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

March 30, 2001

Mr. Nathan L. Haskell
Director, Licensing and Performance Assessment
Palisades Plant
27780 Blue Star Memorial Highway
Covert, MI 49043

SUBJECT: PALISADES PLANT - ISSUANCE OF AMENDMENT RE: OPTION B CONTAINMENT LEAK RATE TESTING (TAC NOS. MB0855)

Dear Mr. Haskell:

The Commission has issued the enclosed Amendment No.194 to Facility Operating License No. DPR-20 for the Palisades Plant. The amendment consists of changes to the Technical Specifications (TS) in response to your application transmitted by letter dated December 7, 2000, as revised by letter dated January 12, 2001. The December 7, 2000, application superceded your earlier amendment request dated July 28, 2000.

The amendment changes the TSs to allow Type B and C containment leak rate testing to be performed in accordance with 10 CFR Part 50, Appendix J, Option B. The amendment also increases the interval in TS Surveillance Requirement 3.6.2.2 for containment air lock door interlock testing from 18 months to 24 months.

A copy of our related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Darl Hood

Darl S. Hood, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-255

Enclosures: 1. Amendment No. 194 to DPR-20
2. Safety Evaluation

cc w/encls: See next page

NRR-058

March 30, 2001

Mr. Nathan L. Haskell
Director, Licensing and Performance Assessment
Palisades Plant
27780 Blue Star Memorial Highway
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Sincerely,

/RA/
Darl S. Hood, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-255

Enclosures: 1. Amendment No. 194 to DPR-20
2. Safety Evaluation

cc w/encls: See next page

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**RTSB Input provided by G.Hubbard's 3/14/01 Memo
*Provided SE input by memo

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Palisades Plant

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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. 194
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 December 7, 2000, as revised January 12, 2001, 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. 194 , 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 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Claudia M. Craig

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: March 30, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 194

FACILITY OPERATING LICENSE NO. DPR-20

DOCKET NO. 50-255

Revise Appendix A of the Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3.6.1-1	3.6.1-1
3.6.1-2	---
3.6.2-4	3.6.2-4
3.6.2-5	---
5.0-21	5.0-21
5.0-22	5.0-22
---	5.0-23
B3.6.1-1	B3.6.1-1
B3.6.1-2	B3.6.1-2
B3.6.1-3	B3.6.1-3
B3.6.1-4	B3.6.1-4
B3.6.1-5	---
B3.6.2-2	B3.6.2-2
B3.6.2-7	B3.6.2-7
B3.6.2-8	B3.6.2-8
B3.6.2-9	---
B3.6.3-10	B3.6.3-10
B3.6.3-11	B3.6.3-11

3.6 CONTAINMENT SYSTEMS

3.6.1 Containment

LCO 3.6.1 Containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Containment inoperable.	A.1	Restore containment to OPERABLE status.	1 hour
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	B.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.1	Perform required visual examinations and leakage rate testing, except for containment air lock testing, in accordance with the Containment Leak Rate Testing Program.	In accordance with the Containment Leak Rate Testing Program
SR 3.6.1.2	Verify containment structural integrity in accordance with the Containment Structural Integrity Surveillance Program.	In accordance with the Containment Structural Integrity Surveillance Program

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.2.1	-----NOTES----- 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Results shall be evaluated against acceptance criteria applicable to SR 3.6.1.1. ---------- Perform required air lock leakage rate testing in accordance with the Containment Leak Rate Testing Program.	In accordance with the Containment Leak Rate Testing Program
SR 3.6.2.2	Verify only one door in the air lock can be opened at a time.	24 months

5.5 Programs and Manuals

5.5.13 Safety Functions Determination Program (SFDP) (continued)

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.14 Containment Leak Rate Testing Program

a. A program shall establish the leakage 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. This program shall be in accordance with the guidelines of Regulatory Guide 1.163, "Performance-Based Containment Leakage-Test Program," dated September 1995, as modified by the following exceptions:

1. Leakage rate testing is not necessary after opening the Emergency Escape Air Lock doors for post-test restoration or post-test adjustment of the air lock door seals. However, a seal contact check shall be performed instead.

   Emergency Escape Airlock door opening, solely for the purpose of strongback removal and performance of the seal contact check, does not necessitate additional pressure testing.

2. Leakage rate testing at P_a is not necessary after adjustment of the Personnel Air Lock door seals. However, a between-the-seals test shall be performed at $\geq$10 psig instead.

3. Leakage rate testing frequency for the Containment 4 inch purge exhaust valves, the 8 inch purge exhaust valves, and the 12 inch air room supply valves may be extended up to 60 months based on component performance.

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 53 psig. The containment design pressure is 55 psig.

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

5.5 Programs and Manuals

5.5.14 Containment Leak Rate Testing Program (continued)

d. Leakage rate acceptance criteria are:

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

2. Air lock testing acceptance criteria are:

a) Overall air lock leakage is $\leq 1.0\ L_a$ when tested at $\geq P_a$ and combined with all penetrations and valves subjected to Type B and C tests. However, during the first unit startup following testing performed in accordance with this program, the leakage rate acceptance criteria is $< 0.6\ L_a$ when combined with all penetrations and valves subjected to Type B and C tests.

b) For each Personnel Air Lock door, leakage is $\leq 0.023\ L_a$ when pressurized to $\geq$ 10 psig.

c) For each Emergency Escape Air Lock door, a seal contact check , consisting of a verification of continuous contact between the seals and the sealing surfaces, is acceptable.

e. "Containment OPERABILITY" is equivalent to "Containment Integrity" for the purposes of the testing requirements.

f. The provisions of SR 3.0.3 are applicable to the Containment Leak Rate Testing Program requirements.

g. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

5.5 Programs and Manuals

5.5.15 Process Control Program

a. The Process Control Program shall contain the current formula, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR 20, 10 CFR 71, Federal and State regulations, and other requirements governing the disposal of the radioactive waste.

b. Changes to the Process Control Program:

1. Shall be documented and records of reviews performed shall be retained as required by the Quality Program, CPC-2A. This documentation shall contain:

a) Sufficient information to support the change together with the appropriate analyses or evaluation justifying the change(s) and

b) A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

2. Shall become effective after approval by the plant superintendent.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment

BASES

BACKGROUND

The containment consists of a concrete structure lined with steel plate, and the penetrations through this structure. The structure is designed to contain radioactive material that may be released from the reactor core following a design basis Loss of Coolant Accident (LOCA). Additionally, this structure provides shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment is a reinforced concrete structure with a cylindrical wall, a flat foundation mat, and a shallow dome roof. The foundation slab is reinforced with conventional mild-steel reinforcing. The internal pressure loads on the base slab are resisted by both the external soil pressure and the strength of the reinforced concrete slab. The cylinder wall is prestressed with a post tensioning system in the vertical and horizontal directions. The dome roof is prestressed utilizing a three way post tensioning system. The inside surface of the containment is lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions.

The concrete structure is required for structural integrity of the containment under Design Basis Accident (DBA) conditions. The steel liner and its penetrations establish the leakage limiting boundary of the containment. Maintaining the containment OPERABLE limits the leakage of fission product radioactivity from the containment to the environment. SR 3.6.1.1 leakage rate requirements comply with 10 CFR 50, Appendix J, Option B (Ref. 4) as modified by approved exemptions.

The isolation devices for the penetrations in the containment boundary are a part of the containment leak tight barrier. To maintain this leak tight barrier:

a. All penetrations required to be closed during accident conditions are either:

    1. capable of being closed by an OPERABLE automatic containment isolation system, or

    2. closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.3, "Containment Isolation Valves";

BASES

BACKGROUND (continued)

b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks";

c. The equipment hatch is properly closed and sealed.

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBAs that result in a release of radioactive material within containment are a Loss of Coolant Accident (LOCA), a Main Steam Line Break (MSLB), and a control rod ejection accident (Ref. 1). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.10% of containment air weight per day (Ref. 3). This leakage rate is defined in 10 CFR 50, Appendix J, Option B as L_a; the maximum allowable leakage rate at pressure P_a. The P_a value of 53 psig represents the analytical value found in Reference 1, rounded up to the next whole number.

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

The containment satisfies Criterion 3 of 10 CFR 50.36(c)(2).

LCO

Containment OPERABILITY is maintained by limiting leakage to $\leq 1.0\ L_a$, except prior to the first startup after performing a required Containment Leak Rate Testing Program leakage test. At this time, the applicable leakage limits must be met.

Compliance with this LCO will ensure a containment configuration, including the equipment hatch, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

Individual leakage rates specified for the containment air lock (LCO 3.6.2) and purge valves which have resilient seals (LCO 3.6.3) are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J. Therefore, leakage rates exceeding these individual limits only result in the containment being inoperable when the leakage results in exceeding the overall acceptance criteria of $1.0\ L_a$.

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, containment is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from containment. The requirements for containment during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

A.1

In the event containment is inoperable, containment must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining containment OPERABILITY during MODES 1, 2, 3, and 4. This time period also ensures that the probability of an accident (requiring containment OPERABILITY) occurring, during periods when containment is inoperable, is minimal.

B.1 and B.2

If containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leak Rate Testing Program. Failure to meet individual air lock and containment isolation valve "local leak rate" leakage limits does not invalidate the acceptability of the overall leakage determination unless their contribution to overall Type A, B, or C leakage causes that leakage to exceed limits. As left leakage prior to the first startup after performing a required Containment Leak Rate Testing Program leakage test is required to be $< 0.6\ L_a$ for combined B and C leakage, and $\leq 0.75\ L_a$ for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq 1.0\ L_a$. At $\leq 1.0\ L_a$ the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by the Containment Leak Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

SR 3.6.1.2

This SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Structural Integrity Surveillance Program.

REFERENCES

1. FSAR, Chapter 14
2. FSAR, Section 14.18
3. FSAR, Section 5.8
4. 10 CFR 50, Appendix J, Option B

BASES

APPLICABLE SAFETY ANALYSES

The DBAs that result in a release of radioactive material within containment are a Loss of Coolant Accident (LOCA), a Main Steam Line Break (MSLB) and a control rod ejection accident (Ref. 1). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.10% of containment air weight per day (Ref. 2). This leakage rate is defined in 10 CFR 50, Appendix J, Option B, as L_a : the maximum allowable containment leakage rate at the calculated maximum peak containment pressure (P_a). For a LOCA, the calculated maximum peak containment pressure is approximately 53 psig. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock.

The containment air locks satisfy Criterion 3 of 10 CFR 50.36(c)(2).

LCO

Each containment air lock forms part of the containment pressure boundary. As part of the containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single OPERABLE door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into or exit from containment.

BASES

ACTIONS (continued)

D.1 and D.2

If the inoperable containment air lock cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of the Containment Leak Rate Testing Program.

This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria, were established during initial air lock and containment Operability testing. Subsequent amendments to the Technical Specifications revised the acceptance criteria for overall Type B and C leakage limits and provided new acceptance criteria for the personnel air lock doors and the emergency air lock doors (Ref. 2). The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall containment leakage rate. The Frequency is required by the Containment Leak Rate Testing Program.

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 2 has been added to this SR requiring the results to be evaluated against the acceptance criteria of SR 3.6.1.1. This ensures that air lock leakage is properly accounted for in determining the combined Type B and C containment leakage rate.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.2

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY while the air lock is being used for personnel transit into and out of containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is not normally challenged when the airlock is used for entry and exit (procedures require strict adherence to single door opening), this test is only required to be performed every 24 months. The 24 month frequency is based on the need to perform this Surveillance under the conditions that apply during plant outage, and the potential for loss of containment OPERABILITY if the Surveillance were performed with the reactor at power.

The 24 month Frequency for the interlock is justified based on generic operating experience. The Frequency is based on engineering judgment and is considered adequate given that the interlock is not normally challenged during use of the airlock.

REFERENCES

1. FSAR, Chapter 14
2. FSAR, Section 5.8
3. 10 CFR 50, Appendix J, Option B

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.3 (continued)

The Note allows valves and blind flanges located in high radiation areas to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, and 3 for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in their proper position, is small.

SR 3.6.3.4

Verifying that the isolation time of each automatic power operated containment isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analysis. The isolation time and Frequency of this SR are in accordance with the Inservice Testing Program.

SR 3.6.3.5

For containment 8 inch purge exhaust and 12 inch air room supply valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, Option B (Ref. 3), is required to ensure the valves are physically closed (SR 3.6.3.1 verifies the valves are locked closed). Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types. Based on this observation and the importance of maintaining this penetration leak tight (due to the direct path between containment and the environment), a Frequency of 184 days was established as part of the NRC resolution of Generic Issue B-20, "Containment Leakage Due to Seal Deterioration" (Ref. 4) as specified in the Safety Evaluation for Amendment No. 90 to the Facility Operating License.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.6

Automatic containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures each automatic containment isolation valve will actuate to its isolation position on an actual or simulated actuation signal, i.e., CHP or CHR. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency was developed considering it is prudent that this SR be performed only during a plant outage, since isolation of penetrations would eliminate cooling water flow and disrupt normal operation of many critical components. Operating experience has shown that these components usually pass this SR when performed on the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section 5.8
2. FSAR, Section 6.7.2
3. 10 CFR 50, Appendix J, Option B
4. Generic Issue B-20

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.194TO FACILITY OPERATING LICENSE NO. DPR-20

CONSUMERS ENERGY COMPANY

PALISADES PLANT

DOCKET NO. 50-255

1.0 INTRODUCTION

By application dated December 7, 2000, as revised January 12, 2001, Consumers Energy Company (the licensee) requested changes to the Technical Specifications (TS) for the Palisades Plant. The proposed amendment would change the TSs to allow Type B and C (local) containment leak rate testing to be performed in accordance with 10 CFR Part 50, Appendix J, Option B, and referenced Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak Test Program," dated September 1995.[1] RG 1.163 specifies a method acceptable to the NRC for complying with Option B. Converting to Option B affects TS 5.5.14 and TS Surveillance Requirements (SRs) 3.6.1.1, 3.6.1.3, and 3.6.2.1. The amendment would also increase the interval in SR 3.6.2.2 for containment air lock door interlock testing from 18 months to 24 months. The application dated December 7, 2000, superceded in its entirety an earlier application dated July 28, 2000.

The January 12, 2001, letter revised the proposed changes to TS page 5.0-22 to be consistent with Technical Specification Task Force (TSTF) traveler TSTF-52, Revision 3. The changes to TS page 5.0-22 did not change the staff's initial proposed no significant hazards consideration determination.

On September 12, 1995, the U.S. Nuclear Regulatory Commission (NRC) approved issuance of a revision to 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors" which was subsequently published in the *Federal Register* on September 26, 1995, and became effective on October 26, 1995. The NRC added Option B, "Performance-Based Requirements," to allow licensees to voluntarily replace the prescriptive testing requirements of 10 CFR Part 50, Appendix J, with testing requirements based on both overall performance and the performance of individual components.

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[1] The proposed changes relate only to Types B and C "local" leakage rate testing. The NRC approved the use of Option B for Type A "integrated" leakage rate testing on October 31, 1996, by Palisades License Amendment No. 174, dated October 31, 1996.

2.0 BACKGROUND

Compliance with 10 CFR Part 50, Appendix J, provides assurance that the primary containment, including those systems and components that penetrate the primary containment, do not exceed the allowable leakage rate specified in the TSs and Bases. The allowable leakage rate is determined so that the leakage rate assumed in the safety analyses is not exceeded.

On February 4, 1992, the NRC published a notice in the *Federal Register* (57 FR 4166) discussing a planned initiative to begin eliminating requirements that are marginal to safety, but which impose a significant regulatory burden. The regulations of 10 CFR Part 50, Appendix J, were considered for this initiative and the NRC staff initiated a study of possible changes to this regulation. The study examined the previous performance history of domestic containments and examined the effect on risk of a revision to the requirements of Appendix J. The results of this study are reported in NUREG-1493, "Performance-Based Leak-Test Program."

On the basis of the results of this study, the NRC staff developed a performance-based approach to containment leakage rate testing. On September 12, 1995, the NRC approved issuance of this revision to 10 CFR Part 50, Appendix J, which was subsequently published in the *Federal Register* on September 26, 1995, and became effective on October 26, 1995. The revision added Option B, "Performance-Based Requirements," to Appendix J to allow licensees to voluntarily replace the prescriptive testing requirements of Appendix J with testing requirements based upon both overall and individual component leakage rate performance.

RG 1.163 was developed as a method acceptable to the NRC staff for implementing Option B. This RG states that the Nuclear Energy Institute (NEI) guidance document NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," provides methods acceptable to the NRC staff for complying with Option B, with four exceptions, which are described therein.

Option B requires that RG 1.163 or another implementation document used by a licensee to develop a performance-based leakage testing program must be included, by general reference, in the plant TSs. The licensee has referenced RG 1.163 in the proposed Palisades TSs.

RG 1.163 specifies an extension in Type A test frequency to at least one test in 10 years based upon two consecutive successful tests. Type B tests may be extended up to a maximum interval of 10 years based upon completion of two consecutive successful tests. Type C tests may be extended up to 5 years based upon two consecutive successful tests.

By letter dated October 20, 1995, NEI proposed TSs to implement Option B. After some discussion, the NRC staff and NEI agreed on final TSs, which the NRC staff transmitted to NEI in a letter dated November 2, 1995. These TSs served as a model for licensees to develop plant-specific TSs in preparing amendment requests to implement Option B. However, the Standard Technical Specifications (STS) have subsequently been revised in accordance with the TSTF generic change traveler TSTF-52, Revision 3, and this is now used as the standard for TSs related to Option B.

In order for a licensee to determine the performance of each component, factors that are indicative of, or that affect performance (such as an administrative leakage limit) must be established. The administrative limit is selected to be indicative of the potential onset of component degradation. Although these limits are subject to NRC inspection to assure that they are selected in a reasonable manner, they are not TS requirements. Failure to meet an administrative limit requires the licensee to return to the minimum value of the test interval.

Option B requires that the licensee maintain records to show that the criteria for Type A, B, and C tests have been met. In addition, the licensee must maintain comparisons of the performance of the overall containment system and the individual components to show that the test intervals are adequate. These records are subject to NRC inspection.

3.0 EVALUATION

In Amendment No. 174, the Palisades TSs were revised to incorporate the requirements of 10 CFR Part 50, Appendix J, Option B, for the Type A test. The Type B and C tests were still performed under the requirements of Option A. In Amendment No. 189, dated November 30, 1999, the Palisades TSs were converted to a set of improved TSs (ITS) based upon NUREG-1432, "Standard Technical Specifications, Combustion Engineering Plants," Revision 1. The ITS reflected the current requirements for containment leakage rate testing (i.e., Type A tests performed per Option B and Type B and C tests performed per Option A).

In the application for amendment dated December 7, 2000, as revised January 12, 2001, the licensee proposed changes to the Palisades TSs to provide for performing the Type B and C tests using the requirements of Option B of Appendix J. The proposed changes (1) delete SR 3.6.1.3, (2) modify SRs 3.6.1.1 and 3.6.1.2 to reflect that the leakage rate testing is in accordance with the containment leakage rate testing program, and (3) modify TS 5.5.14 to reflect that all containment leakage rate testing is to be done in accordance with Appendix J, Option B, and that the details of the testing are relocated to the containment leakage rate testing program.

On the basis of its review, the NRC staff finds that the TS changes proposed by the licensee are in compliance with the requirements of Option B and are consistent with the guidance of RG 1.163, with three exceptions noted by the licensee. These exceptions are discussed in sections 3.1 and 3.2 below. Further, the NRC staff finds that the proposed TS are consistent with TSTF-52, Revision 3.

Additionally, the licensee has proposed a related TS change which goes beyond the scope of the conversion to Option B. This change is discussed in section 3.3, below.

3.1 Air Lock Testing And Seal Adjustment

The licensee proposes two exceptions from the guidance of RG 1.163 concerning the leakage rate testing of the two containment air locks, especially as related to air lock door seal adjustment. The first (exception 1) concerns the emergency escape air lock, and the second (exception 2) concerns the personnel air lock.

NEI 94-01, which is cited by RG 1.163, states that following maintenance on an air lock pressure-retaining boundary, the licensee shall either perform a leakage rate test of the air lock at a pressure of not less than P_a (peak containment accident pressure), or shall perform a leakage rate test of the affected area or component at P_a.

The licensee's proposed exception 1 allows performance of a seal contact check in lieu of the requirements for additional leakage rate testing following post-test door seal adjustments (or door openings) on the emergency escape air lock. The licensee explains that:

> Leakage rate testing is not necessary after opening the Emergency Escape Air Lock doors for post-test restoration or post-test adjustment of the air lock door seals. However, a seal contact check shall be performed instead.
>
> Emergency Escape Airlock door opening, solely for the purpose of strongback removal and performance of the seal contact check, does not necessitate additional pressure testing.

The NRC staff agrees with the licensee's explanation. This practice has already been approved by the Amendment No. 177, dated September 30, 1997, and by an exemption enclosed with Amendment No. 177 to certain requirements of 10 CFR Part 50, Appendix J, Option A. Thus, there is no need to address the issue again in this evaluation.

The second proposed exception to RG 1.163 is closely related to the first exception. The following extraction from the safety evaluation for Amendment No. 177 provides background and justification for exception 2:

> [Background]
>
> <u>Description of Emergency Escape Airlock</u>
>
> The emergency escape airlock was designed and installed prior to the August 1971 issuance of Appendix J.... The airlock consists of a steel cylinder with circular doors at each end interlocked so that only one door can be open at any time. The airlock is designed to withstand all containment conditions with either door or both doors closed. The doors open towards the interior of containment and the door directly in contact with the containment atmosphere is designated as the inner door.
>
> Double gaskets or seals are provided to seal each door. The seal material currently in use is an ethylene-propylene-diamine-monomer (EPDM), which has been selected because of its combined properties of resistance to radiation, sealing capability, and resistance to high temperatures. The airlock barrel may be pressurized to test its leak tightness without pressurizing the containment building. The escape lock doors each have two latching pins centered at the top and bottom of the door (corresponding to 12 o'clock and 6 o'clock positions).

The emergency escape airlock door latching pins only serve to position the door against the stationary bulkhead. The door's design relies on the increase in containment pressure during a postulated event to provide sufficient closing force to produce an effective seal. The two latching pins by themselves do not provide an adequate circumferential closing force to allow meaningful door seal pressure testing.

Description of Present Surveillance Test

During a design basis accident, the pressure applied to the doors forces them against the seals. During airlock pressure testing, a strongback (structural bracing) is necessary to simulate this pressure on the inner door and to protect the inner door locking pins from the forces generated by the internal test pressure in the barrel. The use of a door strongback to complete between-the-seals testing or full airlock pressure testing (inner door only) is required and was part of the original design of the doors. This design does not permit unrestrained between-the-seals testing.

Past TS surveillance testing for both the personnel airlock and the emergency escape airlock has shown that testing at containment design pressure with strongbacks in place causes the seals to take a set that reflects the shape of the seal grooves. With strongbacks installed or test pressure applied, the male portion of the door seal (the seal bead) will be pressed approximately three-eighths of an inch into the seal. The seal will remain in this compressed condition for a 12-to-24-hour period while the test is being performed, causing the seal to take a set in the seal groove of the airlock bulkhead. After completion of the full pressure test, the doors must be opened to remove the strong back and to verify seal contact with the door seal bead in order to assure that the seals rebound to their pre-test condition. Seal adjustment ("fluffing") may be required after testing because the force of the strongbacks on the inner door and the force due to the test pressure on the outer door draws the seal bead on the doors further into the seal groove than obtained with normal door closure forces.

Past test performances have shown that once the strongbacks are removed, the seals may not completely rebound to their pre-test position. After full pressure testing of the airlock, a seal contact check is performed as part of the surveillance test. If the seal contact check reveals gaps, seal adjustment is performed to ensure that the seal material rebounds to its pre-test condition. The licensee considers seal adjustment a normal part of restoration from testing and it is controlled by procedure.

The seal contact check consists of applying a thin layer of grease on the seal face and then closing and reopening the airlock door. This will result in a pattern in the grease that is representative of the door seal bead mating with the seal. If the grease pattern does not show adequate contact, the seals are adjusted in the area of the gap. This is done by lifting the seals slightly out of their groove so that the seal expands to its pre-test position. Following adjustment, a final seal contact check is performed to verify the integrity of the sealing surface. The practice of verifying acceptable seal contact following performance of the airlock leak test and the acceptance criteria for this verification have been incorporated into the maintenance program. This practice has proven to be effective through successful results during Integrated Leakage Rate Tests

(ILRTs) and 6-month full airlock pressure tests. Similarly, positive results from post-test seal adjustments have also been obtained with the personnel airlock door, although an unrestrained between-the-seals test can be done and therefore is performed on those doors as a final test....

[Staff evaluation]

The TS changes are necessary due to the original design of the emergency escape airlock. During special testing in 1992, the licensee showed that the annulus between the door seals could not be successfully tested without the door strongback installed even at pressures as low as 2 psig. This testing, along with information from the vendor, confirms that between-the-seal pressure testing on the emergency escape airlock doors cannot be properly measured or evaluated if the door strongbacks are not installed. Similarly, the inner door does not fully seal with the reverse-direction pressure of a full airlock pressure test unless the strongback is installed.

Since the removal of the inner door strongback after pressure testing requires the outer door to be opened, a between-the-seals test of the outer door would be required by the [regulation]. This test would require the installation of a strongback on the outer door. Further, full pressure testing or the pressure induced by the strongback may cause the seals to take a set. It is therefore necessary to open both doors (one at a time) after any pressure testing to ensure full seal contact, and there is a potential need to readjust the seals to restore seal contact.

As an alternative to the final between-the-seals pressure test required by [Appendix J] for verification of door seal functionality, the licensee has proposed a final door seal contact verification. This seal performance verification is completed following the full pressure airlock test, after the removal of the inner door strongback, and just prior to final closure of the airlock doors. The requested [exemption] would not affect compliance with the present requirement to perform a full pressure emergency escape airlock test at 6-month intervals. It would also not affect the requirement to perform a full pressure emergency escape airlock test within 72 hours of opening either door during periods when containment integrity is required. The seal contact check replaces the pressure test required by [Appendix J] for the door opening(s) and/or seal adjustments associated with restoration from the required full pressure tests (i.e., the licensee has proposed to continue the practice described above under.... <u>Description of Present Surveillance Test</u>)....

During its review, the staff questioned whether post-test seal adjustment or "fluffing" was necessary because the door seals were too old or worn out to rebound properly to their original shape after leakage rate testing or whether past fluffing had damaged the seals, such that replacement of the seals could result in acceptable between-the-seals testing. The licensee's response, dated February 20, 1997, stated that the seals are replaced approximately every 3 years and that the seals have not exceeded their service lives. Also, they stated that fluffing has not damaged the seals, as indicated by continued successful Type B tests on both the emergency escape airlock and on the personnel airlock, on whose seals "fluffing" is also performed.

> The performance of this door seal contact check has led to the successful completion of subsequent emergency escape airlock full pressure tests since the procedural practice began in 1987. Also, no ILRT in that period has failed because of emergency escape airlock door seal leakage. Based on these results, the airlock doors have been proven to function as designed using current methods of testing and maintenance, including seal contact checks. The seal contact check performed on the emergency escape airlock door seals ensures the doors are sealing properly.
>
> On the basis of the above evaluation, the NRC staff concludes that the licensee's proposal, to perform seal contact testing instead of between-the-seals leak rate testing on the emergency escape airlock door seals under the circumstances described above, is acceptable.

A similar exception is proposed for the personnel air lock, again to avoid entering into an endless cycle of seal adjustment following testing, and testing following seal adjustment. Proposed exception 2 states:

> Leakage rate testing at P_a is not necessary after adjustment of the Personnel Air Lock door seals. However, a between-the-seals test shall be performed at $\geq$10 psig instead.

As stated above, NEI 94-01 requires testing at $\geq P_a$ (53 psig for Palisades) following maintenance on the air lock door seals.

Leakage rate testing of the personnel air lock at an internal pressure of $\geq P_a$ is accomplished by installation of strongbacks on the inner door. The strongbacks simulate accident pressure on the inner door and protect the inner door latching pins from the forces generated by the air lock internal test pressure. Following door openings for strongback removal, the licensee performs an unrestrained (no strongbacks installed) reduced pressure ($\geq$10 psig) between-the-seals test. A full pressure between-the-seals leakage rate test can not be performed without strongbacks installed because the door latching pins and associated mechanism, by themselves, do not provide enough closing force to allow successful unrestrained between-the-seals testing at 53 psig. Therefore, the licensee does not perform between-the-seals testing at 53 psig.

Because testing under Option B requires periodic air lock testing at $\geq P_a$, and air lock design requires seal adjustment following testing at $\geq P_a$, the requirement to perform additional testing at $\geq P_a$ following door seal maintenance results in entering into an endless cycle of seal adjustment following testing, and testing following seal adjustment.

The licensee has routinely performed reduced-pressure between-the-seals testing of the personnel air lock at Palisades since 1987. Since that practice has been in place, no full pressure personnel air lock leakage rate test has failed due to seal leakage.

For Amendment No. 177, the NRC staff accepted the principle of seal fluffing at Palisades as a valid maintenance procedure for restoration of the seals after a full pressure air lock leakage rate test. For the emergency escape air lock, fluffing is followed by a seal contact test; in the case of the personnel air lock, it is followed by a reduced pressure leakage rate test, which is superior to a seal contact test. Further, the requested exception 2 is needed to avoid an infinite

loop of testing. Thus, on the same basis that the NRC staff approved the testing allowed by Amendment No. 177 (now designated as exception 1), the NRC staff now finds exception 2 to Regulatory Guide 1.163 to be acceptable.

3.2 Purge Valve Testing Frequency

The licensee's third proposed exception to RG 1.163 states:

> Leakage rate testing frequency for the Containment 4 inch purge exhaust valves, the 8 inch purge exhaust valves, and the 12 inch air room supply valves may be extended up to 60 months based on component performance.

For most containment isolation valves, NEI 94-01 allows their Type C testing intervals to be extended to 60 months, based on good test performance. However, RG 1.163, in Regulatory Position C.2., specifically prohibits containment purge and vent valves from having their intervals extended. They are constrained to 30 months, which is the normal, unextended test interval. This constraint was imposed for two reasons: 1) the high safety significance of potential leakage through these large valves which provides a direct connection between the containment atmosphere and the outside environment; and 2) the poor historical leakage performance of these valves, which are typically large butterfly valves with resilient seals.

In fact, due to these factors, many plants have TS requirements to perform leakage rate tests on their purge and vent valves much more often than the frequency required by the Type C testing program. In the case of Palisades, current SR 3.6.3.5 requires a leakage rate test of the 8-inch and 12-inch valves every 184 days. The primary purpose of this more frequent leakage rate testing is to ensure that the valves' seals have not degraded.

Although the Type C tests and the "seal degradation" tests are similar, they are not conducted in exactly the same way. The Type C tests are performed during plant shutdown, from inside the containment, by pressurizing each individual valve in the accident direction (i.e., air flow going out of containment) and determining a leakage rate. The performance of the Type C tests requires the installation and removal of two 8-inch test flanges and one 12-inch test flange inside of containment. One scaffold, approximately 40 feet high, is required to install the two 8-inch test flanges. A separate scaffold, approximately 12 feet high, is required to install the 12-inch test flange. These areas are difficult to access. Therefore, this testing is costly in terms of resources and dose, and represents some personnel safety hazards. The licensee states that the direct cost for performing these tests one time is approximately $50,000 for scaffolding (via a contractor) in addition to 85 hours of plant operations and mechanical maintenance personnel time. Radiation exposure is typically about 90 mrem.

On the other hand, the seal degradation tests are performed from outside of containment and do not require scaffolding or test flanges. A test tap between each pair of valves (there are two valves in series in each line) allows the space between the pair of valves to be pressurized, and a leakage rate measured. The test pressure for both kinds of tests is greater than or equal to P_a.

Effectively, the only difference between the two kinds of tests performed on the 8-inch and 12-inch valves are the direction of testing on the innermost containment isolation valves. The Type C test is performed by pressurizing between the tested valve and the test flange inside the containment; the seal degradation tests are performed by pressurizing between the valves. Therefore, the inner valve has test pressure applied in the reverse direction to that which would be applied under accident conditions. This would be allowed for Type C testing, if the reverse-direction testing gave equivalent or conservative results compared to testing in the accident direction.

The valves are designed to seal effectively regardless of direction of flow. The licensee states that they have never experienced evidence of leakage between the valves that would indicate the test results would be different based on direction of applied test pressure. Seat leakage is readily detectable when testing from either direction. Although some designs of butterfly valves with resilient seals are known to have direction-dependent leakage characteristics (e.g., see Information Notice 88-73), these valves are not included.

However, there is one other aspect of testing direction to consider, other than valve seat leakage characteristics. All of the purge and vent valves are located outside of containment. Leakage from the shaft or bonnet seals, or from inner flanges when the valves are joined to the pipes by flanges, is therefore leakage to the environment and should be measured.

Because of the valve orientations, the shaft and bonnet seals on the innermost containment isolation valves are exposed to test pressure when test pressure is applied from between the valves. Thus, reverse-direction testing is acceptable in this respect.

However, the 8-inch and 12-inch valves have flanged connections to their pipes, and the inner flanges of the inner valves are not tested by pressurizing between the valves. Thus, the two kinds of leakage rate tests are not quite equivalent for these valves.

Nonetheless, the NRC staff finds that the seal degradation test is sufficient to allow the Type C test interval to be extended for the 8-inch and 12-inch valves on a performance basis, as with other containment isolation valves. The through-valve leakage is by far the most important kind of leakage for purge and vent valves with resilient seals, and the seal degradation test measures that leakage. The inner pipe flanges, which have flexitallic-type gaskets, are not known to be problematic leak paths. There is no reason to expect them to degrade significantly if put on a 60-month testing interval, which is only done if they first demonstrate good leakage performance. In this case, the NRC staff finds that the safety benefit derived from keeping these valves on a 30-month interval is not significant, whereas the burden reduction for the licensee to be derived from a possible 60-month interval is significant. Therefore, the NRC staff finds proposed exception 3 to be acceptable for the 8-inch and 12-inch purge and vent valves at Palisades.

The two 4-inch valves are gate valves and do not have resilient seals. They are also welded into their pipe. They are only tested by the Type C tests. They are not subjected to more frequent testing precisely because they are not expected to suffer the degradation to which resilient-sealed valves are prone. Also, due to the piping layout, Type C testing of these valves alone would still require scaffolding to be erected and an 8-inch flange to be installed inside

containment. Therefore, due to their lower safety significance (because of smaller size, absence of resilient seals and welded connections), and their inclusion in exception 3 to make its burden reduction available to the licensee, the NRC staff finds that the 4-inch valves are also acceptable for inclusion in exception 3.

In summary, the NRC staff finds proposed exception 3 to RG 1.163 to be acceptable for the Palisades Plant.

3.3 Air Lock Door Interlock Surveillance

The licensee proposes to revise SR 3.6.2.2 to require testing of the air lock door interlocks at an interval of 24 months, rather than the current 18 months. Typically, the interlock is installed after each refueling outage, verified operable with this surveillance, and not disturbed until the next refueling outage. If the need for maintenance arises when the interlock is required, the interlock surveillance would be performed following the maintenance. In addition, when an air lock is opened during times when the interlock is required, the operator first verifies that one door is completely shut before attempting to open the other door. Therefore, the interlock is not challenged except during actual testing of the interlock. Consequently, it should be sufficient to ensure proper operation of the interlock by testing the interlock on a 24-month interval.

Historically, this interlock verification has had its frequency chosen to coincide with the frequency of the overall air lock leakage rate test. However, Appendix J, Option B, allows for an extension of the overall air lock leakage rate test frequency to a maximum of 30 months.

For the above reasons, the licensee proposes to change the required frequency for this surveillance to 24 months (and, with the allowance of SR 3.0.2, this provides a total of 30 months, corresponding to the overall air lock leakage rate test frequency under Option B). In this fashion, the interlock can be tested in a Mode where the interlock is not required. Thus, the NRC staff finds this change to be acceptable.

3.4 Summary

In summary, the NRC staff has reviewed the changes to the TSs proposed by the licensee, for Option B implementation, and finds that the TS changes are in compliance with the requirements of Appendix J, Option B, and are consistent with the guidance of Regulatory Guide 1.163 (other than the three noted exceptions), and are therefore acceptable. Further, on the bases of the above discussions, the NRC staff finds the three exceptions to the guidance of Regulatory Guide 1.163, and the additional change to the air lock door interlock surveillance, discussed in sections 3.1 to 3.3 above, to be acceptable.

This amendment also includes changes to the TS Bases to reflect the changes to TS 5.5.14, SR 3.6.1.1, SR 3.65.1.3, SR 3.6.2.1, and SR 3.6.2.2. The staff does not object to the changes proposed to the TS Bases.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Michigan State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (66 FR 7676). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based upon the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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