INDIANA MICHIGAN POWER
A unit of American Electric Power

Indiana Michigan Power
Cook Nuclear Plant
One Cook Place
Bridgman, MI 49106
AEP.com

December 26, 2006

AEP:NRC:6449-01
10 CFR 50.90

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

SUBJECT: Donald C. Cook Nuclear Plant Units 1 and 2
Docket Nos. 50-315 and 50-316
Supplement to Application for Technical Specification (TS) Improvement Regarding Steam Generator Tube Integrity

References:
1. Letter from Joseph N. Jensen, Indiana Michigan Power Company (I&M), to U. S. Nuclear Regulatory Commission (NRC) Document Control Desk, "Application for Technical Specification (TS) Improvement Regarding Steam Generator Tube Integrity," AEP:NRC:6449, Accession Number ML061570157, dated May 26, 2006.
2. Electronic mail message from Peter Tam, NRC, to Michael Scarpello, I&M, et al., "D.C. Cook: Proposed RAI on Application for Amendment re. TSTF-449 (TAC MD2306 and MD2307)," Accession Number ML062370043, dated August 23, 2006.

Dear Sir or Madam:

By Reference 1, Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant Units 1 and 2, proposed to amend Facility Operating Licenses DPR-58 and DPR-74. I&M proposed to modify Technical Specifications (TS) requirements related to steam generator tube integrity consistent with TS Task Force (TSTF) generic change traveler TSTF-449, "Steam Generator Tube Integrity." By Reference 2, the U. S. Nuclear Regulatory Commission (NRC) staff provided draft questions that were discussed in a conference call with I&M personnel on August 30, 2006. During the conference call I&M stated that it would provide a formal response to the draft questions on the docket.

Enclosure 1 provides an affirmation statement pertaining to this letter. Enclosure 2 provides a response to the NRC questions as discussed during the August 30, 2006, conference call. Attachment 1 to this letter provides the marked-up TS page to replace the corresponding page submitted in Attachment 1A to Reference 1. Attachment 2 provides the TS page, with the changes incorporated, to replace the corresponding page submitted in Attachment 2A to Reference 1.

A001

Attachment 3 provides draft TS Bases pages for Unit 1 to replace the corresponding pages submitted in Attachment 3 to Reference 1. Unit 2 TS Bases changes are consistent with Unit 1 draft changes. The proposed TS changes, as supplemented by this letter, remain within the scope of the amendment previously proposed by Reference 1. Therefore, the No Significant Hazards Consideration evaluation and the evaluation of Environmental Considerations provided in Enclosure 2 to Reference 1 continue to bound the proposed changes.

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.

There are no commitments made in this letter. Should you have any questions, please contact Ms. Susan D. Simpson, Regulatory Affairs Manager, at (269) 466-2428.

Sincerely,

Mark A. Peifer
Site Vice President

KAS/rdw

Enclosures:
1. Affirmation
2. Response to Draft Request For Additional Information

Attachments:
1. Donald C. Cook Nuclear Plant Unit 1 Technical Specification Pages Marked To Show Changes
2. Donald C. Cook Nuclear Plant Unit 1 Technical Specification Pages With the Proposed Changes Incorporated
3. Donald C. Cook Nuclear Plant Unit 1 Technical Specification Bases Pages Marked To Show Changes

c: J. L. Caldwell, NRC Region III
K. D. Curry, Ft. Wayne AEP, w/o enclosures/attachments
J. T. King, MPSC
MDEQ – WHMD/RPMWS
NRC Resident Inspector
P. S. Tam, NRC Washington, DC

Enclosure 1 to AEP:NRC:6449-01

AFFIRMATION

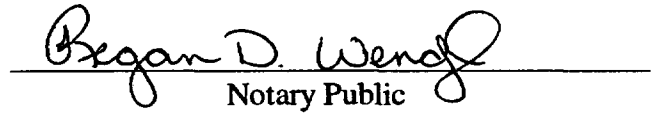
I, Mark A. Peifer, being duly sworn, state that I am Site Vice President of Indiana Michigan Power Company (I&M), that I am authorized to sign and file this request with the Nuclear Regulatory Commission on behalf of I&M, and that the statements made and the matters set forth herein pertaining to I&M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

Indiana Michigan Power Company

Mark A. Peifer
Site Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 26th DAY OF December, 2006



Notary Public

My Commission Expires

REGAN D. WENDZEL
Notary Public, Berrien County, MI
My Commission Expires Jan. 21, 2009

Enclosure 2 to AEP:NRC:6449-01

RESPONSE TO DRAFT REQUEST FOR ADDITIONAL INFORMATION

By Reference 1, Indiana Michigan Power Company (I&M) proposed to modify Technical Specification (TS) requirements related to steam generator tube integrity consistent with TS Task Force (TSTF) generic change traveler TSTF-449, "Steam Generator Tube Integrity." By electronic mail message (Reference 2), Nuclear Regulatory Commission (NRC) staff provided I&M with draft questions to discuss by conference call. Following discussions with NRC staff on August 30, 2006, I&M personnel stated that responses discussed during the conference call would be provided formally. The questions are addressed below.

NRC Question 1

"On Page 3.4.13-2 of your proposed technical specifications (TS) for Cook Unit 1, Condition C states that if identified LEAKAGE is not within limits, then the required action is to reduce LEAKAGE to within limits within 4 hours. This statement appears to imply that even if there is primary-to-secondary LEAKAGE (which is identified LEAKAGE), you would be allowed to reduce this leakage to within limits. TSTF-449, however, does not allow 4 hours to reduce primary-to-secondary leakage to within limits. Please discuss your plans to modify your proposed TSs in order to be fully consistent with TSTF-449. For example, the TS could be modified as follows: 'C. Identified LEAKAGE not within limits for reasons other than primary-to-secondary LEAKAGE.'

Your Bases may also need to be revised to clarify this issue (Page B3.4.13-6)."

I&M Response to NRC Question 1

I&M agrees with the NRC staff statement and has modified Unit 1 TS 3.4.13, Condition C to state, "Identified LEAKAGE not within limits for reasons other than primary-to-secondary LEAKAGE." Unit 1 TS Bases 3.4.13 has also been revised to be consistent with the proposed Unit 1 TS 3.4.13. The changes have been included in Attachments 1, 2, and 3.

NRC Question 2

"On Page B 3.4.13-4 of your proposed Bases there appears to be a typographical error. The sentence reads: '...performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 4).' It appears that Reference 4 should be Reference 6."

I&M Response to NRC Question 2

I&M agrees with the NRC staff statement and has modified TS Bases Page 3.4.13-4 and has included it in Attachment 3 for information.

NRC Question 3

"On Page B 3.4.17-2 of your proposed Bases there appears to be a statement missing. The second sentence of the second paragraph states: '...total primary to secondary LEAKAGE from all SGs of 1 gallon per minute (gpm) as a result of accident induced conditions.' Please discuss your plans to modify your proposal to include the following statement: '...or is assumed to increase to 1 gpm...' as part of this sentence which would make your submittal consistent with TSTF-449."

I&M Response to NRC Question 3

TS Bases Pages 3.4.13-2 and 3.4.17-2 have been revised for both Unit 1 and Unit 2 TS Bases to be consistent with TSTF-449. The Unit 1 TS Bases pages have been included in Attachment 3 for information.

NRC Question 4

"In your proposed Bases, you indicate that the 150 gpd leakage limit is measured at room temperature (e.g., Page B 3.4.13-9). However, in Enclosure 2 you indicate that the primary to secondary leakage is evaluated at hot (615.2-degrees Fahrenheit) conditions. Please discuss your plans to correct the apparent discrepancy."

I&M Response to NRC Question 4

I&M radiological accident analyses include both Control Room Habitability and offsite dose analyses. The Control Room Habitability dose analysis assumes the reactor coolant system (RCS) primary-to-secondary (P/S) leakage is at room temperature, while the offsite dose analysis assumes the RCS P/S leakage is at normal operating temperature and pressure. To remain conservative relative to current and proposed TS limiting condition for operation for surveilling P/S leakage, I&M multiplies the Electric Power Research Institute guideline equations for determining leakage by a volume correction factor of 1.52. The correction factor ensures the smaller mass release assumed in the offsite dose analysis bounds the TS measured leak rate. While this is overly conservative relative to Control Room Habitability dose analyses, this is appropriately conservative for offsite dose analyses. NRC staff suggested we clarify this point in CNP TS Bases. Unit 1 and Unit 2 TS Bases Surveillance Requirement 3.4.13.2 have been

revised to clarify CNP calculation of P/S leak rate. The Unit 1 TS Bases Page 3.4.13-9 has been included in Attachment 3 for information.

References for this Enclosure

1. Letter from Joseph N. Jensen, I&M, to NRC Document Control Desk, "Application for Technical Specification (TS) Improvement Regarding Steam Generator Tube Integrity," AEP:NRC:6449, Accession Number ML061570157, dated May 26, 2006.

2. Electronic mail message from Peter Tam, NRC, to Michael Scarpello, I&M, et al., "D.C. Cook: Proposed RAI on Application for Amendment re. TSTF-449 (TAC MD2306 and MD2307),"Accession Number ML062370043, dated August 23, 2006.

Attachment 1 to AEP:NRC:6449-01

DONALD C. COOK NUCLEAR PLANT UNIT 1 TECHNICAL SPECIFICATION PAGES
MARKED TO SHOW CHANGES

3.4.13-2

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Identified LEAKAGE not within limits for reasons other than primary to secondary LEAKAGE. OR ~~Primary to secondary LEAKAGE not within limits.~~	C.1 Reduce LEAKAGE to within limits.	4 hours
D. Required Action and associated Completion Time of Condition A, B, or C not met. OR Pressure boundary LEAKAGE exists. OR Primary to secondary LEAKAGE not within limit.~~SR 3.4.13.2 not met.~~	D.1 Be in MODE 3. AND D.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~~leakage~~ is within limits by performance of RCS water inventory balance.	72 hours

Attachment 2 to AEP:NRC:6449-01

DONALD C. COOK NUCLEAR PLANT UNIT 1 TECHNICAL SPECIFICATION PAGES WITH THE PROPOSED CHANGES INCORPORATED

3.4.13-2

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Identified LEAKAGE not within limits for reasons other than primary to secondary LEAKAGE.	C.1 Reduce LEAKAGE to within limits.	4 hours
D. Required Action and associated Completion Time of Condition A, B, or C not met. OR Pressure boundary LEAKAGE exists. OR Primary to secondary LEAKAGE not within limit.	D.1 Be in MODE 3. AND D.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.	72 hours

Attachment 3 to AEP:NRC:6449-01

DONALD C. COOK NUCLEAR PLANT UNIT 1 TECHNICAL SPECIFICATION BASES
PAGES MARKED TO SHOW CHANGES

B 3.4.13-2

B 3.4.13-4

B 3.4.13-6

B 3.4.13-9

B 3.4.17-2

BASES

APPLICABLE SAFETY ANALYSES (continued)

minute (gpm) or is assumed to increase to one gpm as a result of accident induced conditions. The LCO requirement 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. ~~at least a 1 gpm primary to secondary LEAKAGE as the initial condition~~.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident and other accidents or transients involve 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 the amount of primary to secondary LEAKAGE in the three intact SGs is 1 gpm minus a faulted SG tube LEAKAGE of 500 gallons per day as an initial condition. The dose consequences resulting from events resulting in a steam discharge to the atmosphere are within a small fraction of the limits defined in 10 CFR 100 and within GDC-19.

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 pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further 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 is established for the pressurizer surge line in the leak before break methodology. Violation of this LCO could result in

BASES

LCO (continued)

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.

~~d. Primary to Secondary LEAKAGE through All Steam Generators (SGs)~~

~~Total primary to secondary LEAKAGE amounting to 1 gpm through the intact SGs produces acceptable offsite and control room doses in the SGTR accident analysis. For the SLB accident, the amount of primary to secondary LEAKAGE in the three intact SGs is assumed to be 1 gpm minus a faulted SG tube LEAKAGE of 500 gallons per day. The LCO limit of 600 gallons per day is more conservative than the 1 gpm value assumed in the offsite dose calculations. This limit is imposed to help minimize the potential for excessive leakage or tube burst in the event of a MSLB or LOCA consistent with the LCO limit on primary to secondary LEAKAGE through any one SG. In addition, the conservative limit is appropriate due to the increased steam release as a result of the replacement SGs. Violation of this LCO could exceed the offsite dose limits for these accidents. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE.~~

~~e. Primary to Secondary LEAKAGE through Any One SG~~

BASES

APPLICABILITY (continued)

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.

ACTIONS

A.1 and A.2

With unidentified LEAKAGE > 0.8 gpm, the pressurizer surge line must be verified not to be the source of unidentified LEAKAGE or the unidentified LEAKAGE must be reduced to within limit within 4 hours. These Required Actions are necessary to satisfy the requirements for the application of Leak-Before-Break methodology to the pressurizer surge line as documented in Reference 4 and approved by the NRC as documented in Reference 5, and are necessary to prevent further deterioration of the RCPB associated with the pressurizer surge line. The Completion Time allows time to verify leakage rates and either identify the unidentified LEAKAGE or reduce LEAKAGE to within limit before the reactor must be shut down.

B.1

Unidentified LEAKAGE > 1.0 gpm must be reduced to ≤ 1.0 gpm 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.1

Identified LEAKAGE~~,~~ ~~or~~ with the exception of primary to secondary 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 reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

D.1 and D.2

If any Required Action and associated Completion Time of Condition A, B, or C is not met, if any pressure boundary LEAKAGE exists, or if primary to secondary LEAKAGE~~the SR 3.4.13.2~~ is not within limit~~met~~, 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

BASES

SURVEILLANCE REQUIREMENTS (continued)

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 measured at room temperature as described in Reference 7. Prior to comparison with the 150 gallons per day TS limit, the measured primary to secondary LEAKAGE is multiplied by a volume correction factor of 1.52. The correction factor ensures the offsite dose analyses, which assume primary to secondary leakage is at normal operating temperature and pressure, remain bounding. The operational LEAKAGE rate limit applies to LEAKAGE through any one SG. If it is not practical to assign the LEAKAGE to an individual SG, then all of the primary to secondary LEAKAGE should be conservatively assumed to be from one SG.

The Surveillance is modified by a Note which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary to secondary LEAKAGE determination, 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.

The Surveillance Frequency of 72 hours is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. The primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Ref. 7).
~~This SR provides the means necessary to determine SG OPERABILITY in an operational MODE. The requirement to demonstrate SG tube integrity in accordance with the Steam Generator Program emphasizes the importance of SG tube integrity, even though this Surveillance cannot be performed at normal operating conditions~~

REFERENCES

1. UFSAR, Section 1.4.3.
2. Regulatory Guide 1.45, May 1973.
3. UFSAR, Section 14.2.4.
4. Letter from Indiana Michigan Power Company (M. W. Rencheck) to the NRC dated October 26, 2000 (Letter C1000-20).

BASES

APPLICABLE SAFETY ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of an SGTR event assumes a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in LCO 3.4.13, "RCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for an SGTR assumes the contaminated secondary fluid is released to the atmosphere via the SG power operated relief valves.

The analysis for design basis accidents and transients other than an SGTR assumes the SG tubes retain their structural integrity (i.e., they are assumed not to rupture.) In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of one gallon per minute (gpm) or is assumed to increase to one gpm as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.16, "RCS Specific Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), 10 CFR 100 (Ref. 3) or the NRC approved licensing basis (e.g., a small fraction of these limits).

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the repair criteria be plugged in accordance with the Steam Generator Program.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. If a tube was determined to satisfy the repair criteria but was not plugged, the tube may still have tube integrity.

In the context of this Specification, an SG tube is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

An SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.7, "Steam Generator Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.