July 30, 2004

LICENSEE: Indiana Michigan Power Company

FACILITY: Donald C. Cook Nuclear Plant, Units 1 and 2

SUBJECT: SUMMARY OF TELEPHONE CONFERENCE HELD ON JULY 8, 2004, BETWEEN THE U.S. NUCLEAR REGULATORY COMMISSION (NRC) AND INDIANA MICHIGAN POWER COMPANY (I&M) REPRESENTATIVES CONCERNING REQUESTS FOR ADDITIONAL INFORMATION ON DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2, LICENSE RENEWAL APPLICATION (TAC NOS. MC1202 AND MC1203)

The U.S. Nuclear Regulatory Commission staff (the staff) and representatives of Indiana Michigan Power Company (the applicant) held a telephone conference call on July 8, 2004, to discuss requests for additional information (RAIs) concerning the Donald C. Cook Nuclear Plant (CNP) license renewal application (LRA). The conference call consisted of discussions on a new RAI and I&M's responses to previously submitted RAIs with which the staff had additional questions. During the conference call, the staff also requested that the applicant provide clarification to questions concerning the LRA to which no RAIs where issued so that the staff could more efficiently complete the safety evaluation report (SER).

On the basis of the discussions, the applicant was able to better understand the staff's RAI. The conference call was also useful in clarifying the staff's questions. No staff decisions were made during the meeting.

Enclosure 1 provides a listing of the telephone conference participants. Enclosure 2 contains the RAIs discussed with the applicant, including a brief description on the status of the item, and the questions for which clarification was requested. The applicant has had an opportunity to comment on this summary.

/RA/

Jonathan Rowley, Project Manager
License Renewal Section A
License Renewal and Environmental Impacts Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Docket Nos.: 50-315 and 50-316

Enclosures: As stated

cc w/enclosures: See next page

LICENSEE: Indiana Michigan Power Company

FACILITY: Donald C. Cook Nuclear Plant, Units 1 and 2

SUBJECT: SUMMARY OF TELEPHONE CONFERENCE HELD ON JULY 8, 2004, BETWEEN THE U.S. NUCLEAR REGULATORY COMMISSION (NRC) AND INDIANA MICHIGAN POWER COMPANY (I&M) REPRESENTATIVES CONCERNING REQUESTS FOR ADDITIONAL INFORMATION ON DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2, LICENSE RENEWAL APPLICATION (TAC NOS. MC1202 AND MC1203)

The U.S. Nuclear Regulatory Commission staff (the staff) and representatives of Indiana Michigan Power Company (the applicant) held a telephone conference call on July 8, 2004, to discuss requests for additional information (RAIs) concerning the Donald C. Cook Nuclear Plant (CNP) license renewal application (LRA). The conference call consisted of discussions on a new RAI and I&M's responses to previously submitted RAIs with which the staff had additional questions. During the conference call, the staff also requested that the applicant provide clarification to questions concerning the LRA to which no RAIs where issued so that the staff could more efficiently complete the safety evaluation report (SER).

On the basis of the discussions, the applicant was able to better understand the staff's RAI. The conference call was also useful in clarifying the staff's questions. No staff decisions were made during the meeting.

Enclosure 1 provides a listing of the telephone conference participants. Enclosure 2 contains the RAIs discussed with the applicant, including a brief description on the status of the item, and the questions for which clarification was requested. The applicant has had an opportunity to comment on this summary.

/RA/
Jonathan Rowley, Project Manager
License Renewal Section A
License Renewal and Environmental Impacts Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Docket Nos.: 50-315 and 50-316

Enclosures: As stated

cc w/enclosures: See next page

Accession No.: ML042120480
Document Name:C:\ORPCheckout\FileNET\ML042120480.wpd

OFFICE	PM:RLEP	LA:RLEP	SC:RLEP
NAME	JRowley	YEdmonds	SLee
DATE	07/30/04	07/30/04	07/30/04

OFFICIAL RECORD COPY

DISTRIBUTION: Dated: July 30, 2004   Accession No: ML042120480

HARD COPY
RLEP RF
J. Rowley (PM)

E-MAIL:
RidsNrrDrip
RidsNrrDe
G. Bagchi
K. Manoly
W. Bateman
J. Calvo
R. Jenkins
P. Shemanski
J. Fair
RidsNrrDssa
RidsNrrDipm
D. Thatcher
R. Pettis
C. Li
M. Itzkowitz (RidsOgcMailCenter)
R. Weisman
M. Mayfield
A. Murphy
S. Smith (srs3)
S. Duraiswamy
Y. L. (Renee) Li
RLEP Staff

R. Gramm
A. Howell
J. Stang
J. Strasma, RIII
M. Kotzalas
OPA
NRR/ADPT secretary (RidsNrrAdpt)

LIST OF PARTICIPANTS FOR TELEPHONE CONFERENCE CONCERNING DRAFT REQUESTS FOR ADDITIONAL INFORMATION JULY 8, 2004

Participants	**Affiliation**
Jonathan Rowley	U.S. Nuclear Regulatory Commission (NRC)
John Fair	NRC
Eric Reichelt	NRC
Neil Haggerty	Indiana Michigan Power Company (I&M)
Bob Kalinowski	I&M
Richard Gumbir	I&M
Mark Rinckel	Framatome*
Allen Cox	Entergy*

*I&M contractor

Enclosure 1

REQUESTS FOR ADDITIONAL INFORMATION (RAIs) DISCUSSED FOR DONALD C. COOK (CNP), UNITS 1 AND 2, LICENSE RENEWAL DURING JULY 8, 2004 TELEPHONE CONFERENCE

Donald C. Cook (CNP) LRA Section B.1.1, "Alloy 600 Aging Management"

RAI B.1.1.2-3

Acceptance Criteria: The applicant stated that the acceptance criteria for volumetric and visual inspections will be based upon the requirements in ASME Section XI.

As a minimum, the applicant is required by 10 CFR 50.55a to comply with the flaw acceptance criteria specified for ASME Class 1 components in the ASME Code Section XI, Articles IWA-3000 and IWB-3000, regardless of whether the material is fabricated from Alloy 600. The applicant may use alternative acceptance criteria either by the applicant or the industry if the alternative criteria have been submitted to and accepted by the staff pursuant to 10 CFR 50.55a(a)(3). The acceptance criteria stated was not definitive enough to determine if the applicant would allow pressure boundary leakage if the fracture mechanics analysis proved that the component could perform its intended function.

The staff requests the applicant to discuss the process for calculating specific numerical values of conditional acceptance criteria to ensure that the structure and component intended functions will be maintained under all current licensing basis (CLB) design conditions. The discussion needs to focus on how pressure boundary leakage due to primary water stress-corrosion cracking (PWSCC) will be handled.

Status: The applicant was unsure as to whether the staff intended for the conditional acceptance criteria applied to past or future activities. The staff stated that the request was for future activities. The staff indicated that the applicant will need to make a commitment to participate in industry initiatives, submit an inspection plan 3 years prior to entering the period of extended operation, and to revise the Future Action Commitment. Indiana Michigan Power Company (I&M) indicated that the question was clear.

RAI 4.3.1-1

Section 4.3.1 of the LRA discusses the fatigue evaluation of the Unit 1 auxiliary spray line that was performed in response to NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems." The LRA indicates that this fatigue evaluation is contained in WCAP-14070, "Evaluation of Cook Units 1 and 2 Auxiliary Spray Piping per NRC Bulletin 88-08," July 1994. Provide a copy of WCAP-14070.

I&M Response to RAI 4.3.1-1:

Attachment 2 to this letter provides a letter for withholding proprietary information and an accompanying affidavit. Attachment 3 to this letter provides a copy of the proprietary report, WCAP-14070-P, "Evaluation of Donald C. Cook Units 1 and 2 Auxiliary Spray Piping per NRC Bulletin 88-08," dated May 2004. Attachment 4 provides a copy of the non-proprietary version of this report, WCAP-14070-NP, dated May 2004.

As Attachment 3 contains information proprietary to Westinghouse Electric Company, LLC (Westinghouse), it is supported by an affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis upon which the Westinghouse proprietary information contained in Attachment 3 may be withheld from public disclosure by the NRC and addresses, with specificity, the consideration listed in Paragraph (b)(4) of 10 CFR 2.390.

Correspondence with respect to the copyright or proprietary aspects of the item listed above or the supporting Westinghouse affidavit should reference CAW-04-1835 and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company, LLC P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Status: The applicant was requested to confirm that there is no time dependency for the frequency of 0.0023 Hz specified in the Note on page 6-3 of WCAP-14070.

RAI 4.3.3-1

Section 4.3.3 of the LRA discusses I&M's evaluation of the impact of the reactor water environment on the fatigue life of components. The discussion references the fatigue sensitive component locations for an early vintage Westinghouse plant identified in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components." The LRA indicates that the design usage factors provided in Table 5-98 of NUREG/CR-6260 were used for the evaluation of the charging nozzle, safety injection nozzle and RHR [residual heat removal] tee. The design usage factors were based on an evaluation of the Turkey Point facility, including a plant specific evaluation of the RHR piping and detailed finite element analyses of the charging and safety injection nozzles. Discuss the applicability of these analyses to the Cook facility. The discussion should include a comparison of piping sizes and thicknesses, including the design of the thermal sleeves between Cook and Turkey Point. The discussion should also include a comparison of the number and type of design transients cycles between Cook and Turkey Point.

I&M Response to RAI 4.3.3-1:

As discussed in LRA Section 4.3.3, since the CNP Class 1 piping was designed to USAS B31.1, fatigue analyses had not been conducted for the charging nozzle, safety injection nozzle, and RHR Class 1 piping. The standard reference plant evaluations for these locations are described in NUREG/CR-6260 Sections 5.5.4, 5.5.5, and 5.5.6. Section 1 of NUREG/CR-6260 addresses the differentiation between newer vintage and older vintage plants based on whether components of the reactor coolant pressure boundary were designed to codes that require an explicit fatigue analysis. CNP is an older vintage Westinghouse plant since the RCS piping at CNP was designed in accordance with USAS B31.1.

A general comparison to Turkey Point Nuclear Plant is provided below for the major RCS design transients, charging nozzles, safety injection nozzles, and RHR piping. Based on this review, I&M will enhance the Fatigue Monitoring Program to address a plant-specific fatigue analysis of the Class 1 portions of RHR piping, as described below. Due to configuration and functional transient similarities between the CNP charging and safety injection nozzles and those evaluated for Turkey Point in NUREG/CR-6260, CNP-specific evaluations for these components are not necessary, as they would be expected to yield nearly identical results to those approved for Turkey Point.

Major reactor coolant system (RCS) design transients for CNP are listed in LRA Table 4.3-1. These transients were compared to the RCS design transients listed in Table 5-83 of NUREG/CR-6260. All RCS design transients are the same with the exceptions of 5% power change (14,500 in NUREG/CR-6260 versus 11,680 and 18,300 for CNP Unit 1 and Unit 2, respectively) and 10% power change (down) (7,000 in NUREG/CR-6260 versus the more limiting 2,000 for CNP). These differences are of minor significance for the American Society of Mechanical Engineers (ASME) Code fatigue evaluations of the charging nozzle, safety injection nozzle, and RHR Class 1 piping.

Charging Nozzles

The 3-inch Schedule 160 charging nozzles at CNP are fabricated from American Society for Testing and Materials (ASTM) A182 Type 316 stainless steel. Thermal sleeves attached to the charging nozzles are fabricated from ASTM A312 Type 304 stainless steel. This is consistent with size (i.e., pipe diameter and schedule) and materials of construction for the charging nozzles at Turkey Point Units 3 and 4.

A detailed comparison of Westinghouse-designed RCS piping nozzle configurations (CNP and Turkey Point) is provided in WCAP-14575-A, *Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components*, Tables 2-1 and 4-4 and Figures 2-11 and 2-12. The charging nozzle and thermal sleeve designs for CNP and Turkey Point are very similar. Specifically, the interior surface of the CNP nozzle contains no reentrant corners, the thermal sleeve is welded to the interior of the nozzle, and the nozzle-to-outboard piping weld is a field weld.

The major charging system transients defined in NUREG/CR-6260, Table 5-89, are based on representative transients from other pressurized water reactor plants reviewed in the study. Similar to Turkey Point, CNP has no fatigue analysis of the charging nozzles; the transients defined in NUREG/CR-6260, Table 5-89, provide a reasonable approximation for CNP.

As described in NUREG/CR-6260, Section 5.5.4.1, axial and radial thermal gradients produced the highest contribution of stress intensity for the charging nozzle. Plant-specific (Turkey Point) moments applied to the nozzle by the attached piping are not a significant contributor to fatigue of the charging nozzle.

In summary, the CNP charging nozzle and thermal sleeve configuration and functional transients are similar to the Turkey Point charging nozzle evaluated in NUREG/CR-6260. Therefore, a plant-specific evaluation for the CNP charging nozzle would be of minimum value, as it would be expected to yield nearly identical results to the results reported in

NUREG/CR-6260 for the Turkey Point charging nozzle with regard to ASME Code calculated cumulative usage factors (CUFs).

Safety Injection Nozzle

A 10-inch Schedule 140 safety injection nozzle is connected to each cold leg at CNP. The 10-inch nozzle directs flow from the accumulators, medium-head safety injection pumps, and residual heat removal (RHR) pumps to the RCS. The 10-inch safety injection nozzles in Loops 2 and 3 include thermal sleeves. The corresponding nozzles in Loops 1 and 4 do not contain thermal sleeves.

The 10-inch safety injection nozzles at CNP are fabricated from ASTM A182 Type 316 stainless steel. The thermal sleeves attached to the nozzles in Loops 2 and 3 are fabricated from ASTM A312 Type 304 stainless steel. The safety injection nozzles and associated thermal sleeves in Loops 2 and 3 at CNP are consistent with the size (i.e., pipe diameter and schedule) and materials of construction for the 10-inch safety injection nozzles at Turkey Point Units 3 and 4.

A detailed comparison of Westinghouse-designed RCS piping nozzle configurations (including CNP and Turkey Point) is provided in WCAP-14575-A, Tables 2-1 and 4-4 and Figures 2-11 and 2-12. The safety injection nozzle and thermal sleeve designs for CNP and Turkey Point are very similar. Specifically, the interior surface of the CNP nozzle contains no reentrant corners, the thermal sleeve is welded to the interior of the nozzle, and the nozzle-to-outboard piping weld is a field weld.

From Section 5.5.5 of NUREG/CR-6260, two transients were identified as the leading contributors to usage for the safety injection nozzle: (1) 70 cycles of emergency injection, and (2) 200 cycles of RHR initiation during cooldown. Two hundred RHR cycles is consistent with CNP cooldown cycles defined in LRA Table 4.3-1. Seventy cycles of emergency injection is a conservative estimate for CNP.

As described in NUREG/CR-6260, Section 5.5.5.1, axial and radial thermal gradients produced the highest contribution of stress intensity for the safety injection nozzle. Plant-specific (Turkey Point) moments applied to the nozzle by the attached piping are not a significant contributor to fatigue of the safety injection nozzle.

In summary, the CNP safety injection nozzle and thermal sleeve configurations and functional transients are similar to the Turkey Point charging nozzle evaluated in NUREG/CR-6260. Therefore, a plant-specific evaluation for the CNP safety injection nozzle would be of minimal value, as it would be expected to yield nearly identical results to the results reported in NUREG/CR-6260 for the Turkey Point safety injection nozzle with regard to ASME Code-calculated CUFs.

Residual Heat Removal System Class 1 Piping

The Class 1 RHR piping at the 4-loop CNP units is a different configuration than the RHR piping at the 3-loop Turkey Point units. Therefore, as an enhancement to the Fatigue Monitoring Program described in LRA Section B.2.2, I&M will perform one or more of the following activities prior to the period of extended operation for Class 1 portions of RHR piping:

(1) A plant-specific fatigue analysis of Class 1 portions of RHR piping, which includes environmental effects, will be performed to ensure that cumulative usage factors remain below 1.0

(2) Repair of the Class 1 portions of RHR piping at the affected locations

(3) Replacement of the Class 1 portions of RHR piping at the affected locations

(4) Manage the effects of fatigue of the Class 1 portions of RHR piping by an NRC-approved inspection program (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method accepted by the NRC). The inspections are expected to be able to detect cracking due to thermal fatigue prior to loss of function. Replacement or repair will then be implemented such that the intended function will be maintained for the period of extended operation, and

(5) Monitor ASME Code activities to use the environmental fatigue methodology approved by the ASME Code committee and NRC.

Status: The staff indicated that the responses does not contain sufficient information for the NRC to conclude the CNP charging and safety injection nozzles are bounded by charging and safety injection (SI) analysis in NUREG-6260 for older vintage Westinghouse plants. Additional information is needed or alternatively a commitment similar to that made for the RHR piping. I&M indicated that they would provide a supplemental response.

RAI 4.6.1-1

Section 4.6.1 of the LRA discusses the evaluation of the containment liner. The LRA indicates that the liner was evaluated in 1999 after the discovery of localized thinning of the liner. Indicate the amount and extent of the localized liner thinning. Describe how the fatigue evaluation of the locally thinned area was performed.

I&M Response to RAI 4.6.1-1:

The containment structure, as shown in Figure 1 below, consists of side walls measuring 113 feet (nominal) in height from the liner on the base to the spring line of the dome and has a nominal inside diameter of 115 feet. The thickness of the cylinder is 3 feet - 6 inches and the thickness of the dome is 3 feet - 6 inches at the spring line tapering uniformly to 2 feet - 6 inches at the peak of the dome. The base mat consists of a 10-foot thick structural concrete slab, increasing to 20 feet adjacent to the recirculation sump area. The base mat is covered by a ¼-inch nominal thickness carbon steel plate.

A 24-inch thick annulus floor, which is approximately 13 feet wide and extends from the outside of the crane wall to the inside of the vertical containment liner, sits on top of the base mat liner plate (see Figure 2 below). The annulus floor extends around the circumference of the containment, except for 16 feet where a concrete wall exists. A gap between the vertical wall liner and the annulus floor slab was provided to allow for differential expansion. The gap is filled with a ½-inch thick crushable spacer material (Dorvon FR100, closed cell polystyrene similar to Styrofoam®). To protect the bottom liner area (shown in Figure 2) from moisture intrusion, a sealant was applied on top of the Dorvon at the annulus floor grade level between the steel containment liner and the concrete annulus floor.

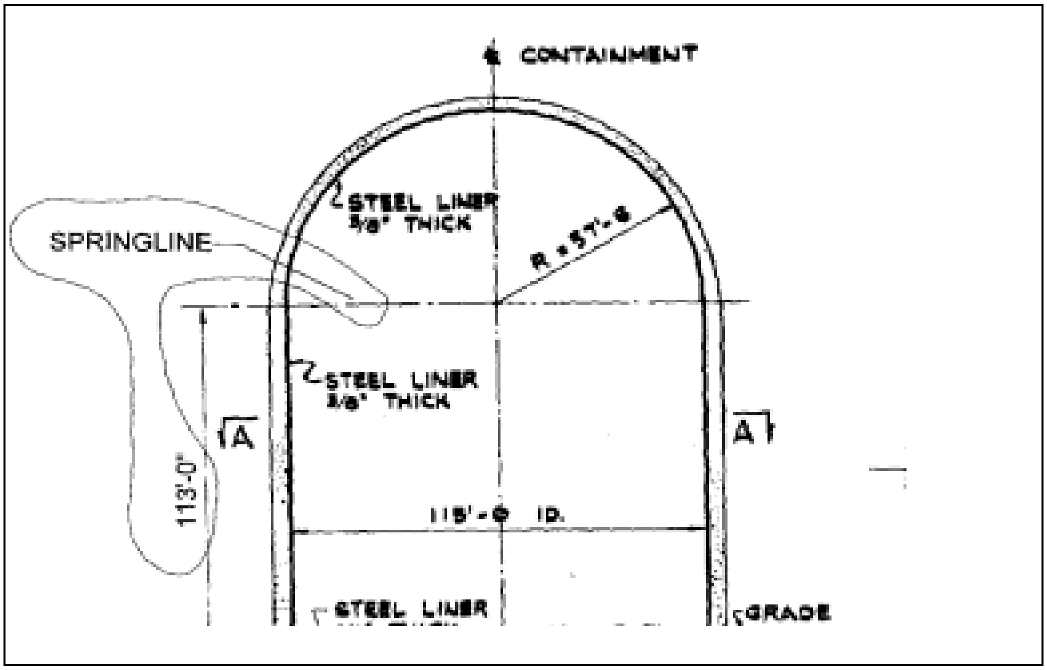

Figure 1—CNP Containment Sectional Elevation (UFSAR Figure 5.2.2-3)

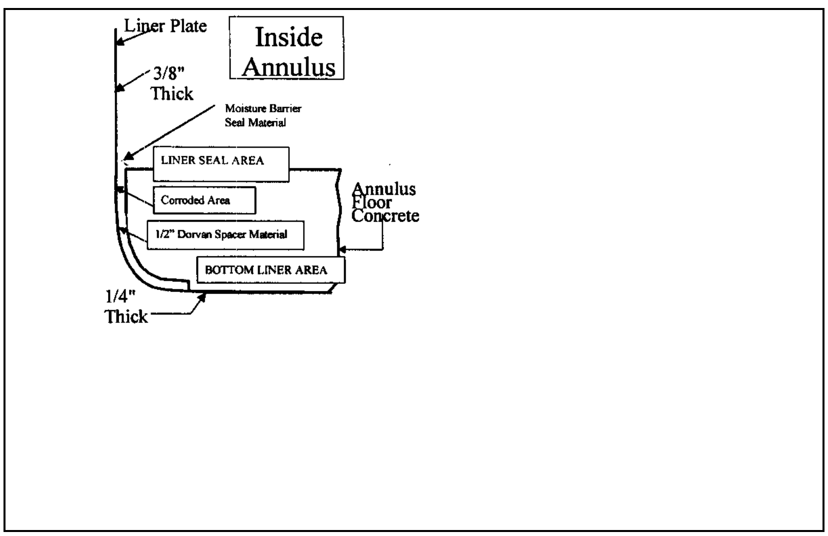

Figure 2-Annulus Floor
(Crane Wall Not Shown—Located 13 feet to the right of the vertical liner plate)

In May 1997, during an inspection of the Unit 2 liner plate with particular attention to the liner seal area (see Figure 2), degraded and missing sealant material were noted. Minor degraded coatings were noted above the sealant, but no corrosion of the liner was observed. During the sealant repair activity in 1999, corrosion/pitting were noted behind the sealant material. A CR was issued and a similar inspection was performed for Unit 1. The final visual inspection results were as follows:

- Unit 1 – 61 pits in the carbon steel liner plate that exceeded 0.125-inch (1/8-inch) in depth, with the worst condition reported as 0.172-inch in depth. Nominal thickness of the liner plate is 3/8-inch.

- Unit 2 – Two pits in the carbon steel liner plate that were 1/8-inch in depth. Nominal thickness of the liner plate is 3/8-inch. It was determined that the Unit 2 containment liner plate will remain within the design basis, as the identified pitting did not exceed the established acceptance criterion.

The reported pitting corrosion in Unit 1 exceeded the acceptance criteria that required a 0.250-inch minimum wall thickness. A root cause investigation and structural integrity evaluation of Unit 1 followed.

The investigation revealed that the corrosion/pitting at the liner seal area was the result of degradation of the sealant over time, which allowed the ingress of moisture between the containment liner plate and the seal material. The crevice between the steel liner plate and the sealant material created a small volume of stagnant water and provided conditions for pitting to proceed. The pitting was localized and decreased with increasing depth from the floor level and ultimately vanished.

The seal failure prompted investigations of two other areas protected by similar sealants – the bottom liner area and annulus slab seal joint. Investigation of the chemistry in these areas revealed high pH and low chlorides that was not conducive to pitting/crevice corrosion. This was corroborated with published corrosion studies and by fiber optic camera inspections, which revealed no active corrosion in the lower portion of the bottom liner area.

The only area of significant material loss was at the liner seal area (Figure 2). The affected areas were recoated and resealed, and no further measurable corrosion is expected. Plant procedures were modified to preclude conditions that could compromise seal integrity. The liner seal area has been added to the ASME Section XI, Subsection IWE, and the ASME Section XI, Subsection IWL, Inservice Inspection Programs to assure continued monitoring of this area of degradation.

Since the Unit 1 liner plate wall thickness had decreased below minimum allowable thickness, a detailed structural evaluation was performed to justify continued operation for Unit 1. The structural evaluation included consideration of the safety functions of the liner, design margins and safety margins in the liner and anchorage system, strains and deformations imposed on the liner anchors and their relation to liner thickness, and studies conducted by Sandia National Laboratories regarding the effect of corrosion on the mechanical properties of steel liner plate material and ultimate pressure capacity of a 1/6-scale mode steel-lined reinforced concrete containment. The evaluation concluded that:

- the design basis for the concrete containment structure, the containment liner, or the liner anchorage system is not degraded by the reported inservice inspection conditions of the containment liner plate,

- the thickness of the remaining sound metal is adequate to maintain the design safety function of the liner as a leak-tight membrane, and
- the ultimate pressure capacity of the concrete containment structure to withstand severe accident pressure is not degraded by the reported inservice corrosion conditions of the containment liner plate.

The structural evaluation of the degraded liner plate for Unit 1 included an assessment of cyclic loading and fatigue life. The discussion of containment liner plate fatigue in LRA Section 4.6.1 is based on the significant fatigue resistance of the corroded liner, as follows:

<u>Cyclic Loading (Reference 1)</u>:

The load cycles experienced by the liner during the plant life are enumerated below:

<u>Thermal Cycles</u>

- 40 cycles of annual outdoor temperature variations during the 40-year life of the plant. Daily variations in environmental temperature do not significantly penetrate the concrete shell to influence cycling on the liner
- 200 cycles of containment interior temperature variations during reactor system startup and shutdown
- 1 cycle of design accident transient

<u>Load Cycles</u>

- 1 cycle of pre-operational structural integrity test (SIT) pressure
- 13 cycles of pre-operational and inservice integrated leakage rate test pressure
- 10 cycles due to earthquake

The predominant stress cycles experienced by the liner stem from thermal cycles associated with reactor startup and shutdown under normal operating condition with the containment temperature ranging from 60°F to 120°F. Conservatively assuming that the thermal expansion of the liner due to $\Delta T = (120-60) = 60°F$ is completely restrained by the concrete containment shell, the cyclic thermal stress range in the liner due to restrained thermal expansion is:

$$E_s \bullet \alpha_s \bullet \Delta T = \{(29 \times 10^6)(6.5 \times 10^{-6})(60)\} = 11.3 \text{ ksi, compressive}$$

(Notation: E_s – Young's modulus of steel liner, 29×10^6 psi; α_s – thermal expansion coefficient of steel liner, 6.5×10^{-6} in/in/°F). Other concurrent loads (dead load and shrinkage) also impose compressive strains, but their magnitude is relatively insignificant.

Fatigue Life

Based on fatigue tests, the design basis fatigue life of the uncorroded liner plate of the CNP containment under a complete stress reversal range from 20 ksi compression to 20 ksi tension is 180,000 cycles (Reference 1).

Considering that the actual cyclic thermal stress range is only 11.3 ksi compression and no tension, and that the number of containment load and thermal cycles experienced by the liner during the plant life are insignificant compared to the large fatigue life of the liner, the fatigue resistance of the liner is not a concern in the design of the containment liner.

Fatigue Resistance of Corroded Liner

Sandia National Laboratories recently conducted experimental studies, sponsored by the NRC, to investigate the mechanical properties of a 1/16-inch thick steel plate material with general corrosion and pitting. Reference 2 contains the details of this investigation.

The number of containment load and thermal cycles that the liner will experience during the plant life is insignificant compared to the large design fatigue life of the uncorroded liner. The corroded plate can easily endure this relatively very small number of load cycles without incurring any fatigue-related degradation. For this reason, Sandia National Laboratories did not perform fatigue tests of corroded steel plates in their NRC-sponsored investigation of corroded steel plates described in Reference 2.

The following additional favorable factors and considerations further reinforce this conclusion:

1. The liner plate is fabricated from normalized ASTM A442 Grade 60 steel with a fine grain microstructure that has excellent fracture toughness. Better fracture toughness increases the fatigue resistance of the material.

2. Stress concentration is the most critical factor affecting the fatigue life of steel. Visual and magnetic particle examinations as well as digital photographs of samples of the corroded liner show no evidence of any sharp notch or crack-like flaws in the corroded areas, indicating relatively low stress and strain concentrations around the corrosion pits and on the corroded rough surfaces.

3. The predominant load cycles that the liner experiences during the plant life is due to thermal cycles associated with the reactor startup and shutdown under normal operating condition. The corresponding thermal stress range in the liner is from 11.3 ksi compression to no tension, which is significantly lower than the complete reversal stress range from 20 ksi compression to 20 ksi tension employed in the fatigue test to establish the design fatigue life of 180,000 cycles for the uncorroded liner plate. The lower the stress range, the higher the steel fatigue life. In addition, a cyclic stress range or stress fluctuation in compression is a significantly lesser fatigue concern than a full steel reversal or stress fluctuation in tension (Reference 3).

4. According to the fatigue design provisions of the American Institute of Steel Construction (AISC) Code for welded steel structures (Reference 3), which are based

on the fatigue design provisions of the American Association of Highway and Transportation Officials (AASHTO) Code for highway bridges (References 4 and 5), a fatigue evaluation is not required if the number of stress cycles is less than 20,000.

5. It is of interest to note that the AASHTO Code allowable stress range for base metal and weld metal at full penetration groove welded splices is 29 ksi at 500,000 cycles, 38 ksi at 200,000 cycles and 49 ksi at 100,000 cycles, which is consistent with the design basis fatigue life of CNP containment liner based on fatigue tests, namely, 180,000 cycles for a stress range of 40 ksi with complete stress reversal. This fatigue life is reduced for steels containing details, which inherently entail large stress concentration. However, the lowest allowable stress range for the most severe stress concentration category (Category E' and complete stress reversal) is 2.6 ksi at 10 million cycles and 16 ksi at 100,000 cycles (Reference 4), and 32 ksi at 10,000 cycles (Reference 5). Noting that the local stress concentration around the relatively shallow and rounded pits in the corroded liner plate is less severe than in the worst stress concentration category of the AASHTO Code, a fatigue failure of the corroded liner plate is considered very unlikely in view of its low stress range of 11.3 ksi, low stress cycles of 300, and the lack of sharp notch or crack-like (planar) surface discontinuities, which are all within the limits of the most severe fatigue criteria of the AASHTO Code for welded steel bridges.

Status: The fatigue evaluation of the corroded liner does not appear to consider reduced wall thickness due to corrosion. In addition, NRC needs to understand the reference to the SNL report regarding the investigation of a 1/16-inch thick steel plate that contains pitting and corrosion. The staff is looking for additional quantitative data versus qualitative data presented in the RAI response. I&M indicated that they would provide a supplemental response.

Request for Clarification:

1. The staff requests the applicant to provide clarification regarding the statement that "I&M has modified the plant design and operations to preclude feedwater nozzle cracking from being a concern" in CNP LRA Table 4.3-1.

2. The staff requests the applicant to provide clarification regarding why the transient, "120 cycles of secondary to primary leak tests for the Mode 51R replacement steam generator," is listed in Table 4.1-10 of the UFSAR but listed in Table 4.3-1 of the CNP LRA.

3. The staff requests the applicant to provide clarification regarding why "Rector Vessel Internals" is listed in Section 4.3 but not in Section 4.3.1 of the CNP LRA.

Status: I&M indicated that they would provide a supplemental response.

Donald C. Cook Nuclear Plant, Units 1 and 2

cc:

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, IL 60532-4351

Attorney General
Department of Attorney General
525 West Ottawa Street
Lansing, MI 48913

Township Supervisor
Lake Township Hall
P.O. Box 818
Bridgman, MI 49106

U.S. Nuclear Regulatory Commission
Resident Inspector's Office
7700 Red Arrow Highway
Stevensville, MI 49127

David W. Jenkins, Esquire
Indiana Michigan Power Company
One Cook Place
Bridgman, MI 49106

Mayor, City of Bridgman
P.O. Box 366
Bridgman, MI 49106

Special Assistant to the Governor
Room 1 - State Capitol
Lansing, MI 48909

Mr. John A. Zwolinski
Director, Design Engineering and Regulatory Affairs
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

David A. Lochbaum
Nuclear Safety Engineer
Union of Concern Scientists
1707 H Street NW, Suite 600
Washington, DC 20036

Michigan Department of Environmental Quality
Waste and Hazardous Materials Div.
Hazardous Waste & Radiological Protection Section
Nuclear Facilities Unit
Constitution Hall, Lower-Level North
525 West Allegan Street
P.O. Box 30241
Lansing, MI 48909-7741

Michael J. Finissi, Plant Manager
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

Mr. Joseph N. Jensen, Site Vice President
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

Mr. Fred Emerson
Nuclear Energy Institute
1776 I Street, N.W., Suite 400
Washington, DC 20006-3708

Richard J.Grumbir
Project Manager, License Renewal
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

Mr. Mano K. Nazar
American Electric Power
Senior Vice President and Chief Nuclear Officer
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107