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AEP
INDIANA
MICHIGAN
POWER

September 1, 2000

C0900-11
10 CFR 50.12
10 CFR 50.90

Docket Nos.: 50-315
50-316

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, DC 20555-0001

Donald C. Cook Nuclear Plant Units 1 and 2
10 CFR 50, APPENDIX J EXEMPTION REQUEST AND
TECHNICAL SPECIFICATION CHANGE REQUEST
CONTAINMENT LEAKAGE RATE TESTING

Pursuant to 10 CFR 50.12, Indiana Michigan Power Company (I&M), the Licensee for Donald C. Cook Nuclear Plant Units 1 and 2, requests a one-time exemption from certain post-modification testing requirements of Appendix J to 10 CFR 50. The proposed exemption would eliminate the requirement to perform post-modification Type A testing following repairs and modifications of the containment barrier during the current Unit 1 outage.

The proposed exemption also affects a Technical Specification (T/S) surveillance requirement. Therefore, pursuant to 10 CFR 50.90, I&M proposes to amend Appendix A, "Technical Specifications," of Facility Operating License DPR-58. The proposed change affects T/S surveillance requirement 4.6.1.2 and the associated T/S Bases to address exemptions to leakage rate testing specified by 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," and Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. A provision to allow future exemptions from 10 CFR 50, Appendix J, is also proposed for Unit 2 (DPR-74).

Attachment 1 provides a detailed description and safety analysis to support the proposed changes. Attachments 2A and 2B provide marked up T/S pages for Unit 1 and Unit 2, respectively. Attachments 3A and 3B provide the proposed T/S pages with the changes incorporated for Unit 1 and Unit 2, respectively. Attachment 4 describes the evaluation performed in accordance with 10 CFR 50.92(c), which concludes that no significant hazard is involved.

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Attachment 5 describes the justification for the exemption based on the criteria in 10 CFR 50.12. Attachment 6 provides the environmental assessment.

I&M requests approval of this activity by October 30, 2000, to support Unit 1 start-up activities.

No previous submittals affect T/S pages that are submitted in this request. If any future submittals affect these T/S pages, then I&M will coordinate changes to the pages with the Nuclear Regulatory Commission Project Manager to ensure proper T/S page control when the associated license amendment requests are approved.

Copies of this letter and its attachments are being transmitted to the Michigan Public Service Commission and Michigan Department of Environmental Quality, in accordance with the requirements of 10 CFR 50.91.

Should you have any questions, please contact Mr. Wayne J. Kropp, Director of Regulatory Affairs, at (616) 697-5056.

Sincerely,

M. W. Rencheck
Vice President Nuclear Engineering

/dmb

Attachments

c: J. E. Dyer
MDEQ - DW & RPD
NRC Resident Inspector
R. Whale

AFFIRMATION

I, Michael W. Rencheck, being duly sworn, state that I am Vice President of Indiana Michigan Power Company (I&M), that I am authorized to sign and file this request with the Nuclear Regulatory Commission on behalf of I&M, and that the statements made and the matters set forth herein pertaining to I&M are true and correct to the best of my knowledge, information, and belief.

Indiana Michigan Power Company

M. W. Rencheck
Vice President Nuclear Engineering

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 1 DAY OF September, 2000.

Danielle M. Schrader
Notary Public

My Commission Expires 4-4-2004

DANIELLE M. SCHRADER
Notary Public, Berrien County, MI
My Commission Expires Apr 4, 2004

ATTACHMENT 1 TO C0900-11

DESCRIPTION AND SAFETY ANALYSIS FOR THE PROPOSED CHANGES

A. Summary of the Proposed Changes

Indiana Michigan Power Company (I&M), the Licensee for Donald C. Cook Nuclear Plant Units 1 and 2, requests a one-time exemption from certain post-modification testing requirements of Appendix J to 10 CFR 50. The proposed exemption would eliminate the requirement to perform post-modification Type A testing following repairs and modifications of the containment barrier during the current Unit 1 outage.

The proposed exemption also affects a Technical Specification (T/S) surveillance requirement. Therefore, pursuant to 10 CFR 50.90, I&M proposes to amend Appendix A, "Technical Specifications," of Facility Operating License DPR-58. The proposed change affects T/S surveillance requirement 4.6.1.2 and the associated T/S Bases to address exemptions to leakage rate testing specified by 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," and Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. A provision to allow future exemptions from 10 CFR 50, Appendix J, is also proposed for Unit 2 (DPR-74).

The proposed T/S changes are described in detail in Section E of this attachment. T/S pages that are marked to show the proposed changes are provided in Attachments 2A and 2B for Unit 1 and Unit 2, respectively. The proposed T/S pages, with the changes incorporated, are provided in Attachments 3A and 3B for Unit 1 and Unit 2, respectively. The justification for the exemption request is provided separately in Attachment 5.

B. Description of the Current Requirements

T/S surveillance requirement 4.6.1.2 requires leakage rate testing in accordance with 10 CFR 50 Appendix J, Option B and RG 1.163. RG 1.163 endorses Nuclear Energy Institute (NEI) document NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," which provides methods acceptable to the Nuclear Regulatory Commission (NRC) for complying with the provisions of 10 CFR 50, Appendix J.

10 CFR 50, Appendix J, states that any major modification or replacement of a component that is part of the primary reactor containment boundary shall be followed by a Type A, Type B, or Type C test, as applicable, for the area affected by the modification. This requirement is also contained in Section 9.2.4 of NEI document NEI 94-10, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J." NEI 94-01 is endorsed by RG 1.163 as an acceptable method of complying with Option B in Appendix J to 10 CFR 50. Type A tests are intended to measure the primary reactor containment overall integrated leakage rate. "Primary reactor containment" is defined as the structure or vessel that encloses the

components of the reactor coolant pressure boundary and serves as an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment. The steam generator shell and the main steam, feedwater, steam generator blowdown/drain, and steam generator sampling systems are considered an extension of the primary reactor containment liner per section 5.4 of the Updated Final Safety Analysis Report (UFSAR). Therefore, certain maintenance or major modification to certain closed piping requires a leakage rate test.

C. Bases for the Current Requirements

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure. Primary containment integrity ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leakage rates assumed in the accident analyses. These restrictions will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions.

D. Need for Revision of the Requirement

The original Unit 1 Westinghouse Model 51 steam generators were replaced with Babcock & Wilcox Model 51R steam generators during the current outage. This replacement involved welding the original steam domes, which were refurbished with modern moisture separation hardware, to the new lower assemblies. In addition, fluid lines associated with the steam generator, including the main steam, feedwater, and blowdown lines, were severed and re-welded to support the replacement. Modifications and repairs not related to the steam generator replacement were also completed on these systems. These actions qualify as maintenance/modifications that affect containment leakage integrity as defined in NEI 94-01 and 10 CFR 50, Appendix J. Therefore, a Type A test to ensure containment integrity is required by T/S.

Type A testing is labor-intensive and costly. Industry figures show that the average cost for a Type A test is approximately $1.6 million, and the testing typically requires four additional days of outage critical path time. In-service testing has been shown to demonstrate leak-tightness of piping systems. The in-service test provides an alternate method which allows leakage testing of only the modified portions of the containment barrier (steam generator shell and associated closed piping) instead of the more labor-intensive Type A test which would be performed on the entire containment barrier. Crediting the performance of in-service testing would avoid the occupational dose associated with the setup and performance of a Type A test.

The use of in-service testing in place of Type A leakage rate testing is not consistent with the current T/S surveillance requirement 4.6.1.2. Additionally, the proposed change would accomplish leakage testing of the modified portions of the containment barrier in Mode 3, in contrast to the current requirement to complete testing prior to entering Mode 4. ASME

Section XI testing requires in-service leakage testing to be performed at approximately normal operating temperature and pressure, which are not reached in Mode 4. The proposed change is needed to accommodate the current Unit 1 outage activities.

E. Description of the Proposed Changes

I&M requests a one-time exemption from the requirements of 10 CFR 50, Appendix J, to allow crediting the performance of ASME Section XI in-service testing per IWA-5000, "System Pressure Tests," in lieu of Type A testing per Appendix J, Option B and RG 1.163. This alternate test method is proposed for the closed piping portions of the containment barrier inside containment after repair or modification, including the steam generator blowdown/drain piping, main steam piping, feedwater piping, steam generator sampling piping, and steam generator shell. ASME Section XI in-service pressure testing allows zero through-wall leakage of pressure boundary components. Leakage at bolted connections or packing is not considered through-wall leakage and would not result in the rejection of the test, nor would ASME Section XI require an evaluation of this leakage. 10 CFR 50, Appendix J, leak testing considers all leakage in its calculations, including packing and gasketed joints. The proposed testing would be performed in Mode 3. I&M also proposes to revise the T/S Bases to reflect the revised requirement.

F. Bases for the Proposed Changes

The proposed change to credit the alternate test method continues to demonstrate that leakage associated with the modified equipment is acceptably low. The Type A test measures overall containment leakage. NUREG-1493, "Performance-Based Containment Leak-Test Program," demonstrates that Type B and C testing detect a very large percentage of containment leaks. The proposed test is similar to a Type B test in that it is a local leakage rate test where pressure is applied on the inside of the closed piping/system. The in-service test is expected to detect any leaks that exist in the closed piping portion of the containment barrier.

The steam generator and associated piping can be demonstrated to be leak-tight by either pressurizing the containment with air to 27 psia or pressurizing the shell side of the steam generator with steam/water to 1020 psia. Leakage can be readily identified with a high differential pressure. A leakage path between the secondary side of the steam generator or associated piping and the containment atmosphere is susceptible to leakage in either direction. Therefore, leakage out of the steam generator should be equal to leakage into the steam generator. The leakage direction is dependent on where the higher pressure exists. The leakage rate is dependent on the differential pressure across a given leakage path.

The proposed test, unlike the Type A test, does not require the leakage rate to be quantified. The acceptance criterion for the proposed test is no visual through-wall leakage; therefore, there is no need to quantify the leakage rate. This acceptance criterion is more conservative than the Type A test, which allows some leakage.

ASME Section XI requires non-destructive examination (NDE) and visual examination of welds and system leakage testing. If any through-wall leakage is detected from the welds, the leakage is required to be repaired before plant service continues. NDE of the welds (ultrasonic or radiographic testing) provides assurance that the joints are free of flaws that could result in significant leakage. This NDE provides the confidence to pressurize the secondary side of the steam generators and demonstrate leak-tight integrity with the unit in Mode 3 under no-load plant conditions.

The proposed in-service testing would demonstrate the leak-tight integrity of the steam generator and associated piping portions of the containment boundary to ensure that the leakage volume will not exceed that assumed in the accident analysis. The bounding accident is the large break loss-of-coolant accident (LOCA), which is described in sections 14.3.4.3 and 14.3.5 of the UFSAR. This accident results in the highest containment pressure and the greatest potential for fission product release through any containment barrier leaks. The accident analyses are unaffected because containment leakage limits are not changed.

The proposed change for the post-modification testing would allow entry into Modes 4 and 3 before the testing is complete because the testing is conducted at normal operating conditions. Entering Modes 3 and 4 prior to quantifying the containment leakage rate is acceptable because in order to have a release through the modified closed piping systems, there would need to be a LOCA concurrent with a through-wall leak, with enough pressure in containment to overcome main steam system pressure. These conditions are unlikely to occur simultaneously in Modes 3 and 4.

The proposed changes to the T/S Bases are consistent with the proposed changes to the T/S. The addition of references to NEI 94-01 and RG 1.163 is consistent with Amendments 209 and 193 to DPR-58 and DPR-74, respectively.

The NRC staff acknowledged in an internal memorandum (Reference 1) that a closed piping system does not fit readily into the definitions of 10 CFR 50, Appendix J. The memo stated that after modifications to a closed system, a local leakage rate test may be more appropriate than a Type A test, and pressurizing the pipe with air or water and assuring no leaks are detected may be appropriate testing. The objective is to assure the leak-tight integrity of the system, which was also stated in the NRC memo. This approach, where a test pressure is applied in the opposite direction as would occur during the design basis accident, is acceptable per the recommendations prescribed in ANSI/ANS-56.8-1994, section 6.2 and NEI 94-01, section 9.2.1, if equivalent or more conservative test results are provided. Both documents are endorsed by RG 1.163.

A number of utilities have used in-service testing as an alternative to Appendix J pressure testing following steam generator replacement to demonstrate its leak-tight integrity as a component of the containment barrier. Other utilities have opted for an in-service test in lieu of Type A testing

following secondary boundary maintenance/modification inside containment. For example, the licensee for North Anna Power Station, Unit 2, received NRC approval (Reference 2) to take credit for ASME Section XI pressure testing in lieu of Type A testing.

G. Discussion of Risk

The in-service test is performed in Mode 3 while the Type A testing is performed before entry into Mode 4. The difference between the tests is the pressure and temperature of the reactor coolant system during testing. A flaw that is not identified by the post-modification NDE could introduce a leakage path that is identified only after the applicable systems are pressurized (Mode 3). Such a leakage path is expected to be small because NDE is capable of revealing very small flaws. If a through-wall leak is detected, repairs are effected before plant service continues. The probability of a Mode 3 containment leak would be greater with the proposed test than with Type A testing because, with Type A testing, containment integrity is verified before Mode 4. However, the lower containment pressure and fission product levels associated with a LOCA prior to or during testing, together with the small leakage rate allowed by a weld joint flaw, would result in lower consequences of a LOCA during testing relative to a LOCA at full power. Therefore, there is no discernable increase in risk by performing the proposed in-service test on the subject closed system in Mode 3 in lieu of a Type A test.

The probability of a LOCA is lower prior to or during testing than the probability of a LOCA in Mode 1. Less time is spent in Modes 3 and 4 than in Mode 1, and the consequences of the LOCA are lower due to the reduced coolant activity. Under the proposed change, there could be a small increase in the probability of a release in the event of a LOCA before the in-service testing is completed; however, that is more than offset by the lower consequences relative to the consequences of a design basis accident. Therefore, there is no significant increase in risk.

H. Impact on Previous Submittals

No previous submittals affect T/S pages that are submitted in this request. If any future submittals affect these T/S pages, then I&M will coordinate changes to the pages with the NRC Project Manager to ensure proper T/S page control when the associated license amendment requests are approved.

I. References

1. Memorandum from G. M. Holohan (NRR) to J. A. Zwolinski (NRC), "Region III Request for Position on Leakage out of Containment through PWR Steam Generator Secondary Side During Containment Integrated Leakage Rate Test," dated February 1, 1991.

2. Letter from D. B. Matthews (NRC) to J. P. O'Hanlon (VEPCO), "Issuance of Exemptions from the Requirements of 10 CFR 50, Appendix J Associated with Type A Testing Requirements – North Anna Power Station, Unit 2," NRC Docket #50-339, dated March 29, 1995.

ATTACHMENT 2A TO C0900-11

TECHNICAL SPECIFICATIONS PAGES
MARKED TO SHOW PROPOSED CHANGES

REVISED PAGES
UNIT 1

3/4 6-2

B 3/4 6-1

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of $\leq L_a$, 0.25 percent by weight of the containment air per 24 hours at P_a, 12.0 psig, and

b. A combined leakage rate of $\leq 0.60\ L_a$ for all penetrations and valves subject to Types B and C tests when pressurized to P_a.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With either (a) the measured overall integrated containment leakage rate exceeding $0.75\ L_a$, or (b) with the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding $0.60\ L_a$, restore the overall integrated leakage rate to $\leq 0.75\ L_a$ and the combined leakage rate for all penetrations and valves subject to Types B and C tests to $\leq 0.60\ L_a$ prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 Perform leakage rate testing in accordance with 10 CFR 50 Appendix J Option B, except as modified by NRC-approved exemptions, and Regulatory Guide 1.163, dated September 1995.

a. Each containment air lock shall be verified to be in compliance with the requirements of Specification 3.6.1.3.

b. The provisions of Specification 4.0.2 are not applicable.

3/4.6.1 PRIMARY CONTAINMENT

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analysis. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions.

3/4 6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analysis at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to $\leq 0.75\ L_a$ during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix "J" of 10 CFR 50, Regulatory Guide 1.163, dated September 1995, and Nuclear Energy Institute document NEI 94-01, except as modified.

A one-time exemption from post-modification Type A testing is allowed for the steam generator and associated piping, as components of the containment barrier, for entry into MODES 4 and 3 following the extended outage that commenced in 1997. For this case, ASME Section XI in-service testing may be used to verify the leak tightness of the repaired or modified portions of the containment barrier.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provide assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

ATTACHMENT 2B TO C0900-11

TECHNICAL SPECIFICATIONS PAGES
MARKED TO SHOW PROPOSED CHANGES

REVISED PAGES
UNIT 2

3/4 6-2

B 3/4 6-1

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of $\leq L_a$, 0.25 percent by weight of the containment air per 24 hours at P_a, 12 psig.

b. A combined leakage rate of $\leq 0.60\ L_a$ for all penetrations and valves subject to Types B and C tests when pressurized to P_a.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With either (a) the measured overall integrated containment leakage rate exceeding $0.75\ L_a$, or (b) with the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding $0.60\ L_a$, restore the overall integrated leakage rate to $\leq 0.75\ L_a$ and the combined leakage rate for all penetrations and valves subject to Types B and C tests to $\leq 0.60\ L_a$ prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 Perform leakage rate testing in accordance with 10 CFR 50 Appendix J Option B, except as modified by NRC-approved exemptions, and Regulatory Guide 1.163, dated September 1995.

a. Each containment air lock shall be verified to be in compliance with the requirements of Specification 3.6.1.3.

b. The provisions of Specification 4.0.2 are not applicable.

3/4.6.1 PRIMARY CONTAINMENT

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to $\leq 0.75\ L_a$, during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix "J" of 10 CFR 50, Regulatory Guide 1.163, dated September 1995, and Nuclear Energy Institute document NEI 94-01, except as modified.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provide assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

ATTACHMENT 3A TO C0900-11

PROPOSED TECHNICAL SPECIFICATIONS PAGES

REVISED PAGES
UNIT 1

3/4 6-2

B 3/4 6-1

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of $\leq L_a$, 0.25 percent by weight of the containment air per 24 hours at P_a, 12.0 psig, and

b. A combined leakage rate of $\leq 0.60\ L_a$ for all penetrations and valves subject to Types B and C tests when pressurized to P_a.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With either (a) the measured overall integrated containment leakage rate exceeding $0.75\ L_a$, or (b) with the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding $0.60\ L_a$, restore the overall integrated leakage rate to $\leq 0.75\ L_a$ and the combined leakage rate for all penetrations and valves subject to Types B and C tests to $\leq 0.60\ L_a$ prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 Perform leakage rate testing in accordance with 10 CFR 50 Appendix J Option B, except as modified by NRC-approved exemptions, and Regulatory Guide 1.163, dated September 1995.

a. Each containment air lock shall be verified to be in compliance with the requirements of Specification 3.6.1.3.

b. The provisions of Specification 4.0.2 are not applicable.

3/4.6.1 PRIMARY CONTAINMENT

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analysis. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions.

3/4 6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analysis at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to $\leq 0.75\ L_a$ during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix "J" of 10 CFR 50, Regulatory Guide 1.163, dated September 1995, and Nuclear Energy Institute document NEI 94-01, except as modified

A one-time exemption from post-modification Type A testing is allowed for the steam generator and associated piping, as components of the containment barrier, for entry into MODES 4 and 3 following the extended outage that commenced in 1997. For this case, ASME Section XI in-service testing may be used to verify the leak tightness of the repaired or modified portions of the containment barrier.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provide assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

ATTACHMENT 3B TO C0900-11

PROPOSED TECHNICAL SPECIFICATIONS PAGES

REVISED PAGES
UNIT 2

3/4 6-2

B 3/4 6-1

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of $\leq L_a$, 0.25 percent by weight of the containment air per 24 hours at P_a, 12 psig.

b. A combined leakage rate of $\leq 0.60\ L_a$ for all penetrations and valves subject to Types B and C tests when pressurized to P_a.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With either (a) the measured overall integrated containment leakage rate exceeding 0.75 L_a, or (b) with the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 L_a, restore the overall integrated leakage rate to $\leq 0.75\ L_a$ and the combined leakage rate for all penetrations and valves subject to Types B and C tests to $\leq 0.60\ L_a$ prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 Perform leakage rate testing in accordance with 10 CFR 50 Appendix J Option B, except as modified by NRC-approved exemptions, and Regulatory Guide 1.163, dated September 1995.

a. Each containment air lock shall be verified to be in compliance with the requirements of Specification 3.6.1.3.

b. The provisions of Specification 4.0.2 are not applicable.

3/4.6.1 PRIMARY CONTAINMENT

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to $\leq 0.75\ L_a$, during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix "J" of 10 CFR 50, Regulatory Guide 1.163, dated September 1995, and Nuclear Energy Institute document NEI 94-01, except as modified.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provide assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

ATTACHMENT 4 TO C0900-11

NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

Indiana Michigan Power Company (I&M) has evaluated this proposed amendment and determined that it does not involve a significant hazard. According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

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

2. create the possibility of a new or different kind of accident from any previously analyzed; or

3. involve a significant reduction in a margin of safety.

The proposed change would revise Technical Specification (T/S) surveillance requirement 4.6.1.2 and the associated T/S Bases to allow crediting the performance of ASME Section XI in-service testing in lieu of the post-modification Type A leakage testing specified by 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," and Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The determination that the criteria set forth in 10 CFR 50.92 are met for this amendment request is indicated below.

1. Does the change involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated?

The proposed in-service testing does not affect accident initiators or precursors. The proposed ASME Section XI in-service test is routinely performed to collect data while the plant is in Mode 3. Conducting the containment leakage test in Mode 3 rather than prior to Mode 4 entry does not affect the probability of an accident. Excessive containment leakage is not a factor until after an accident has already occurred.

The proposed in-service testing does not affect the containment leakage rate limits. Therefore, the consequences of an accident are unchanged. The proposed change to conduct the testing in Mode 3 would not significantly increase the consequences of an accident. In order to have a release through the modified closed piping systems, there would need to be a loss-of-coolant accident concurrent with a through-wall leak, with enough pressure in containment to overcome main steam system pressure. These conditions are extremely unlikely to occur simultaneously in Modes 3 and 4.

Therefore, the probability of occurrence or the consequences of accidents previously evaluated are not increased.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed changes do not introduce any additional physical interface with plant equipment. Therefore, the proposed changes do not degrade the reliability of systems, structures, or components or create a new accident initiator or precursor. No new failure modes are created. The proposed changes demonstrate the leak-tight integrity of the affected portions of the containment barrier through the performance of a visual inspection for through-wall leakage.

Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

T/S 3.6.1.2 is based on limiting total containment leakage volume to the value assumed in the accident analysis at peak accident pressure. The proposed change does not change the allowable leakage rates.

Since the in-service test is performed at a significantly higher pressure than the Type A test and the in-service test acceptance criterion is zero through-wall leakage, versus a nominal amount allowed for the Type A test, the margin of safety will not be reduced. The proposed change would demonstrate the leak-tight integrity of the steam generator and associated piping, as components of the containment barrier, in a fashion at least as rigorous as the Type A test. If the leakage from containment is maintained within the T/S limit, dose rates at the site boundary will not be increased.

Therefore, the proposed activity does not involve a significant reduction in a margin of safety.

In summary, based upon the above evaluation, I&M has concluded that the proposed amendment involves no significant hazards consideration.

ATTACHMENT 5 TO C0900-11

JUSTIFICATION FOR EXEMPTION

Exemption from the requirements of 10 CFR 50, Appendix J to perform Type A testing following the Unit 1 steam generator replacement is justified in accordance with 10 CFR 50.12, "Specific exemptions," Sections (a)(1) and (a)(2)(ii), as follows:

50.12 (a)(1) The exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security.

The proposed exemption is authorized by law and does not have any effect on the common defense and security.

The significant hazards consideration, per 10 CFR 50.92(c), performed in Attachment 4, provides justification that there will not be undue risk to the public health and safety. The reliability of Type B and C test results, the history of Type A test results, and the performance of non-destructive examination and in-service testing on the modified portions of the containment barrier provide assurance that serious deterioration of containment integrity will not occur prior to the next scheduled Type A test.

50.12 (a)(2)(ii) Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule.

The underlying purpose of Appendix J leakage rate testing is to ensure that containment leakage rates remain within allowable limits. Past Type A, B, and C leakage rate tests indicate that the anticipated total integrated leakage rate of the containment is much less than the Appendix J acceptance criteria. The purpose of post-modification testing per Appendix J is to verify the leak tightness of the modified portions of the containment barrier. Under the proposed exemption, that verification will be accomplished by the American Society of Mechanical Engineers (ASME) Section XI in-service testing, eliminating the need for additional Type A leakage rate testing.

Section 9.2.4 of Nuclear Energy Institute (NEI) document NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," which is incorporated by reference in 10 CFR 50, Appendix J, Option B, states "repairs and modifications that affect the containment leakage integrity require leakage rate testing prior to returning the containment to operation." The Unit 1 steam generator replacement and other modifications and repairs to closed piping system portions of the containment barrier inside containment fall under the scope of this requirement. However, in this case, the affected area of the primary containment boundary is also part of the pressure boundary of an ASME Class 2 component/piping system. Therefore, the repairs and modification are subject to the repair and replacement requirements of ASME Section XI. The proposed exemption will allow crediting the ASME Section XI in-service testing for verification of leak tightness of the modified portions

of the containment barrier prior to entry into Mode 2 in lieu of a Type A test. The ASME Section XI surface examination, volumetric examination, and system pressure testing requirements more than fulfill the intent of the requirements of Appendix J.

Performing Type A testing is not necessary to achieve the underlying purpose of the rule. Donald C. Cook Nuclear Plant (CNP) history indicates no Type A test failures (as-found or as-left) have occurred and that the overall calculated leak rates are well within the established values of 10 CFR 50, Appendix J, and CNP Technical Specifications. In-service testing is expected to reveal any leakage of the modified portion of the containment barrier that would have been detected by the Type A test. The next Type A test is scheduled in 2002, in accordance with the requirements of 10 CFR 50, Appendix J, Option B.

ATTACHMENT 6 TO C0900-11

ENVIRONMENTAL ASSESSMENT

Indiana Michigan Power Company (I&M) has evaluated this license amendment request against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. I&M has determined that this license amendment request meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50 that changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or that changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria.

(i) The amendment involves no significant hazards consideration.

As demonstrated in Attachment 4, this proposed amendment does not involve significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

There will be no significant change in the types or significant increase in the amounts of any effluents released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed changes will not result in significant changes in the operation or configuration of the facility. There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposal result in any change in the normal radiation levels within the plant. Therefore, there will be no significant increase in individual or cumulative occupational radiation exposure resulting from this change.