24 hours of turning off a primary coolant pump. Thus the time is limited to 24 hours when the reactor can be operated after a pump has been shut off.

The last phrase was added to enforce previous statements pertaining to 1-pump and 2-pump operations.

4. In 3.1.1, Operable Components, Item c, delete the first sentence, delete the parenthetical in the second sentence, change "126.9" to "124.3" and delete the third sentence beginning with "In the event."

   The allowed power and flow rates for 4-pump and 3-pump operations are adequately specified in Section 2.3. The parenthetical describing the primary coolant flow rate measurement technique was deleted because the heat balance method has been adopted. The new safety analysis calculations were performed using a full power core flow rate based upon a hot zero power flow rate of 124.3 M-lbm/hr. The procedure for changing the TM/LP trip was deleted since adequate justification does not exist at the present time.

5. In 3.1.1, Operable Components, Item e(1), change "operating transient differential" to "operating differential" and change "1530 psi" to "1380 psi."

   The maximum transient differential pressure cannot be controlled by operator actions. Therefore, it is being deleted and replaced by the operating differential pressure limit of 1380 psi. The 1530 psi limit was added to the Palisades technical specification by Amendment No. 20 (April 26, 1976) before RG 1.121 was issued.

   The structural integrity of the steam generator tubes are assured by appropriately selecting the tube plugging criteria. The NRC has approved our plugging criteria in the SER dated June 11, 1984. Since the requirements of RG 1.121 are satisfied, the current transient differential limit of 1530 psi is not required. Also, the Combustion Engineering Standard Technical Specifications (NUREG-0212, Rev 2) reference RG 1.121 in Section 4.4.6.4 as the basis for determining the plugging limit.

6. In 3.1.1, Operable Components, Item g, deleted the "100%" adjective, revise the inlet temperature LCO equation and delete the note. Delete Figure 3.0, Reactor Inlet Temperature vs Operating Pressure and added a new Figure 3.0, ASI LCO for T Inlet Function. Also, add the following two paragraphs:

"When the ASI exceeds the limits specified in Figure 3.0, within 15 minutes, initiate corrective actions to restore the ASI to the acceptable region. Restore the ASI to acceptable values within one hour or be at less than 70% of rated power within the following two hours."

"If the measured primary coolant system flow rate is greater than 130 M lbm/hr, the maximum inlet temperature shall be less than or equal to the T_{Inlet} LCO at 130 M lbm/hr."

The "100%" was deleted because the LCO has been extended to include power levels less than 100%. The ASI restraints were added to allow higher core inlet temperatures at full power operating conditions. The revised inlet temperature LCO was developed in the transient analysis report ANF-87-150(P), Vol 2.

The inlet temperature LCO provided protection against penetrating DNB during the most limiting transient from full power operation. The most limiting transient for Palisades is the inadvertent drop of a full length control rod without a reactor trip. The transient analysis report shows that adequate thermal margin is available for power level $\leq$ 70% to allow ASI to exceed values expected to bound plant operations.

7. In 3.1.1, <u>Basis</u>, following the sentence ending with "operated at rated capacity.", insert the following sentence "By imposing a minimum shutdown cooling pump flow rate of 1500 gpm, sufficient time is provided for the operator to terminate the boron dilution under asymmetric flow conditions[6]." Added Reference 6 to the list of references for Section 3.1.1.

   This statement is supported by the analysis documented in ANF-87-150(P), Volume 2, Section 15.4.6.3.2.

8. In 3.1.1, <u>Basis</u>, add a second paragraph as follows:

   "The FSAR safety analysis was performed assuming four primary coolant pumps were operating for accidents that occur during reactor operation. Therefore, reactor startup above hot shutdown is not permitted unless all four primary coolant pumps are operating. Operation with less than four primary coolant pumps is permitted for a limited time to allow the restart of a stopped pump or for reactor internals vibration monitoring and testing."

   This change to the basis explains that the FSAR safety analysis is based on four primary coolant pumps operating and that a limited time is allowed for restart of a pump or for testing.

9. In 3.1.1, <u>Basis</u>, add reference number "(3)" to "1380 psi" in the third paragraph.

The stated reference demonstrates steam generator tube integrity by meeting Regulatory Guide 1.121, August 1976, requirements.

10. In 3.1.1, Basis, delete the last paragraph on page 3-2 beginning with "The maximum transient."

    The maximum steam generator transient differential pressure limit is being deleted. The basis for the deletion is described above.

11. In 3.1.1, Basis, in the paragraph beginning with "The transient analysis" change "126.9" to, "124.3."

    This is the vessel flow rate used in the new transient analysis performed to establish the proposed RPS set points.

12. In 3.1.1, Basis, replace the sentence beginning with "A DNB analysis" with the following:

    "A DNB analysis was performed in a parametric fashion to determine the core inlet temperature as a function of pressure and flow for which the minimum DNBR is equal to 1.17. This analysis includes the following uncertainties and allowance: 2% of rated power for power measurement; ±0.06 for ASI measurement; ±50 psi for pressurizer pressure; ±7°F for inlet temperature; and 3% measurement and 3% bypass for core flow. In addition, transient biases were included in the derivation of the following equation for limiting reactor inlet temperature:[4]."

    These are the assumptions documented in the transient analysis report ANF-87-150(P), Volume 2, Section 15.0.7.1, for the development of the inlet temperature LCO.

13. In 3.1.1, Basis, Replace the existing inlet temperature LCO equation with the following equation:

    $$T_{Inlet} \leq 543.3 + 0.575(P-2060) + 0.00005(P-2060)**2 + 1.173(W-120) - 0.0102(W-120)**2$$

    The new RPS hardware along with using the XNB DNB correlation provides additional thermal margin which allows higher inlet temperatures. The supporting analysis is documented in ANF-87-150(P), Volume 2, Section 15.0.7.1.

14. In 3.1.1, Basis, Delete the paragraph beginning with "A temperature measurement uncertainty" up to sentence "The limits of validity..."

    The inlet temperature LCO equation incorporates all of the appropriate uncertainty and bias factors.

15. In 3.1.1, Basis, Change "1850" and "2250" and "110" to, "1800," "2200" and "100" respectively and add the following statements: 1) "ASI as shown in Figure 3.0." and 2) "With measured primary coolant system flow rates > 130 M lbm/hr, limiting the maximum allowed inlet temperature to the T_{Inlet} LCO at 130 M lbm/hr increases the margin to DNB for higher PCS flow rates.

    These are the limits of validity for the revised inlet temperature LCO equation.

16. In 3.1.1, Basis, Add the following paragraph describing the ASI alarm following the above stated limits for the inlet temperature LCO equation.

    "The Axial Shape Index alarm channel is being used to monitor the ASI to ensure that the assumed axial power profiles used in the development of the inlet temperature LCO bound measured axial power profiles. The signal representing core power ($\dot{Q}$) is the auctioneered higher of the neutron flux power and the Delta-T power. The measured ASI calculated from the excore detector signals and adjusted for shape annealing (Y_I) and the core power constitute an ordered pair (Q,Y_I). An alarm signal is activated before the ordered pair exceed the boundaries specified in Figure 3.0."

    The modified RPS will provide a new alarm when the measured ASI is not within the acceptable range defined in the new Figure 3.0.

17. In 3.1, References, change Reference 1 and 2 to Updated FSAR and change Reference 3 from XN-NF-77-18, to Palisades 1983/1984 Steam Generator Evaluation and Repair Program Report, Section 4, April 19, 1984.

    The transient differential limit is being replaced by the operating differential limit of 1380 psi.

18. In 3.1, References, change Reference Number 4 from "XN-NF-77-22" to "ANF-87-150(P), Volume 2, Section 15.0.7.1.

    Reference 4 was changed to the new design basis document.

19. In 3.1.7, Basis, change Reference 1 to new Updated FSAR Section 4.3.9.4 and change Reference 2 to "ANF-87-150(P), Volume 2, Section 15.2.1.

    The loss of load transient reanalysis is documented in ANF-87-150(P).

20. In Section 3.5, Basis delete the last paragraph and in References, change Reference 1 to "Updated FSAR, Section 10.2.1, Reference 2 to "ANF-87-150(P), Volume 2, Section 15.2.7 and delete Reference 3.

Reference 1 identifies updated FSAR reference and Reference 2 was changed to the new design basis document. MSIV closure time is an input for the steam line break analysis.

20A. In Figure 3-6, Central Rod Insertion Limits, change to title the top figure from "TWO OR THREE PUMP OPERATION" to "THREE PUMP OPERATION."

Plant operation with less than 3 PCP operating is being deleted from the Technical Specifications.

20B. Change specification 3.10.1c to read as follows:

"At less than the hot shutdown condition, boron concentration shall be greater than cold shutdown boron concentration for normal cooldowns and heatups, i.e., non-emergency conditions."

This changes the required shutdown margin for less than four coolant pumps at less than hot shutdown from 2% shutdown to cold shutdown boron concentration. This ensures that if a main steam line break occurred in this situation, the reactor would not experience a return to power. This makes the shutdown margin requirment consistent with the present intent of 3.10.1b. This extra shutdown margin is not imposed in emergency situations where the time necessary to borate to the cold shutdown boron concentration would delay operator actions necessary to mitigate accidents such as a steam generator tube rupture.

21. In Section 3.10, Control Rod and Power Distribution Limits, delete section 3.10.2 and the reference to 3.10.2.b contained in Section 3.10.7.

The cycle specific safety analysis confirms that the individual rod worth is acceptable. Therefore, the present Technical Specification requirements are not necessary.

22. In 3.10, Basis, first paragraph, change the third sentence to read as follows:

"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 References 1 and 2 and additional analysis."

Also, change reference 5 to 2 in the last line of the first paragraph. Add new reference 2, ANF-87-150(P), Volume 2 to the list of references, and delete the second paragraph which begins with "The maximum individual rod worth...".

The current analysis of record (reference 2) supports a 2.5 second control rod drop time.

The second paragraph of the basis section was deleted to be consistent with the removal of Technical Specification 3.10.2 above.

In last paragraph change footnote (5) to (1).

In References delete 1 thru 4, change 5 to 1 and add new reference 2. References 1 thru 4 were not used in text.

22A. In 3.10, Basis, add the following after the third sentence in the first paragraph:

"Requiring the boron concentration to be at the cold shutdown boron concentration at less than hot shutdown assures adequate shutdown margin exists to ensure a return to power does not occur if an unanticipated cooldown accident occurs. This requirement applies to normal operating situations and not during emergency conditions where it is necessary to perform operations to mitigate the consequences of an accident."

This basis change is self explanatory.

23. In 3.11.2, Applicability, add item (3) which follows:

"3) Item b., above is applicable for each channel of the TM/LP and Axial Shape Index (ASI) alarm."

This change identifies the use of excore measured ASI in the Thermal Margin Calculator.

24. In 3.11.2, ACTION, add ACTION 3 which follows:

"ACTION 3:

When the measured AO uncertainty is greater than specified in Specification 4.18.2, the TM/LP trip function and the ASI alarm setpoints shall be conservatively adjusted within twelve (12) hours or that channel shall be declared inoperable. The operability requirements for TM/LP and ASI are given in Table 3.17.1 and 3.17.4, respectively.

This change specifies the applicability requirements for the excore measured ASI as related to the TM/LP trip. Extension of the LCO is justified since the plant operations will be bounded by the licensing basis.

25. In 3.11.2, Basis, change "The APL considers both LOCA and DNB based LHR limits," to, "The APL considers LOCA based LHR limits."

    With the current modification of the RPS, ASI monitoring has become an integral parameter in the TM/LP trip. Therefore, the APL function does not have to provide DNB protection.

26. In 3.12, Moderator Temperature Coefficient of Reactivity, change Specifications to read:

    "The moderator temperature coefficient (MTC) shall be less positive than + 0.5×10^{-4} $\Delta\rho/°F$ at $\leq$ 2% of rated power."

    This change is supported by the safety analysis performed to support Palisades operation (ANF-87-150(P), Volume 2).

27. In 3.12, Basis, delete the entire section and replace it with the following:

    "The limitations on moderator temperature coefficient (MTC) are provided to ensure that the assumptions used in the safety analysis (1) remain valid."

    The procedural discussions are no longer required since the specification has been changed from rated power conditions to $\leq$ 2% of rated power conditions.

28. In 3.12, Reference, change reference 1 to "ANF-87-150(P), Volume 2, Section 15.0.5.

    Reference 1 was changed to the new basis document.

29. In 3.17, Basis, (1) in sentence beginning with "If the bypass is not affected" change "high power" to, "variable high power" and (2) in the sentence beginning with "At rated power" change "high power" to "variable high power."

    The high power trip has been replaced with the variable high power trip.

30. In 3.17, Basis, delete "and from exceeding dropped rod peaking factors in the event that a turbine runback signal is required from the power range channels" from the first paragraph on page 3-77.

    The deleted sentence is obsolete as the turbine runback feature was disabled prior to being licensed at rated power of 2530 MWt.

30A. In 3.17, Basis, add a new last paragraph as follows:

"The Zero Power Mode Bypass can be used to bypass the low flow, steam generator low pressure and TM/LP trips[(2)] for all four Reactor Protective system channels to perform control rod testing or to perform low power physics testing below normal operating temperatures. The requirement to maintain cold shutdown boron concentration when in the bypass condition provides additional assurance that an accidental criticality will not occur. To allow low power physics testing at reduced temperature and pressure, the requirement for cold shutdown boron concentration is not required and the allowed power is increased to 10^{-1}%.

This change explains the use of the Zero Power Mode bypass switch.

31. In 3.17, References, change Reference 1 to Updated FSAR and change Reference 2 to Updated FSAR, Section 7.2.5.2.

32. In Table 3.17.1, Item 2, change "High-Power Level" to, " Variable High Power Level." Under the "Permissible Bypass Conditions", add the following at the end of the condition statement for Items 4, 6 and 9.

"...and greater than cold shutdown boron concentration."

The high power trip will be replaced with a variable high power trip.

This change adds the requirement to be at greater than cold shutdown boron concentration when using the Zero Power Mode bypass to bypass the Terminal Margin/Low Pressure Trip, Low Flow Trip, and Low Steam Generator Pressure Trip. These trips are in the tripped condition when the primary coolant system is below normal operating temperature and pressure. The Zero Mode bypass is used to energize the control rod drive clutches to perform testing when at less than operating conditions. Because a rod withdrawal accident is possible and is not analyzed under these conditions, the additional boron provided by going to cold shutdown boron concentration provides additional assurance that the reactor will not go critical should a rod withdrawal accident occur when the reactor is above cold shutdown.

32A. In Table 3.17.1, add the following at the end of footnote (e):

"...and cold shutdown boron concentration is not required."

This change allows low power physics testing to be performed and is necessary due to changes made in Item 32 above. The Zero Power Mode bypass is only required for low power physics testing when the testing is being performed at temperatures and pressure below the normal operating range. Special test procedures and precautions will be in place during this time.

32B. In Table 3.17.2, Item 4, add reference to footnote (f) and add footnote (f) to the table.

Footnote (f) was added to couple the operability requirements for AO and the operability requirements for TM/LP trip.

33. In Table 3.17.4, Item 16, change "Excore Detector" to "Excore Detector Deviation Alarms," change "None" to "Not Required Below 25% of Rated Power," change footnote "g" to state "Calculate the Quadrant Power Tilt using the excore readings at least once per 8 hours when the excore detectors deviation alarms are inoperable, or at least once per 8 hours using symmetric incore detectors when the difference between the excore and incore measured Quadrant Power Tilt exceeds 2%."

This change will clarify Item 16. Requiring calculations every 8 hours and not requiring the alarms to be operable below 25% of rated power is consistent with Technical Specification 3.23.3. Addition of calculating Quadrant Power Tilt using symmetric incore detectors when the difference between the excore and incore measured Quadrant Power Tilt exceeds 2% is consistent with current practice.

34. In Table 3.17.4, add new Item 17, Axial Shape Index (see page changes) and add footnote (i) as follows:

"AO operability requirements are given in Specification 3.11.2."

ASI is a new feature of the modified RPS. Footnote (i) was added to couple the operability requirements for AO and the operability requirements for the ASI alarm.

35. In 3.23.1, Action 3, add to "to monitor LHR" to the second line of the paragraph.

The words were added to clarify when the action is applicable.

36. In 3.23.1, Basis, delete "In addition, the limitation ... enveloped by the design power distribution" and add reference (3) at the end of the first sentence.

Prior to the inclusion of ASI monitoring in the TM/LP trip, it was necessary to limit linear heat rates (LHR) in order to ensure that thermal margin was not compromised when operating with LHR's that differed from those used in the safety analysis.

However, with the current modification to the RPS, ASI monitoring has become an integral parameter in the TM/LP trip. Therefore, with the modified RPS, if thermal margin is threatened by LHR, the reactor will trip provided that the radial peaking factor limits (Table 3.23-2) are observed. Reference 3, XN-NF-78-16, provides the basis for Figure 3.23-1.

37. In 3.23.1, References, delete Reference 2, and Reference 5.

    Statements made based upon these references were deleted.

38. In Figure 3.23-1, delete the line labeled "DNB" and remove the excess shading.

    See the preceding justification for Section 3.23.1, Basis.

39. In 3.23.2, Radial Peaking Factors, change "the following quantity (1.0 + 0.5(1-P)) where P is the" to, "the following quantity. The quantity is (1.0 + 0.3(1-P)) for P ≧ .5 and the quantity is 1.15 for P < .5. P is the... ."

    The allowed radial peaking factor at partial powers was reduced to provide additional thermal margin for transients initiated at power levels less than full power.

40. In 3.23.2, Applicability, change 50% to 25%.

    The lowest power level at which meaningful power distribution data can be obtained is 25%. This change is being made to ensure plant operations even at low power levels are bounded by the licensing basis analysis.

41. In 3.23.2, Action, change to read as follows:

    "ACTION:

    1. For P < 50% of rated with any radial peaking factor exceeding its limit, be in at least hot shutdown within 6 hours.

    2. For P ≥ 50% of rated with any radial peaking factor exceeding its limit, reduce thermal power within 6 hours to less than the lowest value of:

$$\left[1 - 3.33\left(\frac{F_r}{F_L} - 1\right)\right] \times \text{Rated Power}$$

    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."

This change is necessary to ensure that the Palisades Tech Specs adequately protects against unanalyzed part-power (i.e., P < 50% of rated) transients.

42. In 3.23.2, Basis, insert "variable" before "high-power trip" in the first sentence.

    Change to conform with hardware modification.

43. In 3.23.3, change APPLICABILITY to read "...above 25%..." and change ACTION statements 1 and 2 to read as follows:

    "1. With quadrant power tilt determine to exceed 5% but less than or equal to 10%:

    a. Correct the quadrant power tilt within 2 hours after exceeding the limit, or

    b. 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

    c. Reduce power, at the normal shutdown rate, to less than 85% of rated power and determine that the radial peaking factors are within the limits of Section 3.23.2. At reduced power, determine at least once every 8 hours that the radial peaking factors are within the limits of Section 3.23.2."

    "2. With quadrant power tilt determine to exceed 10%:

    a. Correct the quadrant power tilt within 2 hours after exceeding the limit, or

    b. Reduce power to less than 50% of rated power within the next 2 hours and determine that the radial peaking factors are within the limits of Section 3.23.2. At reduced power, determine at least once every 8 hours that the radial peaking factors are within the limits of Section 3.23.2."

    These changes ensure that the thermal margin limits established by the transient analysis are not violated by excessive quadrant tilt.

D. Chapter 4

1. Change the following in Table 4.1.1:

    a. In Item 1.d., add footnote 6. (Delete obsolete footnote 6 in three places in Table).

One of the changes as a result of the RPS Modification is the addition of a Variable High Power Trip (VHPT). The VHPT setpoint is maintained by the Variable High Power Function in the Thermal Margin Calculator. The addition of footnote 6 assures that this function is tested to verify proper performance at least once every 18 months. The test will be performed by applying known power inputs (flux power and/or ΔT power) to the Variable High Power Function. The Variable High Power Function will be tested for both 3 and 4 primary coolant pump operation.

Performing the above testing every 18 months is deemed adequate, since the Variable High Power Function is also being tested less extensively on a monthly basis during the performance of Item 1.c.

The obsolete footnote (6) referred to a 1981 deferred surveillance.

b. In Item 1.c., delete the reference to footnote 4.

The VHPT setting will be tested for the operating pump combination and power level only. Since the High Power Trip will be changed to a VHPT, verification of trip settings involves more extensive testing which will now be covered by Item 1.d. The monthly testing required by Item 1.c. will provide assurance that the settings and trip circuitry are performing per design. Values for minimum and maximum setpoints (see Table 2.3.1) for both 3 and 4 primary coolant pump operation will be checked during performance of Item 14.a. The testing and checks performed by Items 1.d. and 14.a. provide adequate assurance that the 3 primary coolant pump VHPT settings will perform per design.

c. In Item 4.b.(1), change

"known resistance substituted for RTD coincident with known pressure input."

to, "known resistance substituted for RTD coincident with known pressure and power input."

Thermal Margin/Low Pressure (TM/LP) trip settings will be a function of core inlet temperature, power level, and excore measured axial offset (ASI). Previous equipment used to calculate TM/LP settings did not use an ASI input, and core power was indirectly input based on core inlet and exit temperatures.

The TM/LP temperature input calibration will require a known power input (lower core power, upper core power and total core power) in addition to a known pressure input.

Surveillance requirements for power and ASI inputs into TM/LP are adequately covered by Items 1, 12 and 13 and need not be reiterated by Item 4 for TM/LP.

The TM/LP utilizes the power and ASI as inputs into the Power Peaking Function and the Axial Function respectively. During performance of Item 4.b.(1), the Power Peaking Function and the Axial Function will be tested to verify proper performance.

d. In Item 4.c., delete reference to footnote 4.

Footnote 4 is no longer applicable for TM/LP, since the same TM/LP settings are applied to both 3 and 4 primary coolant pump operations. For 3 pump operation, TM/LP is protected by the VHPT and the 1750 psia minimum low pressure setting.

e. Add a new Item 12 to the table (see page changes).

One of the changes as a result of the RPS Modification is the addition of the Axial Shape Index (ASI) alarm. ASI is monitored against setpoints to assure that the core inlet temperature equation remains valid, and ASI is used as an input into TM/LP.

The ASI alarm setpoints are calculated by the LOCA Peaking Function and the Local Power Density Function, using a total core power input. The surveillance requirements for the power input are adequately covered by Items 1 and 13 and need not be reiterated by Item 12 for ASI.

The ASI value is calculated by the ASI Function based on lower core and upper core power inputs from the power range safety channels. The ASI value is checked against the total core axial offset measured by the incores at least every 31 days of power operation. This surveillance requirement is covered by 4.18.2.1.b. and need not be reiterated by Item 12, ASI.

The surveillance requirement of Item 12.a. will test the Axial Shape Index Function, LOCA Peaking Function, and the Local Power Density Function to verify proper performance. The ASI alarming function will also be verified. The surveillance requirements given above, along with the constant check required by Item 14, provide adequate assurance that the ASI alarming function is performing per design and the ASI value is accurate.

f. Add a new Item 13 to the table and in Section 4.1 Basis, third paragraph add "and ΔT power channels" (See page changes).

This item has been added since ΔT Power is now a part of the RPS. Items 13.a. and b. provide assurance that the ΔT Power value is accurate. The temperature input used to calculate ΔT Power is calibrated by Item 4.b.(1) and need not be reiterated by Item 13 for ΔT Power. The surveillance requirement of Item 13.c. will provide adequate assurance that the circuitry used for the ΔT Power calculation is performing per design.

g. Add a new Item 14 to the table (see page changes).

At least once every 92 days the constants used by the Thermal Margin Calculator will be verified. The constants are used in calculating ΔT Power, TM/LP temperature input, ASI Function, Axial Function, Power Peaking Function, LOCA Peaking Function, Local Power Density Function, VHPT minimum and maximum trip settings, TM/LP minimum pressure setting, TM/LP calculated pressure settings, and core inlet temperature setpoint.

The Surveillance requirements of Items 1.c., 1.d., 4.a.(1), 4.b.(1), 4.c., 12.a., 13.a., 13.b., 13.c. as well as 4.18.2.1.b. indirectly provide assurance that the correct constants are being used. Additionally, changes to constants are administratively controlled, and a keylock must be used to modify any constants. The surveillance requirement of Item 14.a. provides adequate assurance that correct constants are being used by the Thermal Margin Calculator.

h. In Footnote 3, change

"Adjust the nuclear gain pot on the ΔT cabinet until readout agrees with heat balance calculations".

to, "Adjust the nuclear power or ΔT power until readout agrees with heat balance calculations when above 15% of rated power".

This footnote clarifies that the adjustment is applicable to both nuclear power and ΔT power. Nuclear adjustment will be performed using the gain adjustment on Panel C-27. The ΔT power will be adjusted by changing a constant in the Thermal Margin Calculator. Normally the "BIAS" constant in the ΔT power equation will be used to adjust ΔT power. If necessary, the "Kα" constant may also be changed to adjust ΔT power. Changes to constants will be controlled through Operating Procedures.

Footnote 3 was also expanded to only require the daily check and adjustment, if necessary, to be performed when above 15% of rated power, which is consistent with the Standard CE Technical Specifications. Above 15% power, the heat balance is considered accurate enough to make the daily check meaningful.

2. Change the following in Table 4.1.3:

   a. In Item 9, change "Flux - T Power Comparator"

      to, "Flux - ΔT Power Comparator".

      In addition, the surveillance method for Item 9.b. is changed from "Internal Test Signal"

      to, "Use simulated signals".

      The Flux - ΔT Power Comparator will no longer have an internal test signal capability. A simulated signal will be used to verify the performance of the Flux - ΔT Power alarm.

   b. In Item 8.a., change "...all rod drive control system interlocks..."

      to, "...all manual rod drive control system interlocks..."

      The Reactor Regulating System is not used to provide automatic control of the reactor, therefore there is no need to test the automatic rod drive control system interlocks.

   c. Delete footnotes (1) & (2) and reference to them in Items 2.c., 3.c. and 10.a. The obsolete footnotes refer to past refueling outages.

3. Change the following in Section 4.15:

   a. Change "...shall be made with four primary coolant pumps in operation before the reactor is made critical".

      to, "...shall be made with four primary coolant pumps in operation. This measurement shall be made within the first 31 days of rated power operation."

      By omitting "before the reactor is critical", more than one method may be used to verify primary system flowrate. The surveillance requirement of 4.15 is presently met by performing a flow measurement at hot zero power using the primary coolant pump Δ pressure method. A more accurate flow measurement method is the calorimetric method, which is used by most Combustion Engineering plants. The words "before the reactor is made critical" must be deleted to allow use of the calorimetric method, since it is performed at or close to rated power. A time requirement was specified to assure the measurement is made within a reasonable time following the refueling or steam generator tube plugging outage.

4. Change the following in 4.18.2.1:

a. Change Item b from,

"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"

to "Individual excore channel measured AO shall be compared to the total core AO measured by the incores. If the difference is greater than 0.02, the excore monitoring system shall be recalibrated."

Excore measured AO was previously determined using an average value from all power range safety channels. The average value was referred to as the Power Ratio and was recorded on NR-0100. The Power Ratio was multiplied by the Shape Annealing Factor (SAF) to obtain an excore measured AO. The new equipment being installed as a result of the RPS Modification allows the excore measured AO to be determined from each individual power range safety channel. Section 4.18.2.1(b) was rewritten to cover this enhanced monitoring capability.

5. Change the following in 4.19.1.2:

a. Change Item a from

"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."

to "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 for each operable channel using the previous 24 hourly recorded values".

b. Change Item d from

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

to "Continuously verify that the measured AO is within 0.05 of the established target AO for at least 3 of the 4, 2 of the 3 or 2 of the 2 operable channels, whichever is the applicable case."

Due to the equipment changes as stated in the discussion regarding Specification 4.18.2.1 changes, Specification 4.19.1.2 also had to be rewritten since each individual power range safety channel is now capable of monitoring Linear Heat Rate.

Specification 4.19.1.2(a) was expanded to require the last 24 hour history be checked for each operable channel. Specification 4.19.1.2(a) was also expanded to clarify that the last 24 hour history is verified by reviewing the last 24 hourly recorded values. Previously, the last 24 hour history could be verified to be within 0.05 continuously using the Power Ratio Recorder, NR-0100. This continuous 0.05 verification can no longer be performed, since the new equipment only records the last 24 hour history in 1 hour intervals. Reviewing the previous 24 hourly recorded values is sufficient to determine that the axial power distribution is near equilibrium, and significant xenon oscillations are not present.

Specification 4.19.1.2(d) was changed to require continuous verification of the ±0.05 limit while using the excore monitoring system to monitor Linear Heat Rate. The monitoring capabilities of the new equipment will now allow this continuous verification. The ASI alarm setpoints in the Thermal Margin Calculator will be reset to conservative values to meet the ±0.05 requirement and Figure 3.0 requirements, whichever are more restrictive.

Changes that have been made to Specification 4.19.1.2 which allow use of each operable channel for monitoring LHR and allow coincident signals to be the basis for determining when the measured AO has deviated from the target AO by more than 0.05 are supported by XN-NF-80-47. XN-NF-80-47, "Palisades Power Distribution Control Procedures", was the original basis of 4.19.1.2.

Since the last 24 hour history can not be verified continuously, Specification 4.19.1.2(a) conservatively requires that each operable channel be within the 0.05 limit over the last 24 hours. Two coincident signals indicating greater than the 0.05 limit will not be required. Any one value outside the 0.05 limit will mean that the excores can not be used for monitoring LHR.

6. Add new section 4.20 to read as follows:

"4.20 MODERATOR TEMPERATURE COEFFICIENT (MTC) SURVEILLANCE REQUIREMENTS

4.20.1 The MTC shall be determined to be within its limits by confirmatory measurements prior to initial operation above 2% of rated thermal power, after each refueling."

This change ensures that the moderator coefficient is maintained within its specified limit.

II. ANALYSIS OF NO SIGNIFICANT HAZARDS CONSIDERATION

The above proposed Technical Specifications changes are divided into three categories for ease in clarity in providing the bases for determination of no significant hazards considerations. The first is the change related to inconsistencies between the plant safety analysis and Technical Specifications resulting from Generic Letter 86-13. The second part will be related to the changes to the Technical Specifications resulting from the Reactor Protection System (RPS) modification. The third part is the change of the steam generator "transient differential pressure" limit to "operating differential" limit.

The changes to the Technical Specifications resulting from the RPS modifications and the analyzed steam generator transient differential pressure revision have been determined by the Plant Review Committee and Nuclear Safety Board to involve an unreviewed safety question. The analysis below pursuant to 10CFR50.92 justifies the conclusion the changes do not involve a significant hazards consideration.

1. Inconsistencies between Safety Analysis and Technical Specifications

   The changes proposed resulting from Generic Letter 86-13 review are to decrease the consequences of the main steam line break and the rod ejection accident and prevent the occurrence of the rod withdrawal accident. The changes limit reactor operation with less than 4 primary coolant pumps operating. It requires increased shutdown margin and decreased K-effective to prevent a return to power during a main steam line break, and reduces the possibility of criticality occurring during a rod ejection or rod withdrawal accidents when less than 4 pumps are operating and/or the zero power mode bypass is used. Therefore, there has been no increase in probability or consequences of an accident previously evaluated.

   No new or different type of accident is created by increasing shutdown margin, the boron concentration, or requiring the reactor to be in hot shutdown vs hot standby. All of these changes are more restrictive requirements than the present Specifications.

The margin of safety is increased by the changes which align plant operation to the safety analysis.

2. Reactor Protection System Modifications

The changes to the Reactor Protection System (RPS) Technical Specifications do not result in a significant increase in the probability of an accident. ANF report ANF-87-150(P), Volume 2, provides analysis of Standard Review Plan (SRP) Chapter 15 events in support of Palisades operation with up to 29.3% steam generator tube plugging and the modified RPS. The modified RPS includes a variable overpower trip and an improved thermal margin/low pressure trip with axial monitoring. The changes to the Technical Specifications provide appropriate limits for RPS initiation. However, the changes do not affect the initiation of events but instead ensure RPS response to the events. Surveillance requirements have been modified to ensure equipment operability such that the RPS will be operable and capable of performing its function.

The possibility of a new or different kind of accident has not been created by the changes to the Technical Specifications corresponding to the modifications of the RPS. As noted above, the RPS Technical Specifications are mitigating parameters that are provided to ensure appropriate RPS response to an initiating event. The parameters, as determined in ANF-87-150(P), are established such that appropriate margin exists to the acceptance criteria design limits.

The changes to the RPS Technical Specifications, including increasing the steam generator tube plugging limit, do affect the consequences of an accident and the previously defined margin of safety. The following analysis provides justification for conclusion that no significant increase in consequences and no significant reduction in the margin of safety are involved with these changes.

In the Disposition of Events report (ANF-87-150(P), Volume 1), each event described in the Standard Review Plan (SRP) was reviewed and dispositioned into one of the following categories:

1. 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;

2. The event is bounded by another event which is to be analyzed;

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

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

The SRP events which were dispositioned as categories 1, 3, and 4 are listed in the attached tables 1, 2 and 3 respectively. The remaining 22 SRP events were bounded by category 1 events. The accident category for each analyzed event is listed in Table 1 and the acceptance criteria for each accident category is given on pages 11 and 12 of the transient analysis report (ANF-87-150(P), Volume 2). The results reported in the transient analysis report confirm that event acceptance criteria are met for current fuel cycle number 7. The results support operation with up to 29.3% average steam generator tube plugging at a rated thermal power of 2530 MWt.

The plant operating parameters and associated uncertainties used in the existing licensing basis for cycle 7 are the same values used in the new transient analysis report. Even though the transient analysis was performed assuming a larger number of plugged steam generator tubes, additional operating margin was obtained. The additional operating margin was obtained by (1) using the approved XNB DNB correlation in place of the W-3 correlation, (2) upgrading the reactor protective system, and (3) monitoring additional plant parameters with new technical specification requirements. As a result, higher inlet temperatures are allowed and the revised TM/LP trip is less restrictive at nominal operating conditions. More details on the impact of the three changes are given in the following paragraphs.

XNB DNB Correlation

The W-3 DNB correlation was developed by Westinghouse before the Palisades Plant received its operating license in 1971. The approved license limit for the W-3 correlation is 1.3 based upon its ability to match the DNB test data. Since then, additional DNB test data has been obtained using more accurate data collection techniques and using test assemblies accurately representing the Palisades fuel design. Since the data scatter was reduced and data not representative of the Palisades design could be removed from the data base, the license limit for the ANF developed XNB DNB correlation is 1.17. Even though both correlations provide 95% probability at a 95% confidence level that DNB will not occur, the XNB correlation provides additional operating margin for the Palisades Plant while maintaining the same margin of safety.

RPS Modification

The two main components of the RPS modification are the replacement of analog circuits with a digital Thermal Margin Calculator, the replacement of the high power trip with the variable high power trip and the enhancement of the TM/LP trip.

The Thermal Margin Calculator has the following attributes that reduces the measurement uncertainty associated with the TM/LP trip.

1. A high degree of accuracy in setting each the TM/LP trip coefficients is provided.

2. The coefficients do not drift during plant operations and monitoring of the values of each input parameters is greatly enhanced during plant operations.

3. The form of the TM/LP trip in the Thermal Margin Calculator is exactly the same form as used in the transient analysis performed by ANF. Additional biasing of the TM/LP trip will no longer be required to ensure that plant operations are bounded by the transient analysis.

The increased accuracy of the Thermal Margin Calculators provide additional operating margin while maintaining the existing margin of safety.

The new variable high power trip will terminate slow reactivity transients (ie, the Boron Dilution Transient) initiated at reduced power levels at 10% above the starting power level. Therefore, additional operating margin is available because these transients no longer restrict full power operations.

The TM/LP trip was enhanced by using the maximum of the neutron flux power and the delta-T power as an input parameter instead of using the hot leg and cold leg temperatures to implicitly measure core power. The TM/LP trip actuation is no longer strongly dependent on the response times of the PCS RTDs and the loop transport time between the cold leg and hot leg RTDs. Therefore, the uncertainty in the TM/LP trip associated with core power measurement has been significantly reduced.

The new TM/LP trip function also includes a term dependent on the measured Axial Shape Index (ASI). The current licensing basis analysis is based upon design axial power profiles at 100% power and at 50% power. The expected axial power distribution at full power with all control rods withdrawn will result in more available thermal margin than the assumed design axial power profiles. On the other hand, the fuel will be protected if adverse axial power distribution should develop during plant power operations.

Each component of the RPS modification reduces the uncertainty associated with the TM/LP trip. Therefore, additional operating margin was gained without reducing the margin of safety.

Monitoring Additional Plant Parameters

The benefits of monitoring the neutron flux power level and ASI in the TM/LP trip was described in the preceding discussion. The RPS modification also includes continuous monitoring with an alarm for the measured ASI. The new T-inlet LCO was developed based upon allowed power levels as a function of ASI which is included in the proposed technical specification change. Therefore, the probability of initiating a transient outside the specified envelope is very small. Additional operating margin was made available when plant parameters are near nominal conditions without decreasing the margin of safety.

The margin of safety for accident categories also include limits on maximum PCS pressure and off-site radiological consequences. The calculated 2.3% of fuel failure for event 15.4.1 is greater than for the same transient in FSAR section 14.2.2.2. This would appear to be a reduction in available margin of safety, but the two transients are not equivalent. The FSAR analysis was performed by Combustion Engineering for the initial fuel cycle using input assumptions similar to the current ANF analysis except for one parameter. The FSAR analysis was performed using four PCP flow rate while the ANF analysis assumed only three PCP were operating. A peak heat flux equivalent to 92% of rated power occurred at 13.9 seconds in the ANF analysis. If four PCP flow rate had been used in the analysis, the XNB DNBR would have been greater than 1.17 since the peak heat flux did not exceed the equivalent of 100% of rated power. Since no fuel failure would have been predicted, the margin of safety for radiological consequences has not decreased as a result of the new operating conditions.

The accident category for a Control Rod Ejection is a limiting fault event. The analysis contained in the transient analysis report is the first Palisades plant specific Control Rod Ejection analysis performed in accordance with the requirements of the Standard Review Plan. The NRC staff accepted the previous results because no fuel pellet realized an average enthalpy greater than 280 cal/gm. During the Systematic Evaluation Program, the NRC staff did not require that this accident be reanalyzed because they determined that the radiological consequences would be acceptable for the assumed 10% fuel failures. ANF determined that the radiological consequence of the calculated 12.2% fuel failures are below the 10 CFR 100 dose limits and the whole body dose is less than 25% of the respective 10 CFR 100 limit. The consequences of this event are acceptable, but the margin of safety has decreased because the calculated 12.2% fuel failures is greater than the 10% figure assumed by the NRC during SEP. The hardware modification made to the RPS did not affect the margin of safety for this event. The deposited enthalpy analysis criterion will be addressed on a cycle specific basis in accordance with the approved ANF methodology.

The acceptance criteria for Operating Events through Infrequent Events require that the pressure in the reactor coolant loop and main steam systems should be less than 110% of design value, or 2750 psia. The Loss of External Load (15.2.1) event analysis predicted that the maximum PCS pressure would be less than or equal to 2584.7 psia. The flow rate of two of the three safety valves is sufficient to maintain the PCS pressure below the acceptance criteria. The calculated peak pressures in the new transient analysis are higher than in the current licensing basis. The higher calculated pressures are the result of changes to the pressurizer model in the PTSPWR2 computer code. Since the margin between the maximum safety valve lift pressure of 2605 psia (ie 2580 +/- 25 psia) has not changed and the relieving capacity of only two safety valves is required, the actual margin of safety has not decreased as a result of the proposed RPS modifications and resulting transient analysis.

10CFR50, Appendix A, Criteria 10, 20, 25, and 29 require that the design and operation of the plant and the reactor protective system assure that the Specified Acceptable Fuel Design Limits (SAFDLs) not be exceeded during Anticipated Operational Occurrences (A00). 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 A00s. A summary of the analysis results is contained in the attached Table 4.

The SRP events listed in Table 4 which are not included in the definition of A00s are 15.3.3, 15.4.1 and 15.4.8. Significant margin exists between the SAFDLs and the transient analysis results for the A00 events. Therefore, the margin of safety as determined by the SAFDLs was not decreased as a result of the RPS modification.

The preceding review of the transient analysis identified that the margin of safety decreases only for the Control Rod Ejection event. The original margin of safety could not be determined since the FSAR analysis was not performed using the current guidelines contained in SRP Section 15.4.8. The increased doses are still bounded by the 10 CFR 100 limits.

3. Steam Generator Operating Differential Pressure - 1380 psi

The changes to the Technical Specifications was made to identify the Steam Generator Operating Differential Pressure limit of 1380 psi. The present Specification describes the transient differential pressure of 1530 psi. The reanalysis calculates the maximum transient differential pressure at 1604.4 psi. No changes to the plant were made to affect the change in maximum differential pressures. The changes occurred due to changes in the full vendor's transient analysis methodology. As a result, the probability of an accident has not been increased. The consequences have not been increased because although the analyzed transient differential pressure is higher it does not result in failure of the steam generator tubes. The possibility of a new or different type of accident has not been created by this reanalysis of the maximum transient differential pressure.

The calculational methods have resulted in a reduction in the margin of safety however, this reduction is not considered significant as described below:

The loss of load event was analyzed by ANF and the results are reported in the Transient Analysis report ANF-87-150(P), Volume 2. The calculated maximum differential pressure (Case 2) is 1604.4 psi. The previous analysis for the same event predicted that the current transient limit of 1530 psia would not be exceeded (XN-NF-77-18). Since both analyses used essentially the same input data, the different results are caused by changes made to the PTSPWR2 computer code. The current approved version of the code utilizes an improved non-equilibrium pressurizer model. The predicted PCS pressure response during plant transients are more realistic using the current version of the code. Therefore, the reduced margin of safety was not caused by a change at the Palisades plant, but by a change in transient analysis methodology at the fuel vendor.

The structural integrity of the steam generator tubes is assured by appropriately selecting the tube plugging criteria. The NRC has approved current plugging criteria in the SER dated June 11, 1984. The appropriate requirements of RG 1.121 were shown to be satisfied without reference to the transient differential limit of 1530 psi. Even though some local yielding of the steam generator tubes may occur for the calculated loss of load event, a large margin to tube failure still exists. (Reg. Guide 1.121 requires a factor of three between operating differential pressure and tube rupture for an allowable defect.) The steady state steam generator differential pressure limit will be given in Specification 3.1.1.e as 1380 psi and the steam generator tube plugging limit is given in Specification 4.14, Augmented Inservice Inspection Program for Steam Generators. Therefore, the 1530 psi transient differential limit can be replaced with the more appropriate steady state limit of 1380 psi with no significant reduction of the margin of safety for the Palisades plant.

III. CONCLUSION

The Palisades Plant Review Committee and the Nuclear Safety Board have reviewed this Technical Specifications Change Request and have determined the changes resulting from the Reactor Protection system

Modifications and the analyzed steam generator transient differential pressure do 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 of accident 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 Specifications 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 Robert J Nicholson

Robert J Nicholson, Vice President
Energy Supply Services

Sworn and subscribed to before me this 25th day of March 1988.

Beverly Ann Avery

Beverly Ann Avery, Notary Public
Jackson County, Michigan
My commission expires December 27, 1988

TABLE 1

ACCIDENT CATEGORY USED FOR EACH ANALYZED EVENT

Event		Accident Category
15.1.3	Increase in Steam Flow	Moderate
15.2.1	Loss of External Load	Moderate
15.2.7	Loss of Normal Feedwater FLow	Moderate
15.3.1	Loss of Forced Reactor Coolant FLow	Moderate
15.3.3	Reactor Coolant Pump Rotor Seizure	Infrequent
15.4.1	Uncontrolled Bank Withdrawal at Subcritical or Low Power	Infrequent
15.4.2	Uncontrolled Bank Withdrawal at Power	Moderate
15.4.3	Control Rod Misoperation	Moderate
15.4.6	CVCS MalfunctionResulting in Decreased Boron Concentration	Moderate
15.4.8	Control Rod Ejection	Limiting Fault

TABLE 2

ANALYSIS SUMMARY
EVENTS BOUNDED BY EXISTING FSAR ANALYSIS

15.1.5	Steam Line Breaks
15.4.3(6)	Core Barrel Failure
15.4.4	Start-up of an Inactive Loop
15.6.3	Radiological Consequences of Steam Generator Tube Failure
15.6.4	Loss-of-Coolant Accidents
15.7	Radiological Release From a System or Component

TABLE 3

ANALYSIS SUMMARY
NON-APPLICABLE EVENTS

15.2.5	Steam Pressure Regulator Failure
15.3.2	Flow Controller Malfunction
15.4.3(3)	Malpositioning of the Part-Length Control Rod Group
15.4.5	Flow Controller Malfunction
15.4.9	Spectrum of Rod Drop Accidents (BWR)
15.6.4	Radiological Consequences of a Main Steamline Failure Outside Containment

TABLE 4

PALISADES MODIFIED REACTOR PROTECTION SYSTEM ANALYSIS RESULTS

Event		Peak Pressure, psia	Peak LHGR, kw/ft	MDNBR
15.1.3	Increase in Steam Flow	2110	13.2	1.497
15.2.1	Loss of External Load	2585	12.7	1.776
15.2.7	Loss of Feedwater	2272		
15.3.1	Loss of PCS FLow	2161	12.7	1.455
15.3.3	Pump Rotor Seizure	2010		1.409
15.4.1	URW at Low Power	2426		1.036*
15.4.2	URW at Power	2154	15.3	1.304
15.4.3	Control Rod Misoperation			
	Dropped Rod	2010		1.301
	Single Rod Withdrawal	2072	17.4	1.273
15.4.6	Boron Dilution	(Adequacy of SDM Demonstrated)		
15.4.8	Control Rod Ejection	2452		**

* Less than 2.3% Fuel Failures

** 12.2% Fuel Failures

ATTACHMENT

Consumers Power Company
Palisades Plant
Docket 50-255

AFFIDAVIT AND
ADVANCED NUCLEAR FUELS CORPORATION REPORT

PALISADES MODIFIED REACTOR PROTECTION SYSTEM REPORT:
ANALYSIS OF CHAPTER 15 EVENTS
ANF-87-150(P)

February 1988

A F F I D A V I T

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

I, H. E. Williamson being duly sworn, hereby say and depose:

1. I am Manager, Licensing and Safety Engineering, for Advanced Nuclear Fuels Corporation, ("ANF"), and as such I am authorized to execute this Affidavit.

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

3. I am familiar with the document ANF-87-150(P), Volume 2, entitled "Palisades Modified Reactor Protection System Report Analysis of Chapter 15 Events," referred to as "Document." Information contained in this Document has been classified by ANF as proprietary in accordance with the control system and policies established by ANF for the control and protection of information.

4. The Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by ANF and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in the Document as proprietary and confidential.

5. The Document has been made available to the U.S. Nuclear Regulatory Commission in confidence, with the request that the information contained in the Document will not be disclosed or divulged.

6. The Document contains information which is vital to a competitive advantage of ANF and would be helpful to competitors of ANF when competing with ANF.

7. The information contained in the Document is considered to be proprietary by ANF because it reveals certain distinguishing aspects of the analysis of the SRP Chapter 15 events which secure competitive advantage to ANF for fuel design optimization and marketability, and includes information utilized by ANF in its business which affords ANF an opportunity to obtain a competitive advantage over its competitors who do not or may not know or use the information contained in the Document.

8. The disclosure of the proprietary information contained in the Document to a competitor would permit the competitor to reduce its expenditure of money and manpower and to improve its competitive position by giving it extremely valuable insights into the analysis of the SRP Chapter 15 events and would result in substantial harm to the competitive position of ANF.

9. The Document contains proprietary information which is held in confidence by ANF and is not available in public sources.

10. In accordance with ANF's policies governing the protection and control of information, proprietary information contained in the Document has been made available, on a limited basis, to others outside ANF only as required and under suitable agreement providing for nondisclosure and limited use of the information.

11. ANF policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

12. This Document provides information which reveals the analysis of the SRP Chapter 15 event, developed by ANF over the past several years. ANF has invested many thousands of dollars and many man-months of effort in developing the analysis of the SRP Chapter 15 events revealed in the Document. Assuming a competitor had available the same background data and incentives as ANF, the competitor might, at a minimum, develop the information for the same expenditure of manpower and money as ANF.

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

FURTHER AFFIANT SAYETH NOT.

H. E. Williamson

SWORN TO AND SUBSCRIBED
before me this 1st day of
March, 1988.

Gloria R. Fitzgerald
NOTARY PUBLIC

8804130031

MEMO ROUTE SLIP
Form AEC-93 (Rev. May 14, 1947) AECM 0240

☐ See me about this.	☐ For concurrence.	☒ For action.
☐ Note and return.	☐ For signature.	☐ For information.

TO (Name and unit)	INITIALS / DATE	REMARKS
Don Lanham		Per the attached request, please remove the reports previously submitted by
		letters dated 12/23/87, 3/25/88, and 5/4/88. Notify all addressees on the corresponding RIDS
		sheets to either return their copies or to notify you of their destruction. When this is

FROM (Name and unit): Project Manager Palisades Plant Docket 50-255

REMARKS: completed, let me know so I can write a letter to the licensee. I have attached the appropriate RIDS sheets.

Thomas V. Wambach

PHONE NO. 21323 DATE 7/11/88

USE OTHER SIDE FOR ADDITIONAL REMARKS

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Consumers Power
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June 17, 1988

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DOCKET 50-255 - LICENSE DPR-20 - PALISADES PLANT -
REACTOR PROTECTION SYSTEM MODIFICATION AND TECHNICAL SPECIFICATIONS CHANGES
(TAC NO 66901).

Attachment 1 provides is a partial response to the Nuclear Regulatory Commission request for additional information (RAI) concerning the Palisades Reactor Protection System modification and Technical Specifications Change Request (TSCR). In addition to the response included in Attachment 2 are 4 Technical Specifications page changes containing minor revisions. Consumers Power Company previously submitted a draft TSCR and a TSCR on December 23, 1987, and March 25, 1988 respectively. These attached page changes replace those in the March 25, 1988 submittal. We expect to submit our response for the remaining items in the RAI in about one week.

Attachment 3 contains a description of the Reactor Protection System Modification. Attachment 4 provides requested wiring diagrams.

Attachments 5 and 6 contain proprietary GAMMA-METRICS (G-M) reports. Affidavits executed by Clinton L Lingren of G-M are included, in accordance with 10CFR2.790. The original affidavit supporting the proprietary nature of the G-M Instruction Manual (Attachment 5) was submitted with our March 25, 1988 Technical Specification Change Request. However, the manuals were withheld at that time by Consumers Power Company. As noted in the June 10, 1988 GAMMA-METRICS letter enclosed in Attachment 6 and Consumers Power Company letter dated April 7, 1988 the NRC is authorized to copy these proprietary documents for internal NRC distribution and review.

Don

Also, enclosed with this submittal are non-proprietary Advanced Nuclear Fuels (ANF) reports numbers ANF-87-150(NP) Volume 1 and 2 and XN-NF-86-91(NP). These replace previous proprietary and non-proprietary reports transmitted to the NRC on December 23, 1987, March 25, 1988 and May 4, 1988. With submittal of these reports we request that the previously transmitted proprietary and non-proprietary reports either be returned or destroyed. The enclosed reports are identical to the proprietary versions currently in your possession. ANF has consented to release the proprietary information in these reports in order to support review of the proposed Palisades Technical Specifications Change and completion of the Reactor Protection System modification.

James L Kuemin (Signed)

James L Kuemin
Staff Licensing Engineer

CC Administrator, Region III, NRC
NRC Resident Inspector - Palisades

Attachments