



UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D.C. 20555-0001

CONSUMERS ENERGY COMPANY

DOCKET NO. 50-255

PALISADES PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.188  
License No. DPR-20

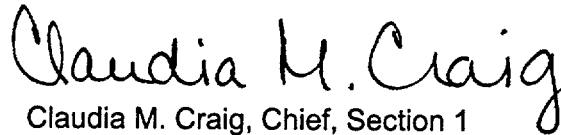
1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment by Consumers Energy Company (the licensee) dated October 29, 1999, as supplemented November 2, 1999, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
  - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
  - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public; and (ii) that such activities will be conducted in compliance with the Commission's regulations;
  - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public;
  - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to the license amendment and Paragraph 2.C.(2) of Facility Operating License No. DPR-20 is hereby amended to read as follows:

The Technical Specifications contained in Appendix A, as revised through Amendment No. 188 , and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. Consumers Energy Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION



Claudia M. Craig, Chief, Section 1  
Project Directorate III  
Division of Licensing Project Management  
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: November 15, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 188

FACILITY OPERATING LICENSE NO. DPR-20

DOCKET NO. 50-255

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE

iv  
6-17  
6-20  
6-21  
6-22

INSERT

iv  
6-17  
6-20  
6-21  
6-22

PALISADES PLANT TECHNICAL SPECIFICATIONS  
TABLE OF CONTENTS

<u>SECTION</u>	<u>DESCRIPTION</u>	<u>PAGE NO</u>
5.0	<u>DESIGN FEATURES</u> . . . . .	5-1
5.1	SITE . . . . .	5-1
5.2	CONTAINMENT DESIGN FEATURES . . . . .	5-1
5.2.1	Containment Structures . . . . .	5-1
5.2.2	Penetrations . . . . .	5-2
5.2.3	Containment Structure Cooling Systems . . . . .	5-2
5.3	NUCLEAR STEAM SUPPLY SYSTEM (NSSS) . . . . .	5-2
5.3.1	Primary Coolant System . . . . .	5-2
5.3.2	Reactor Core and Control . . . . .	5-3
5.3.3	Emergency Core Cooling System . . . . .	5-3
5.4	FUEL STORAGE . . . . .	5-4
5.4.1	New Fuel Storage . . . . .	5-4
5.4.2	Spent Fuel Storage . . . . .	5-4a
	Figure 5-1 Site Environment TLD Stations . . . . .	5-5
6.0	<u>ADMINISTRATIVE CONTROLS</u> . . . . .	6-1
6.1	RESPONSIBILITY . . . . .	6-1
6.2	ORGANIZATION . . . . .	6-1
6.2.1	Onsite and Offsite Organizations . . . . .	6-1
6.2.2	Plant Staff . . . . .	6-2
6.3	PLANT STAFF QUALIFICATIONS . . . . .	6-3
6.4	PROCEDURES . . . . .	6-4
6.5	PROGRAMS AND MANUALS . . . . .	6-5
6.5.1	Offsite Dose Calculation Manual . . . . .	6-5
6.5.2	Primary Coolant Sources Outside Containment . . . . .	6-6
6.5.3	Post Accident Sampling Program . . . . .	6-6
6.5.4	Radioactive Effluent Controls Program . . . . .	6-7
6.5.5	Reserved . . . . .	6-8
6.5.6	Primary Coolant Pump Flywheel Surv. Program . . . . .	6-8
6.5.7	Inservice Inspection and Testing Program . . . . .	6-8
6.5.8	Steam Generator Tube Surveillance Program . . . . .	6-9
6.5.9	Secondary Water Chemistry Program . . . . .	6-14
6.5.10	Ventilation Filter Testing Program . . . . .	6-15
6.5.11	Fuel Oil Testing Program . . . . .	6-16
6.5.12	Technical Specification Bases Control Program . . . . .	6-16
6.5.13	Reserved . . . . .	6-17
6.5.14	Containment Leak Rate Testing Program . . . . .	6-17
6.5.15	Process Control Program . . . . .	6-18
6.6	REPORTING REQUIREMENTS . . . . .	6-19
6.6.1	Occupational Radiation Exposure Report . . . . .	6-19
6.6.2	Radiological Environmental Operating Report . . . . .	6-19
6.6.3	Radioactive Effluent Release Report . . . . .	6-19
6.6.4	Monthly Operating Report . . . . .	6-19
6.6.5	Core Operating Limits Report . . . . .	6-20
6.6.6	Reserved . . . . .	6-22
6.6.7	Accident Monitoring Instrument Report . . . . .	6-22
6.6.8	Containment Structural Integrity Surveillance Report . . . . .	6-22
6.6.9	Steam Generator Tube Surveillance Report . . . . .	6-22
6.7	HIGH RADIATION AREA . . . . .	6-23

## 6.0 ADMINISTRATIVE CONTROLS

### 6.5.13 Reserved

### 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  $\geq 10$  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.

6.0 ADMINISTRATIVE CONTROLS

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:
    - 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.

6.0 ADMINISTRATIVE CONTROLS

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)

## 6.0 ADMINISTRATIVE CONTROLS

### 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.