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INDIANA
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November 18, 2004

AEP:NRC:4034-20
10 CFR 54

Docket Nos. 50-315
50-316

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

Donald C. Cook Nuclear Plant, Units 1 and 2
LICENSE RENEWAL APPLICATION – CLARIFICATION TO REQUEST FOR ADDITIONAL INFORMATION RESPONSES
(TAC NOS. MC1202 AND MC1203)

Reference: Letter from M. K. Nazar, Indiana Michigan Power Company (I&M), to Nuclear Regulatory Commission (NRC) Document Control Desk, "Donald C. Cook Nuclear Plant Units 1 and 2, Application for Renewed Operating Licenses," AEP:NRC:3034, dated October 31, 2003 [Accession No. ML033070177].

Dear Sir or Madam:

By the referenced letter, I&M submitted an application to renew the operating licenses for Donald C. Cook Nuclear Plant (CNP), Units 1 and 2.

During the conduct of its review of this license renewal application (LRA), the NRC Staff identified areas where additional information was needed. This letter provides I&M's response to one NRC Staff request for additional information (RAI) pertaining to augmented inservice inspections of the CNP containment liners and clarifies several previously submitted RAI responses pertaining to the license renewal scoping and aging management review results. In addition, this letter provides the resolution to an LRA error that was identified during the conduct of NRC inspection activities of the aging management programs (AMPs) credited in the LRA.

The enclosure to this letter provides an affirmation pertaining to the statements made in this letter. The attachment to this letter provides the RAI response and clarifications and a discussion of the LRA error identified during the AMP inspection. There are no new commitments made in this submittal.

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Should you have any questions, please contact Mr. Richard J. Grumbir, Project Manager, License Renewal, at (269) 697-5141.

Sincerely,

Joseph N. Jensen
Site Vice President

NH/rdw

Enclosure: Affirmation

Attachment: Clarification to Request for Additional Information Responses

c: J. L. Caldwell – NRC Region III
K. D. Curry – AEP Ft. Wayne, w/o enclosure/attachment
J. T. King – MPSC, w/o enclosure/attachment
C. F. Lyon – NRC Washington DC
MDEQ – WHMD/HWRPS, w/o enclosure/attachment
NRC Resident Inspector
J. G. Rowley – NRC Washington DC

AFFIRMATION

I, Joseph N. Jensen, 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

Joseph N. Jensen
Site Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 18 DAY OF November, 2004

Notary Public

My Commission Expires 6/10/2007

Clarification to Request for Additional Information Responses

By letter dated October 31, 2003 (Reference 1), Indiana Michigan Power Company (I&M) submitted an application to renew the operating licenses for Donald C. Cook Nuclear Plant (CNP), Units 1 and 2. During the conduct of its review of this license renewal application (LRA), the Nuclear Regulatory Commission (NRC) Staff identified areas where additional information was needed. This attachment provides I&M's response to one NRC Staff request for additional information (RAI) pertaining to augmented inservice inspections of the CNP containment liners and clarifies several previously submitted RAI responses pertaining to the license renewal scoping and aging management review results. In addition, this letter provides the resolution to an LRA error that was identified during the conduct of NRC inspection activities of the aging management programs (AMPs) credited in the LRA.

RAI 3.5-9

For the aging management of containment liner, Table 3.5.2-1 (p. 3.5-28) refers basically to containment ISI (AMP B.1.15), and Containment Leakage Rate Program (AMP B.1.8). None of the programs addresses the containment liner areas subjected to Augmented Inspection requirements of IWE-1240. Please identify the inside liner plate areas (both Units) subjected to augmented inspections; specifically, address how the aging of liner plate areas behind the ice condenser is and will be monitored during the period of extended operation.

I&M Response to RAI 3.5-9:

American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI (ASME Section XI), IWE-1240, provides criteria for identifying metal containment surfaces requiring an increased level of examination per ASME Section XI, IWE-2500, and Table IWE-2500-1, Category E-C.

For the CNP containments, two areas on the inside of the liner plate were identified as potentially meeting the criteria of ASME Section XI, IWE-1240;

(1) The area of the containment liner plate surfaces behind and below the moisture barrier at the junction of the containment liner and concrete floor at elevation 598'-9", and

(2) The liner ice condenser top deck curtain support area. (See the Liner Plate Behind the Ice Condenser Structure section below for additional information regarding access to the liner plate behind the top deck vent curtain.)

These two areas were reviewed using the criteria of ASME Section XI, IWE-1240 (e.g., material and material coating, location, environment, operational conditions, industry concerns, plant examination data, and/or plant unique experience and other factors, as applicable). The review determined that augmented inspection is not warranted for either of these areas.

In accordance with the criteria of ASME Section XI, IWE-1220 (b), the areas of the containment liner plate described in items (1) and (2), above, are inaccessible and are exempt from the inspections specified in ASME Section XI, IWE-2000. The areas were determined to be acceptable for service after completion of corrective actions (liner surface preparation, recoating, floor-liner seal modification/replacement at elevation. 598') to repair containment liner corrosion/pitting, as detailed in NRC Inspection Manual Chapter 0350, Case Specific Checklist Item No. 12. The surface rust originally discovered under the top curtain support was due to the coating being removed during construction to install the vent curtain support. Negligible wall loss occurred, even after approximately 25 years of service with bare metal exposed under the supports. This uncoated liner in the vicinity of the top curtain supports was prepared and recoated prior to the closure of the Manual Chapter 0350 checklist item. No augmented examinations per ASME Section XI, Subsection IWE or per 10 CFR 50.55a were deemed applicable to this area.

Liner Plate Behind the Ice Condenser Structure

Permanently installed wall panels separating the liner from the ice baskets prevent ingress of water/moisture between the ice baskets and the liner plate. The liner plate areas behind each unit's ice condenser are inspected as part of inservice inspection through three existing 14-inch diameter circular ports. These inspections are identified in the "D. C. Cook Containment Inservice Inspection Program Plan," which is part of the ASME Section XI, Subsection IWE Program described in LRA Section B.1.15. The ports are opened and the liner visually examined every inspection period, which is three times in the ten-year inspection interval. The inspection results have not indicated the need for augmented inspection.

Other than the inspection port openings, access to the concrete liner behind the permanent ice condenser wall panels would require destructive activities, such as dismantling or modification of the wall panels, removal of divider barrier seal material, or removal/displacement of the ice condenser top deck curtain. Sections of the divider barrier seal material are removed each outage for testing (i.e., test coupons). With the divider barrier seal test coupons removed, a limited view of the liner plate exists through a small gap in the divider barrier steel. I&M implements a recurring task work order to perform a "best effort" visual examination of the liner through this small gap to identify any liner degradation.

There is a narrow gap at the top of the ice condenser in the area of the top deck doors behind the top deck vent curtain. There is limited viewing with a direct visual examination due to the size of the gap and configuration of the structure. Access to this gap and the liner plate below the vent curtain support would require the following potentially destructive activities:

(1) Special decking must be erected over the ice condenser top deck doors to access the vent curtain area. The top deck doors are designed to open on sufficiently low force to allow steam/air flow through the ice condensers without exceeding the containment design pressure. Consequently, these light weight top deck doors are susceptible to damage by work activities in their vicinity, such as staging decking on top of them. The decking materials would have to be carried and hoisted by hand due to inaccessibility with the crane.

(2) The vent curtain must be removed or displaced to access the small area of liner below the vent curtain support and the gap between the ice condenser structure and the liner.

In summary, no areas of the containment liner plate for either CNP unit require augmented examination per ASME Section XI, IWE-1240.

Additional Information Regarding RAI 2.3.3.1-1

By RAI 2.3.3.1-1, the NRC Staff requested I&M to identify a source of makeup water that is within the scope of license renewal for meeting the 10 CFR 54.4(a)(2) criteria as functionally supporting the spent fuel pool (SFP) system's intended function. I&M's response to this RAI, provided in a letter dated September 2, 2004 (Reference 6), credited fire water, which is supplied via the hose reel stations, as the in-scope source of makeup water for the SFP. The NRC Staff subsequently requested that a second in-scope source of SFP makeup water be identified to satisfy the facility's licensing basis requirement for diverse makeup water sources.

In response to this request, I&M will additionally credit refueling water from either unit's refueling water storage tank (RWST) as a second diverse in-scope source of makeup water to the SFP. The capacity of this makeup source has been evaluated and determined to exceed the maximum calculated SFP boil-off rate (Reference 2).

The additional components included in the license renewal scope are depicted on license renewal drawing LRA-12-5136 and include those stainless steel components (such as valves, pump, orifices, demineralizer, filters, piping, and pipe appurtenances) in the flow path from the RWST isolation valves to the SFP. The makeup flow path from the RWST isolation valves (1-SI-183 and 2-SI-184 at location F9) includes the refueling water purification pump (at location D9) and the spent fuel pit filter (at location N5) and terminates at the SFP (at location E4). Neither the spent fuel pit demineralizer (at location A4) nor the refueling water purification filter (at location B8) is in the makeup flow path; however, these components are included because they provide a pressure boundary function. Note that the components in this flow path are included in the scope of license renewal in accordance with the 10 CFR 54.4(a)(2) scoping criterion; therefore, they have not been highlighted on the license renewal drawing.

The aging effects requiring management for these stainless steel components exposed to an internal treated borated water environment include loss of material and cracking, and will be managed by AMPs such as the System Walkdown Program and Water Chemistry Control Program, as listed in LRA Table 3.3.2-11 for the affected stainless steel components in LRA Table 3.3.2-11 (Valve, Pump casing, and Piping) exposed to an internal treated borated water environment.

LRA Table 3.3.2-11 is modified to include loss of material and cracking of the stainless steel component types Orifice, Demineralizer, and Filter exposed to an internal treated borated water environment managed by the System Walkdown Program and Water Chemistry Control Program AMPs. In addition to the AMPs credited, effectiveness of the Water Chemistry Control Program will be verified through the Chemistry One-Time Inspection Program, described in LRA Section B.1.41, consistent with I&M's response to RAI B.1.41-2 in the referenced October 18, 2004, letter (Reference 5).

The first paragraph of LRA Section 2.3.3.11, *Spent Fuel Pool Cooling*, on Page 2.3-82 is modified as provided below. New text is *italicized*, as follows:

> The purpose of spent fuel pool cooling is to remove, from the spent fuel pool, the heat generated by stored fuel elements. The components of the CNP spent fuel pool cooling provide no *10 CFR 54.4(a)(1) or 10 CFR 54.4(a)(3)* intended functions. The maintenance of pool inventory, which assures cooling, is provided by the spent fuel pit as discussed in Section 2.3.3.1. *Those components in the flow path from the RWST isolation valves (1-SI-183 and 2-SI-184) to the SFP are credited as one of the diverse sources of makeup water to the SFP, and perform a 10 CFR 54.4(a)(2) intended function.* The spent fuel pool is shared by the two units. The design incorporates two separate cooling trains sharing a common return to the spent fuel pool. Piping is arranged so that failure of any pipe does not drain the spent fuel pool below the top of the stored fuel elements.

Additional Information Regarding RAI 2.1-5

The NRC Staff requested I&M to clarify whether piping thermal insulation serves an intended function in accordance with 10 CFR 54.4, and to provide the basis for the determination as to whether this piping thermal insulation requires aging management. In an RAI response dated August 11, 2004 (Reference 3), I&M stated that piping insulation is not in scope and not subject to aging management review, except in certain specific applications where the insulation is required to maintain post-accident temperature in areas housing safety-related equipment. The NRC Staff has indicated a need for additional information regarding the review of piping thermal information. Based on a conference call with the NRC Staff on November 10, 2004, I&M understands that the specific information required pertains to the review of the piping thermal insulation at containment penetrations.

Thermal insulation on hot piping at containment penetrations does not meet the scoping criteria of 10 CFR 54.4. The insulation on hot containment piping penetrations is not required to ensure the functions of 10 CFR 54.4(a)(1) are accomplished or to demonstrate compliance with NRC regulations identified in 10 CFR 54.4(a)(3). The insulation does not meet 10 CFR 54.4(a)(2), as its failure will not prevent satisfactory accomplishment of any of the functions identified in 10 CFR 54.4(a)(1). Insulation on hot piping at containment penetrations does support maintaining the environment for surrounding structural elements. However, maintaining the environment during normal operation is not an intended function identified in 10 CFR 54.4(a)(1). Therefore, thermal insulation on hot piping at containment penetrations does not meet the scoping criteria of 10 CFR 54.4. This is consistent with the previously approved staff position documented in NUREG-1766, *Safety Evaluation Report Related to the License Renewal of North Anna Nuclear Station, Units 1 and 2, and Surry Nuclear Power Station, Units 1 and 2.*

Notwithstanding the above, I&M agrees to include the insulation on hot containment piping penetrations in the scope of license renewal. The intended function that was applied to the insulation is to prevent excessive heat transmission to the containment concrete surrounding the piping penetrations. The insulation is encapsulated with stainless steel jacketing in the annulus between the penetration piping and the penetration sleeve. There are no applicable aging effects for insulation in the indoor air environment. A review of CNP operating experience for the past five years verified that the plant has not experienced aging-related degradation of piping insulation in dry indoor environments. Therefore, based upon the material, environment, and operating experience, the insulation is not expected to degrade, and an AMP is not required.

Additional Information Regarding RAI 2.4-2

In RAI responses dated May 7, 2004, and August 11, 2004 (References 4 and 3, respectively), I&M stated that the divider barrier seals and divider barrier hatch seals are short-lived and not subject to aging management review because they are periodically inspected in accordance with Technical Specifications 4.6.5.9 and 4.6.5.5.2, respectively. I&M's position was based on NRC statements regarding NUREG-1800, Section 2.1.3.2.2, "Long-Lived," in the Statements of Consideration for the Final License Renewal Rule. Upon further consideration, I&M has determined that it would be more appropriate to credit inspection activities that implement plant Technical Specification requirements as the license renewal AMP for the divider barrier seals and divider barrier hatch and personnel access seals than to credit the Technical Specification requirements as the basis for exclusion from aging management review.

To implement this change, the Updated Final Safety Analysis Report (UFSAR) supplement for the Structures Monitoring - Divider Barrier Seal Inspection Program provided in LRA Sections A.2.1.37 and the affected program element descriptions from LRA Section B.1.34 are modified as provided below. New text is *italicized,* as follows:

A.2.1.37 STRUCTURES MONITORING – DIVIDER BARRIER SEAL INSPECTION PROGRAM

The Divider Barrier Seal Inspection Program detects cracking and change in material properties of *the* elastomeric *divider barrier seals, divider barrier hatch, and personnel access door seals, and* pressure seals for penetrations and openings through the containment divider barrier. The program detects aging effects through *analysis of main divider barrier seal test coupons and* visual examination of the *three types of* seals *between the upper and lower containment compartments.*

B.1.34 Structures Monitoring – Divider Barrier Seal Inspection Program

Program Description

The Divider Barrier Seal Inspection Program is an existing plant-specific program. There is no comparable NUREG-1801 program.

The divider barrier in each containment is the physical boundary that separates upper containment from lower containment. Several containment internal structures constitute the divider barrier. Elastomeric seals are provided for *the divider barrier that separates the upper and lower containment compartments; and for the personnel access doors, equipment hatches, and* penetrations and openings through the divider barrier where it is necessary to limit potential ice condenser bypass leakage subsequent to a postulated pipe rupture or loss of coolant accident. Cracking and change in material properties are aging effects requiring management for the pressure seals.

Aging Management Program Elements

Scope

The scope of this program is the *elastomeric* containment divider barrier *seals; and* elastomeric *hatch seals, personnel access door seals, and* pressure seals around penetrations and openings through the divider barrier.

Parameters Monitored or Inspected

Parameters monitored by this program are cracking and change in material properties of elastomeric pressure seals.

Detection of Aging Effects

This program detects cracking and change in material properties prior to loss of the pressure seals' intended functions.

In accordance with plant Technical Specifications, (1) the physical properties of the main divider barrier seals are periodically verified through analysis of seal test coupons, and (2) visual inspections of the divider barrier seals and hatch and personnel access door seals are performed to identify apparent deterioration of the seal material. The seals around penetrations and openings (including the bulkhead gate) are visually inspected to ensure the absence of apparent deterioration (cracks or defects). The frequency of *the penetration and openings seals* inspection is at least once every 10 years.

Monitoring and Trending

This program monitors aging effects through *analysis of main divider barrier seal test coupons and* visual examination of the *other* seals. The Corrective Action Program provides reasonable assurance that trends entailing repeat failures to meet acceptance criteria will be identified and addressed with appropriate corrective actions.

Acceptance Criteria

The acceptance criteria for the divider barrier seal test coupons is provided in plant Technical Specifications. The acceptance criteria for *visual* seal inspections are that seals must be free of unacceptable deterioration (excessive cracks or defects) and unacceptable misalignment.

Corrective Actions

Discrepancies noted during the inspection are documented in the Corrective Action Program in accordance with the implementing procedure. Specific corrective actions will be implemented in accordance with the CNP Corrective Action Program. *Required actions for failure to meet the Technical Specification surveillance requirements applicable to the main divider barrier seals* *and divider barrier hatch and personnel access door seals are provided in the plant Technical Specifications.*

In addition, in LRA Tables 2.4-1, on Page 2.4-16 and 3.5.2-1, on Page 3.5-40, the Component entry "Removable gate (bulkhead) seals" is modified to read as follows:

Main divider barrier seals
Divider barrier hatch seals
Personnel access door seals
Removable gate (bulkhead) seals

Additional Information Regarding RAI 2.4-5

In response to an NRC staff request pertaining to activities credited to ensure the cable feed-through assemblies will perform their pressure-retaining function throughout the period of extended operation, I&M provided, in a letter dated August 11, 2004 (Reference 3), a discussion that credited the electrical penetration pressure testing that was performed to satisfy the 10 CFR 50.49 environmental qualification (EQ) requirements and 10 CFR 50 Appendix J containment leakage rate testing. Subsequently, the NRC staff requested that I&M specifically credit 10 CFR 50 Appendix J, Type B testing (or any other AMP) for license renewal, to ensure the pressure boundary integrity of cable feed-through assemblies. The NRC staff further requested that I&M identify the AMP that it is crediting for license renewal.

Pressure testing during environmental qualification of the electrical containment penetrations provides assurance that the pressure boundary integrity of cable feed-through assemblies will be maintained during the period of extended operation. In addition, the Containment Leakage Rate Testing Program, as described in LRA Section B.1.8, is credited with managing the effects of aging on containment electrical penetrations throughout the period of extended operation by providing assurance that leakage through these penetrations does not exceed allowable values. Local leakage rate testing (defined as "Type B testing" in 10 CFR 50 Appendix J), performed under the Containment Leakage Rate Testing Program, provides ongoing confirmation of the integrity of resilient seals around the perimeter of the cable feed-through assemblies. Integrated leakage rate testing (defined as "Type A testing" in 10 CFR 50 Appendix J), performed under the Containment Leakage Rate Testing Program, provides additional confirmation of pressure boundary integrity of the feed-through assemblies.

Additional Information Regarding RAI 2.4-3

In letters dated May 7, 2004, and August 11, 2004 (References 4 and 3, respectively), I&M provided responses to RAI 2.4-3, regarding the structures and components that are essential to ensure the availability of cooling water for safe shutdown and perform an intended function per 10 CFR 54.4(a). Subsequently, the NRC Staff requested the following information:

- Determine whether the trash baskets and associated trash collection equipment are essential to ensure availability of cooling water for safe shutdown [RAI 2.4-3, Part (3)].
- Describe the physical location and function of the discharge jets and specifically indicate whether they are essential to ensure availability of cooling water for safe shutdown and perform an intended function per 10 CFR 54.4(a) [RAI 2.4-3, Part (4)].
- Identify the components that are relied on for de-icing to ensure an adequate supply of cooling water for safe shutdown, and verify that all of these components are included in the license renewal scope.

Part (3) of the response to RAI 2.4-3 did not address the trash baskets and associated trash collection equipment used for trash collection. Trash collection equipment is used to collect the trash that is removed from the traveling screens by the screen wash system and direct it to the trash baskets. After being filled, the baskets are used to transport the trash for disposal. The trash baskets and associated trash collection equipment are not in the flow path for water entering the screen house and providing suction to the essential service water (ESW) system. Failure of the trash baskets and associated trash collection equipment would not impact the ability to provide water to the ESW system. Therefore, they do not meet the scoping criteria of 10 CFR 54.4.

Part (4) of the response to RAI 2.4-3 did not clearly address the "discharge jets." The discharge jets are at the end of the discharge piping in the lake and act as a diffuser to direct flow away from the intake pipes and distribute the water so as to minimize the environmental effects of the warm water. The discharge jets are located downstream of the discharge corrugated piping and the discharge elbows shown on UFSAR Figure 10.6-1. As discussed in I&M's supplemental response to RAI 2.4-3, included in the August 11, 2004, letter (Reference 3), the discharge piping and discharge elbows are not relied upon to ensure the availability of cooling water to the ESW pumps. Therefore, the discharge piping, elbows, and jets do not meet the 10 CFR 54.4 scoping criteria.

The operation of sluice gates and roller gates is needed to establish the flow path for de-icing. De-icing is not credited for emergency operation. Icing is a concern only for the higher flow rates associated with power operation. The concern during power operation is flow restriction caused by ice fouling the traveling screens. The flow restriction can result in the circulating water system being unable to provide the flow necessary to support power operation. During emergency operation, required flow is a small fraction (approximately one percent) of total circulating water capacity. At the significantly lower flow rate required for emergency operation, ice fouling of the traveling screens will not prevent the required flow from reaching the suction of the ESW system pumps. During emergency operation, de-icing is not required to assure the availability of the cooling water supply to the ESW system. The mechanical components credited to ensure an adequate supply of cooling water for safe shutdown during cold weather operation are the ESW pumps, intake piping, strainers, and valves.

Additional Information Regarding RAI 3.3.2.1.11-1

Clarification Requested by the NRC Staff:

In the I&M Supplemental Response to RAI 3.3.2.1.11-1 [Reference 5], the applicant states, "The remaining systems (CONT, DRAIN, PASS, RMS, RWD, and SD) have copper alloy, carbon steel, stainless steel, or glass components that may be pressurized and contain raw water or untreated water. As discussed above, glass exposed to a raw or treated water environment has no aging effect requiring management. I&M will include components containing raw or untreated water subject to aging management review that were included for 10 CFR 54.4(a)(2) in these systems in the Chemistry One-Time Inspection Program." For components in a raw or untreated water environment where an aging effect is likely to occur, the GALL report recommends use of a mitigative program with a one-time inspection program to verify the program effectiveness or the use of periodic inspections. For components in these systems, it is not clear to the staff if the applicant is crediting a mitigative program in conjunction with the one-time inspections program or if a one-time inspection alone is credited to manage these aging effects.

Clarification to I&M's Supplemental Response to RAI 3.3.2.1.11-1:

The Chemistry One-Time Inspection Program, in conjunction with the System Walkdown Program and Corrective Action Program, is credited with managing aging effects of the containment (CONT), process drains (DRAIN), post-accident sampling (PASS), radiation monitoring (RMS), radioactive waste disposal (RWD), and station drainage (SD) systems. Although the water source to most of the drain lines associated with these systems is treated water, the internal environment of these systems is conservatively considered raw or untreated water and therefore, the Water Chemistry Control Program cannot be credited.

The Chemistry One-Time Inspection Program will be limited to inspections of a specific material-environment combination in the above systems. If indications that age-related degradation is sufficiently slow such that the component intended function will be maintained throughout the extended period of operation, then no further action will be required. However, if unacceptable age-related degradation is observed, the results of the initial inspection will be used to determine the scope and frequency of subsequent examinations. Appropriate corrective actions will be determined in accordance with the Corrective Action Program and may consist of component refurbishment, repair, or replacement.

I&M finds this program acceptable because:

(1) The components in these systems are associated with drainage and liquid waste, which are low pressure (generally atmospheric) systems. Consequently, a large corrosion allowance is available, and any leakage that would occur would be localized.

(2) Most of the carbon steel components are not continually exposed to fluids. Most of the components that are continually exposed to fluid are composed of stainless steel, which is inherently resistant to corrosion.

(3) For stainless steel and copper components in these systems, the inherent corrosion resistance of these materials results in slow degradation.

(4) A review of condition reports concerning drains and corrosion written during the past five years did not identify conditions that are indicative of abnormal corrosion in these systems after more than 25 years of operation.

In conclusion, the Chemistry One-time Inspection Program, combined with the System Walkdown Program and Corrective Action Program, will provide assurance that loss of material is occurring at a rate slow enough to ensure that the intended functions of these 10 CFR 54.4 (a)(2) components will be maintained for the period of extended operation. This is consistent with previously approved NRC Staff positions documented in NUREG-1787, *Safety Evaluation Report Related to the License Renewal of Virgil C. Summer Nuclear Station,* and NUREG-1796, *Safety Evaluation Report Related to the License Renewal of the Dresden Nuclear Power Station, Units 2 and 3, and Quad Cities Nuclear Power Station, Units 1 and 2.*

Additional Information Regarding RAI 3.4-4

Clarification Requested by the NRC Staff:

Provide a basis for not managing loss of pre-load for bolting within the scope of license renewal.

I&M's Supplemental Response to RAI 3.4-4:

Normal maintenance practices intended to preclude loss of pre-load are applied to pressure-retaining bolted connections when they are assembled. These maintenance practices are proceduralized and include proper torque selection, torque patterns, thread engagement criteria, and inspection of bolting materials including gaskets.

Clarification to Supplemental Response to RAI 3.6-2

In supplemental responses to RAIs 3.6-2 and 3.6-5, submitted by letters dated October 18, 2004, and September 2, 2004 (References 5 and 6, respectively), I&M revised LRA Section A.2.1.24 for the Non-EQ Instrumentation Circuits Test Review Program. In an October 19, 2004, teleconference, the NRC Staff identified that the proposed text for the supplemental response to RAI 3.6-2 did not reflect the changes from the earlier response to the RAI 3.6-5 response. In response to this issue, the following revised LRA Section A.2.1.24 is provided, combining the text provided in both RAI responses. New text is *italicized*, as follows:

[Please note that the sentence discussing the frequency at which calibration or surveillance results are reviewed has been revised to reflect a grammatical correction. This sentence, which previously specified that these reviews are performed "at a frequency not to exceed 10 years," has been revised to state, "...at a frequency of not less than once per ten years..." This change does not affect the responses to either supplemental RAI response.]

Combined Section A.2.1-24

A.2.1.24 NON-EQ INSTRUMENTATION CIRCUITS TEST REVIEW PROGRAM

The Non-EQ Instrumentation Circuits Test Review Program will manage aging effects for electrical cables that:

- Are not subject to the environmental qualification requirements of 10 CFR 50.49, and
- Are used in instrumentation circuits with sensitive, high-voltage, low-level signals, *such as radiation monitoring and nuclear instrumentation, which are* exposed to adverse localized environments caused by heat, radiation, or moisture.

An adverse localized environment is defined as being significantly more severe than the specified service environment for the cable. This program will detect aging effects by reviewing calibration or surveillance results for components within the program scope *at a frequency of not less than once per ten years or as part of corrective actions when acceptance criteria are exceeded at the normal calibration frequency. A proven cable test for detecting insulation deterioration on in-scope instrumentation cables that are disconnected during calibration will be performed at a frequency determined by engineering evaluation, but will not be less than once per ten years.* The Non-EQ Instrumentation Circuits Test Review Program will be implemented prior to the period of extended operation.

Correction to LRA Table 3.4.2-3

During an NRC special inspection of the license renewal AMPs, it was discovered that some material entries associated with auxiliary feedwater system piping in LRA Table 3.4.2-3, on pages 3.4-28 and 3.4-29, are incorrect. The incorrect table entries resulted from a formatting error made during final preparation of this LRA table. A comprehensive extent of condition review of other LRA tables determined this to be an isolated incident.

To correct this error, the following changes are made to LRA Table 3.4.2-3:

(1) On LRA Page 3.4-28, all material is "Carbon steel." This change affects the "Treated water (internal)" environment line items on this page. Information in the remaining columns for the affected line items is correct for carbon steel.

(2) On LRA Page 3.4-29, the material subjected to the "Treated water >270°F (internal)" environment is "Carbon steel" vice "Stainless steel." Information in the remaining columns for the affected line items is correct for carbon steel. The remaining two environments for piping on this page [i.e., Outdoor air (external) and Treated water (internal)] apply to stainless steel piping.

References

1. Letter from M. K. Nazar, I&M, to NRC Document Control Desk, "Donald C. Cook Nuclear Plant Units 1 and 2, Application for Renewed Operating Licenses," AEP:NRC:3034, dated October 31, 2003 [Accession No. ML033070177].
2. Letter from M. W. Rencheck, I&M, to NRC Document Control Desk, "Response to Nuclear Regulatory Commission Request for Additional Information Regarding the Decay Time License Amendment Request (TAC Nos. MB1975 and MB1976)," C0901-03, dated September 5, 2001 [Accession No. ML012530168].
3. Letter from J. N. Jensen, I&M, to NRC Document Control Desk, "Donald C. Cook Nuclear Plant, Units 1 and 2, Docket No. 50-315 and 50-316, License Renewal Application – Supplemental Responses to Requests for Additional Information (TAC Nos. MC 1202 and MC 1203)," AEP:NRC:4034-11, dated August 11, 2004 [Accession No. ML042470402].
4. Letter from M. K. Nazar, I&M, to NRC Document Control Desk, "Donald C. Cook Nuclear Plant, Units 1 and 2, Docket No. 50-315 and 50-316, License Renewal Application – Response to Requests for Additional Information on Scoping and Screening Results (TAC Nos. MC 1202 and MC 1203)," AEP:NRC:4034-01, dated May 7, 2004 [Accession No. ML041390360].
5. Letter from J. N. Jensen, I&M, to NRC Document Control Desk, "Donald C. Cook Nuclear Plant, Units 1 and 2, Docket No. 50-315 and 50-316, License Renewal Application – Response to Requests for Additional Information (TAC Nos. MC 1202 and MC 1203)," AEP:NRC:4034-17, dated October 18, 2004 [Accession No. ML042960028].
6. Letter from J. N. Jensen, I&M, to NRC Document Control Desk, "Donald C. Cook Nuclear Plant, Units 1 and 2, Docket No. 50-315 and 50-316, License Renewal Application – Supplemental Responses to Requests for Additional Information (TAC Nos. MC 1202 and MC 1203)," AEP:NRC:4034-15, dated September 2, 2004 [Accession No. ML042530551].