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AEP-NRC-2022-05
10 CFR 50.90

Docket Nos.: 50-315
50-316

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Donald C. Cook Nuclear Plant, Unit 1 and Unit 2
Application to Revise Technical Specifications to Adopt TSTF-554, Revision 1,"Revise Reactor Coolant Leakage Requirements"

Pursuant to 10 CFR 50.90, Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant (CNP), is submitting a request for an amendment to the Technical Specifications (TS) for CNP Unit 1 and Unit 2.

I&M requests adoption of TSTF-554, Revision 1, "Revise Reactor Coolant Leakage Requirements," which is an approved change to the Standard Technical Specifications into the CNP Unit 1 and Unit 2 TS. The proposed amendment revises the TS definition of "Leakage," clarifies the requirements when pressure boundary leakage is detected, and adds a Required Action when pressure boundary leakage is identified. The change is requested as part of the Consolidated Line Item Improvement Process.

Enclosure 1 to this letter provides an affirmation statement pertaining to the information contained herein. Enclosure 2 provides a description and assessment of the proposed changes. Enclosures 3 and 4 provide Unit 1 and Unit 2 TS pages, respectively, marked to show the proposed changes. New clean Unit 1 and Unit 2 TS pages with proposed changes incorporated will be provided to the Nuclear Regulatory Commission (NRC) Licensing Project Manager when requested.

Enclosures 5 and 6 to this letter provide existing Unit 1 and Unit 2 TS Bases pages, respectively, marked to reflect the proposed changes. TS Bases markups are included for information only. Changes to the existing TS Bases, consistent with the technical and regulatory analyses, will be implemented under TS 5.5.12, "Technical Specifications Bases Control Program."

Approval of the proposed amendment is requested in accordance with the normal NRC review schedule for such changes. Once approved, the amendment shall be implemented within 60 days.

There are no regulatory commitments made in this submittal.

In accordance with 10 CFR 50.91, a copy of this application, with enclosures, is being provided to the designated Michigan state officials.

If you should have any questions regarding this submittal, please contact Mr. Michael K. Scarpello, Regulatory Affairs Director, at (269) 466-2649.

Sincerely,

Q. Shane Lies
Site Vice President
Indiana Michigan Power Company

BMC/kmh

Enclosures:

1. Affirmation
2. Description and Assessment of Technical Specification Changes
3. Donald C. Cook Nuclear Plant Unit 1 Technical Specification Pages Marked to Show Proposed Changes
4. Donald C. Cook Nuclear Plant Unit 2 Technical Specification Pages Marked to Show Proposed Changes
5. Donald C. Cook Nuclear Plant Unit 1 Technical Specification Bases Pages Marked to Show Proposed Changes (For Information Only)
6. Donald C. Cook Nuclear Plant Unit 2 Technical Specification Bases Pages Marked to Show Proposed Changes (For Information Only)

c: R. J. Ancona – MPSC
EGLE – RMD/RPS
J. B. Giessner – NRC Region III
NRC Resident Inspector
R. M. Sistevaris – AEP Ft. Wayne, w/o enclosures
J. E. Walcutt – AEP Ft. Wayne, w/o enclosures
S. P. Wall – NRC Washington, D.C.
A. J. Williamson – AEP Ft. Wayne, w/o enclosures

Enclosure 1 to AEP-NRC-2022-05

AFFIRMATION

I, Q. Shane Lies, being duly sworn, state that I am the Site Vice President of Indiana Michigan Power Company (I&M), that I am authorized to sign and file this request with the U. S. 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

Quinton S. Lies

Q. Shane Lies
Site Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 7 DAY OF April 2022

Regan D. Weyl
Notary Public

My Commission Expires 01/21/2025

Enclosure 2 to AEP-NRC-2022-05

Description and Assessment of Technical Specification Changes

1.0 DESCRIPTION

Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant (CNP) Unit 1 and Unit 2, requests adoption of TSTF-554, Revision 1, "Revise Reactor Coolant Leakage Requirements," which is an approved change to the Standard Technical Specifications (STS), into the CNP Unit 1 and Unit 2 Technical Specifications (TS). The proposed amendment revises the TS definition of "Leakage" and the Reactor Coolant System (RCS) Operational Leakage TS to clarify the requirements.

2.0 ASSESSMENT

2.1 Applicability of Safety Evaluation

I&M has reviewed the safety evaluation for TSTF-554, Revision 1, provided to the Technical Specifications Task Force in a letter dated December 18, 2020. This review included a review of the Nuclear Regulatory Commission (NRC) staff's evaluation, as well as the information provided in TSTF-554, Revision 1. As described herein, I&M has concluded that the justifications presented in TSTF-554, Revision 1, and the safety evaluation prepared by the NRC staff are applicable to CNP Unit 1 and Unit 2, and justify this amendment for the incorporation of the changes into the CNP Unit 1 and Unit 2 TS.

2.2 Optional Changes and Variations

I&M is proposing the following variations from the TS changes described in TSTF-554, Revision 1, or the applicable parts of the NRC staff's safety evaluation.

The CNP Unit 1 and Unit 2 TS contain requirements that differ from the STS on which TSTF-554, Revision 1, was based, but are encompassed in the TSTF-554, Revision 1, justification. CNP Unit 1 and Unit 2 TS LCO 3.4.13, RCS Operational LEAKAGE, limits unidentified LEAKAGE to 0.8 gpm rather than the 1 gpm allowed by the STS. This more conservative limitation on unidentified LEAKAGE does not affect the applicability of TSTF-554, Revision 1, to the CNP Unit 1 and Unit 2 TS.

The physical configuration of the CNP Unit 1 and Unit 2 RCS differ slightly from the RCS description presented in Section 2.1 of the NRC staff's safety evaluation. The NRC staff's safety evaluation, Section 2.1, Reactor Coolant System Description, states that the containment sump instrumentation displays results in the main control room. CNP Unit 1 and Unit 2 containment sump instrumentation displays information via pump run time meters located on the containment auxiliaries subpanel in the auxiliary building. This information was most recently reviewed by NRC staff as part of a safety evaluation issued on January 24, 2020, as Agencywide Documents Access and Management System Accession No. ML19329A011 and does not affect the applicability of TSTF-554, Revision 1, to the CNP Unit 1 and Unit 2 TS.

The affected portions of the CNP Unit 1 and Unit 2 TS have some editorial differences from the corresponding STS on which TSTF-554, Revision 1, was based. These editorial differences, described below, are administrative in nature and do not affect the applicability of TSTF-554, Revision 1, to the CNP Unit 1 and Unit 2 TS.

Items a.1 and a.2 under "Identified LEAKAGE" end with a comma rather than a semicolon as shown in TSTF-554, Revision 1. I&M is replacing the comma at the end of items a.1 and a.2 under "Identified LEAKAGE" with a semicolon with this LAR for editorial consistency with the TSTF-554, Revision 1, markup, and to be consistent with the guidance in TSTF-GG-05-01, "Writer's Guide for Plant-Specific Improved Technical Specifications," Revision 1 (August 2010).

Item a.3 under "Identified LEAKAGE" already ends with a semicolon, so the markups in this license amendment request editorially differ from TSTF-554, Revision 1, which shows the punctuation at the end of Item a.3 under "Identified LEAKAGE" changing from a period to a semicolon.

3.0 REGULATORY ANALYSIS

3.1 No Significant Hazards Consideration Analysis

Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant (CNP) Unit 1 and Unit 2, requests adoption of TSTF-554, Revision 1, "Revise Reactor Coolant Leakage Requirements," that is an approved change to the Standard Technical Specifications, into the CNP Unit 1 and Unit 2 Technical Specifications (TS). The proposed amendment revises the TS definition of "Leakage," clarifies the requirements when pressure boundary leakage is detected, and adds a Required Action when pressure boundary leakage is identified.

I&M has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. *Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?*

   Response: No

   The proposed amendment revises the TS definition of "Leakage," clarifies the requirements when pressure boundary leakage is detected, and adds a Required Action when pressure boundary leakage is identified.

   The proposed change revises the definition of pressure boundary leakage. Pressure boundary leakage is a precursor to some accidents previously evaluated. The proposed change expands the definition of pressure boundary leakage by eliminating the qualification that pressure boundary leakage must be from a "nonisolable" flaw. A new TS Action is created which requires isolation of the pressure boundary flaw from the Reactor Coolant System. This new action provides assurance that the flaw will not result in any accident previously evaluated.

Pressure boundary leakage, and the actions taken when pressure boundary leakage is detected, is not assumed in the mitigation of any accident previously evaluated. As a result, the consequences of any accident previously evaluated are unaffected.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No

The proposed amendment revises the TS definition of "Leakage," clarifies the requirements when pressure boundary leakage is detected, and adds a Required Action when pressure boundary leakage is identified. The proposed change does not alter the design function or operation of the RCS. The proposed change does not alter the ability of the RCS to perform its design function. Since pressure boundary leakage is an evaluated accident, the proposed change does not create any new failure mechanisms, malfunctions, or accident initiators not considered in the design and licensing bases.

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

3. *Does the proposed amendment involve a significant reduction in a margin of safety?*

Response: No

The proposed amendment revises the TS definition of "Leakage," clarifies the requirements when pressure boundary leakage is detected, and adds a Required Action when pressure boundary leakage is identified. The proposed change does not affect the initial assumptions, margins, or controlling values used in any accident analysis. The amount of leakage allowed from the RCS is not increased. The proposed change does not affect any design basis or safety limit or any Limiting Condition for Operation.

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

Based on the above, I&M concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

3.2 Conclusion

In conclusion, based on the considerations discussed above, (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.

4.0 ENVIRONMENTAL EVALUATION

The proposed change would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed change.

Enclosure 3 to AEP-NRC-2022-05

Donald C. Cook Nuclear Plant Unit 1 Technical Specification Pages Marked to Show Proposed Changes

1.1 Definitions

	dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil."
ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME	The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC, or the components have been evaluated in accordance with an NRC approved methodology.
INSERVICE TESTING PROGRAM	The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).
LEAKAGE	LEAKAGE shall be: a. <u>Identified LEAKAGE</u> 1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank[;] ~~,~~ 2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known ~~either~~ to not ~~to~~ interfere with the operation of leakage detection systems ~~or not to be pressure boundary LEAKAGE,~~[;] or 3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE); b. <u>Unidentified LEAKAGE</u> All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE; and

1.1 Definitions

	c. Pressure Boundary LEAKAGE LEAKAGE (except primary to secondary LEAKAGE) through a ~~nonisolable~~ fault in an RCS component body, pipe wall, or vessel wall. LEAKAGE past seals, packing, and gaskets is not pressure boundary LEAKAGE.
MASTER RELAY TEST	A MASTER RELAY TEST shall consist of energizing all master relays in the channel required for channel OPERABILITY and verifying the OPERABILITY of each required master relay. The MASTER RELAY TEST shall include a continuity check of each associated required slave relay. The MASTER RELAY TEST may be performed by means of any series of sequential, overlapping, or total steps.
MODE	A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE – OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter 13, Initial Tests and Operation, of the UFSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 0.8 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE; and

d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressure boundary LEAKAGE exists	A.1 Isolate affected component, pipe, or vessel from the RCS by use of a closed manual valve, closed and de-activated automatic valve, blind flange, or check valve.	4 hours
B~~A~~. RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE.	B~~A~~.1 Reduce LEAKAGE to within limits.	4 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C~~B~~. Required Action and associated Completion Time ~~of Condition A~~ not met. ~~OR~~ ~~Pressure boundary LEAKAGE exists.~~ OR Primary to secondary LEAKAGE not within limit.	C~~B~~.1 Be in Mode 3. AND C~~B~~.2 Be in Mode 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.13.1	----------NOTES---------- 1. Not required to be performed until 12 hours after establishment of steady state operation. 2. Not applicable to primary to secondary LEAKAGE. ----------	
	Verify RCS operational LEAKAGE is within limits by performance of RCS water inventory balance.	In accordance with the Surveillance Frequency Control Program

Enclosure 4 to AEP-NRC-2022-05

Donald C. Cook Nuclear Plant Unit 2 Technical Specification Pages Marked to Show Proposed Changes

1.1 Definitions

	dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil."
ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME	The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC, or the components have been evaluated in accordance with an NRC approved methodology.
INSERVICE TESTING PROGRAM	The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).
LEAKAGE	LEAKAGE shall be: a. <u>Identified LEAKAGE</u> 1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank[;,] 2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known ~~either~~ [to] not ~~to~~ interfere with the operation of leakage detection systems ~~or not to be pressure boundary LEAKAGE,~~[;] or 3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE); b. <u>Unidentified LEAKAGE</u> All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE; and

1.1 Definitions

	c. Pressure Boundary LEAKAGE LEAKAGE (except primary to secondary LEAKAGE) through a ~~nonisolable~~ fault in an RCS component body, pipe wall, or vessel wall. LEAKAGE past seals, packing, and gaskets is not pressure boundary LEAKAGE.
MASTER RELAY TEST	A MASTER RELAY TEST shall consist of energizing all master relays in the channel required for channel OPERABILITY and verifying the OPERABILITY of each required master relay. The MASTER RELAY TEST shall include a continuity check of each associated required slave relay. The MASTER RELAY TEST may be performed by means of any series of sequential, overlapping, or total steps.
MODE	A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE – OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter 13, Initial Tests and Operation, of the UFSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 0.8 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE; and

d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressure boundary LEAKAGE exists	A.1 Isolate affected component, pipe, or vessel from the RCS by use of a closed manual valve, closed and de-activated automatic valve, blind flange, or check valve.	4 hours
~~B~~A. RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE.	~~B~~A.1 Reduce LEAKAGE to within limits.	4 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C~~B~~. Required Action and associated Completion Time ~~of Condition A~~ not met. ~~OR~~ ~~Pressure boundary LEAKAGE exists.~~ OR Primary to secondary LEAKAGE not within limit.	C~~B~~.1 Be in Mode 3. AND C~~B~~.2 Be in Mode 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.13.1	----------NOTES---------- 1. Not required to be performed until 12 hours after establishment of steady state operation. 2. Not applicable to primary to secondary LEAKAGE. --- Verify RCS operational LEAKAGE is within limits by performance of RCS water inventory balance.	In accordance with the Surveillance Frequency Control Program

Enclosure 5 to AEP-NRC-2022-05

Donald C. Cook Nuclear Plant Unit 1 Technical Specification Bases Pages Marked to Show Proposed Changes (For Information Only)

BASES

APPLICABLE SAFETY ANALYSES (continued)

to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the safety analysis.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, primary to secondary LEAKAGE is a factor in the dose releases outside containment in other accidents or transients involving secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The UFSAR (Ref. 3) analysis for SGTR assumes the contaminated secondary fluid is released via the steam generator power operated relief valves (and safety valves if their setpoint is reached) if offsite power is not available or if the condenser steam dump system fails to operate. The safety analysis for the SLB accident assumes 0.25 gpm per steam generator (1.0 gpm for all SGs) primary to secondary LEAKAGE as an initial condition. The dose consequences resulting from events resulting in a steam discharge to the atmosphere are within the limits defined in 10 CFR 50.67.

The RCS Operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

~~No p~~Pressure boundary LEAKAGE is prohibited ~~allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable~~ as the leak itself could cause further RCPB deterioration, resulting in higher LEAKAGE. ~~Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.~~

b. Unidentified LEAKAGE

The 0.8 gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air particulate monitoring equipment can detect within a reasonable time period. The limit of >.8 gpm is necessary to satisfy the requirements for the application of Leak-Before-Break methodology as documented in References 4 and 8, and approved by the NRC as documented in References 5 and 9. Separating the sources of leakage (i.e., leakage from an identified source versus leakage from an unidentified source) is necessary for prompt identification of

BASES

LCO (continued)

potentially adverse conditions, assessment of the safety significance, and corrective action. ~~Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.~~

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include ~~pressure boundary LEAKAGE or~~ controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). ~~Violation of this LCO could result in continued degradation of a component or system.~~

d. Primary to Secondary LEAKAGE Through Any One SG

The limit of 150 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 6). The Steam Generator Program operational LEAKAGE performance criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 and 6, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

BASES

ACTIONS

A.1

If pressure boundary LEAKAGE exists, the affected component, pipe, or vessel must be isolated from the RCS by a closed manual valve, closed and de-activated automatic valve, blind flange, or check valve within 4 hours. While in this condition, structural integrity of the system should be considered because the structural integrity of the part of the system within the isolation boundary must be maintained under all licensing basis conditions, including consideration of the potential for further degradation of the isolated location. Normal LEAKAGE past the isolation device is acceptable as it will limit RCS LEAKAGE and is included in identified or unidentified LEAKAGE. This action is necessary to prevent further deterioration of the RCPB.

B.1

Unidentified LEAKAGE or identified LEAKAGE in excess of the LCO limits must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

C~~B~~.1 and C~~B~~.2

If any of the Required Action~~s~~ and associated Completion Time~~s of Condition A is not~~ cannot be met, ~~if any pressure boundary LEAKAGE exists,~~ or if primary to secondary LEAKAGE is not within limit, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. ~~It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.~~ The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. ~~It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.~~ Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance.

The RCS water inventory balance must be performed with the reactor at steady state operating conditions. The Surveillance is modified by two Notes. Note 1 states that this SR is not required to be performed until 12 hours after establishing steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable unit conditions are established.

Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. ~~It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.~~ These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

Note 2 states that this SR is not applicable to primary to secondary LEAKAGE because LEAKAGE of 150 gallons per day cannot be measured accurately by an RCS water inventory balance.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.4.13.2

This SR verifies that primary to secondary LEAKAGE is less than or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.17, "Steam Generator Tube Integrity," should be evaluated. The primary to secondary LEAKAGE is

Enclosure 6 to AEP-NRC-2022-05

Donald C. Cook Nuclear Plant Unit 2 Technical Specification Bases Pages Marked to Show Proposed Changes (For Information Only)

BASES

APPLICABLE SAFETY ANALYSES (continued)

to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the safety analysis.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, primary to secondary LEAKAGE is a factor in the dose releases outside containment in other accidents or transients involving secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The UFSAR (Ref. 3) analysis for SGTR assumes the contaminated secondary fluid is released via the steam generator power operated relief valves (and safety valves if their setpoint is reached) if offsite power is not available or if the condenser steam dump system fails to operate. The safety analysis for the SLB accident assumes 0.25 gpm per SG (1.0 gpm for all SGs) primary to secondary LEAKAGE as an initial condition. The dose consequences resulting from events resulting in a steam discharge to the atmosphere are within the limits defined in 10 CFR 50.67.

The RCS Operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

~~No p~~Pressure boundary LEAKAGE is prohibited ~~allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable~~ as the leak itself could cause further RCPB deterioration, resulting in higher LEAKAGE. ~~Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.~~

b. Unidentified LEAKAGE

The 0.8 gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air particulate monitoring equipment can detect within a reasonable time period. The limit of 0.8 gpm is necessary to satisfy the requirements for the application of Leak-Before-Break methodology as documented in References 6 and 8, and approved by the NRC as documented in References 7 and 9. Separating the sources of leakage (i.e., leakage from an identified source versus leakage from an unidentified source) is necessary for prompt identification of

BASES

LCO (continued)

potentially adverse conditions, assessment of the safety significance, and corrective action. ~~Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.~~

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include ~~pressure boundary LEAKAGE or~~ controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). ~~Violation of this LCO could result in continued degradation of a component or system.~~

d. Primary to Secondary LEAKAGE Through Any One SG

The limit of 150 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 4). The Steam Generator Program operational LEAKAGE performance criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 and 6, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

BASES

ACTIONS

A.1

If pressure boundary LEAKAGE exists, the affected component, pipe, or vessel must be isolated from the RCS by a closed manual valve, closed and de-activated automatic valve, blind flange, or check valve within 4 hours. While in this condition, structural integrity of the system should be considered because the structural integrity of the part of the system within the isolation boundary must be maintained under all licensing basis conditions, including consideration of the potential for further degradation of the isolated location. Normal LEAKAGE past the isolation device is acceptable as it will limit RCS LEAKAGE and is included in identified or unidentified LEAKAGE. This action is necessary to prevent further deterioration of the RCPB.

B.1

Unidentified LEAKAGE or identified LEAKAGE in excess of the LCO limits must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

C~~B~~.1 and C~~B~~.2

If any of the Required Action~~s~~ and associated Completion Time~~s of Condition A is not~~ cannot be met, ~~if any pressure boundary LEAKAGE exists,~~ or if primary to secondary LEAKAGE is not within limit, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. ~~It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.~~ The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. ~~It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.~~ Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance.

The RCS water inventory balance must be performed with the reactor at steady state operating conditions. The Surveillance is modified by two Notes. Note 1 states that this SR is not required to be performed until 12 hours after establishing steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable unit conditions are established.

Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. ~~It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.~~ These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

Note 2 states that this SR is not applicable to primary to secondary LEAKAGE because LEAKAGE of 150 gallons per day cannot be measured accurately by an RCS water inventory balance.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.4.13.2

This SR verifies that primary to secondary LEAKAGE is less than or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.17, “Steam Generator Tube Integrity,” should be evaluated. The primary to secondary LEAKAGE is