ATTACHMENT 1

CONSUMERS ENERGY COMPANY
PALISADES PLANT
DOCKET 50-255

TECHNICAL SPECIFICATIONS CHANGE REQUEST
COLR REFERENCES AND Pa

Proposed Pages

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6.5.14 Containment Leak Rate Testing Program

Programs shall be established to implement the leak rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. The Type A test program shall meet the requirements of 10 CFR 50, Appendix J, Option B and shall be in accordance with the guidelines of Regulatory Guide 1.163, "Performance-Based Containment Leakage-Test Program, dated September 1995." The Type B and Type C test program shall meet the requirements of 10 CFR 50, Appendix J, Option A, as modified by the exemption from certain requirements of 10 CFR 50 Appendix J which was granted in an NRC letter to Consumers Power Company dated December 6, 1989.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 53 psig (FSAR Table 14.18.1-4).

The maximum allowable containment leak rate, L_a, at P_a, shall be 0.1% of containment air weight per day.

Leak rate acceptance criteria are:

a. Containment leak rate acceptance criteria is $\leq 1.0\ L_a$. During the first plant startup following testing in accordance with this program, the leak rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests;

b. Air lock leak rate acceptance criteria is $\leq 0.023\ L_a$ for each door, when pressurized to ≥ 10 psig.

The Surveillance interval extensions of LCO 4.0.2 are not applicable to the Containment Leak Rate Testing Program requirements.

The provisions of LCO 4.0.3 are applicable to the Containment Leak Rate Testing Program requirements.

6.6.5 Core Operating Limits Report (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

3.1.1 ASI Limits.
3.10.5 Regulating Group Insertion Limits
3.23.1 Linear Heat Rate (LHR) Limits
3.23.2 Radial Peaking Factor Limits

b. The analytical methods used to determine the core operating limits shall be those approved by the NRC, specifically those described in the latest approved revision of the following documents:

1. EMF-96-029(P)(A) Volumes 1 and 2, "Reactor Analysis System for PWRs," Siemens Power Corporation, (LCOs 3.1.1, 3.10.1, 3.10.5, 3.23.1, & 3.23.2)

2. ANF-84-73 Appendix B (P)(A), "Advanced Nuclear Fuels Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," Advanced Nuclear Fuels Corporation. (Bases report not approved) (LCOs 3.1.1, 3.10.5, 3.23.1, & 3.23.2)

3. XN-NF-82-21(P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company. (LCOs 3.1.1, 3.23.1, & 3.23.2)

4. EMF-84-093(P)(A), "Steam Line Break Methodology for PWRs, "Siemens Power Corporation. (LCOs 3.10.1, 3.10.5, 3.23.1, & 3.23.2)

5. XN-75-32(P)(A) Supplements 1 through 4, "Computational Procedure for Evaluating Fuel Rod Bowing," Exxon Nuclear Company. (Bases document not approved) (LCOs 3.1.1, 3.10.5, 3.23.1, & 3.23.2)

6. EXEM PWR Large Break LOCA Evaluation Model as defined by: (LCOs 3.10.5, 3.23.1, & 3.23.2)

   a) XN-NF-82-20(P)(A) Supplement 2, "Exxon Nuclear Company Evaluation Model EXEM/PWR ECCS Model Updates," Exxon Nuclear Company.

   b) XN-NF-82-20(P)(A) Supplements 1, 3, and 4, "Exxon Nuclear Company Evaluation Model EXEM/PWR ECCS Model Updates," Advanced Nuclear Fuels Corporation.

   c) XN-NF-82-07(P)(A), "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," Exxon Nuclear Company.

   d) XN-NF-81-58(P)(A) Supplements 1 and 2, "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company.

6.6.5 COLR (continued)

e) ANF-81-58(P)(A) Supplements 3 and 4, "RODEX2 Fuel Rod Thermal Mechanical Response Evaluation Model," Advanced Nuclear Fuels Corporation.

f) XN-NF-85-16(P)(A) Volume 1 and Supplements 1, 2, and 3; Volume 2, and Supplement 1, "PWR 17x17 Fuel Cooling Tests Program," Advanced Nuclear Fuels Corporation.

g) XN-NF-85-105(P)(A) and Supplement 1, "Scaling of FCTF Based Reflood Heat Transfer Correlation for Other Bundle Designs," Advanced Nuclear Fuels Corporation.

7. XN-NF-78-44(NP)(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," Exxon Nuclear Company. (LCOs 3.10.5, 3.23.1, & 3.23.2)

8. ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation. (LCOs 3.1.1, 3.10.5, 3.23.1, & 3.23.2)

9. EMF-92-153(P)(A) and Supplement 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," Siemens Power Corporation. (LCOs 3.1.1, 3.23.1, & 3.23.2)

10. XN-NF-621(P)(A), "Exxon Nuclear DNB Correlation for PWR Fuel Designs," Exxon Nuclear Company. (LCOs 3.1.1, 3.23.1, & 3.23.2)

11. XN-NF-82-06(P)(A) and Supplements 2, 4, and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup," Exxon Nuclear Company. (LCOs 3.1.1, 3.10.5, 3.23.1, & 3.23.2)

12. ANF-88-133(P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWD/MTU," Advanced Nuclear Fuels Corporation. (LCOs 3.1.1, 3.10.5, 3.23.1, & 3.23.2)

13. XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," Exxon Nuclear Company. (LCOs 3.1.1, 3.10.5, 3.23.1, & 3.23.2)

14. EMF-92-116(P)(A), "Generic Mechanical Design Criteria for PWR Fuel Designs," Siemens Power Corporation. (LCOs 3.1.1, 3.10.5, 3.23.1, & 3.23.2)

15. EMF-2087(P)(A), "SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications," Siemens Power Corporation. (LCOs 3.10.5, 3.23.1, & 3.23.2)

16. ANF-87-150 Volume 2, "Palisades Modified Reactor Protection System Report: Analysis of Chapter 15 Events," Advanced Nuclear Fuels Corporation. [Approved for use in the Palisades design during the NRC review of license Amendment 118, November 15, 1988] (LCOs 3.1.1, 3.23.1, & 3.23.2)

6.6.5 COLR (continued)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid cycle revisions or supplements, shall be provided, upon issuance for each reload cycle, to the NRC.

6.6.6 Reserved

6.6.7 Accident Monitoring Instrument Report

When a report is required by Condition 3.17.4.7c, "Accident Monitoring Instrumentation," a report shall be submitted within the following 30 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels to OPERABLE status.

6.6.8 Containment Structural Integrity Surveillance Report

Reports shall be submitted to the NRC covering Prestressing, Anchorage, and Liner and Penetration tests within 90 days after completion of the tests.

6.6.9 Steam Generator Tube Surveillance Report

The following reports shall be submitted to the Commission following each inservice inspection of steam generator tubes:

a. The number of tubes plugged in each steam generator shall be reported to the Commission within 15 days following the completion of each inspection, and

b. The complete results of the steam generator tube inservice inspection shall be reported to the Commission within 12 months following completion of the inspection. This report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged.

c. Results of steam generator tube inspections that fall into Category C-3 shall require 24 hour verbal notification to the NRC prior to resumption of plant operation. A written followup within the next 30 days shall provide a description of investigations and corrective measures taken to prevent recurrence.

ATTACHMENT 2

CONSUMERS ENERGY COMPANY
PALISADES PLANT
DOCKET 50-255

TECHNICAL SPECIFICATIONS CHANGE REQUEST
COLR REFERENCES AND Pa

Existing Pages Marked to Show Proposed Changes

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The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is ~~52.64~~53 psig (FSAR Table 14.18.1-4).

The maximum allowable containment leak rate, L_a, at P_a, shall be 0.1% of containment air weight per day.

Leak rate acceptance criteria are:

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The Surveillance interval extensions of LCO 4.0.2 are not applicable to the Containment Leak Rate Testing Program requirements.

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6.0 ADMINISTRATIVE CONTROLS

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~~8. ANF-1224(P)(A), "Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," and Supplement 1(P)(A); Advanced Nuclear Fuels Corporation. (LCOs 3.1.1, 3.23.1, & 3.23.2)~~

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