INDIANA MICHIGAN POWER

**Indiana Michigan Power Company**
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106
aep.com

August 21, 2009

AEP-NRC-2009-52
10 CFR 50.55a

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

SUBJECT: Donald C. Cook Nuclear Plant Unit 1
Docket No. 50-315
Third and Fourth Ten-Year Interval Inservice Inspection Program Relief Request ISIR-31

Dear Sir or Madam:

Pursuant to 10 CFR 50.55a(a)(3)(ii), Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant Unit 1, hereby requests Nuclear Regulatory Commission approval of the following request for the Unit 1 third and fourth ten-year interval inservice inspection program:

Relief Request ISIR-31 proposes visual examination during system leakage test of the pressurizer surge nozzle as an alternative to the American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section XI, Table IWB-2500-1 volumetric examination requirement or the enhanced visual examination requirement of 10 CFR 50.55a(b)(2)(xxi), which may be performed instead of an ultrasonic examination. Compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. The details of the 10 CFR 50.55a requests are enclosed.

I&M requests approval by February 3, 2010, to allow use of the alternative during the Unit 1 Cycle 23 refueling outage.

There are no new or revised commitments in this letter. Should you have any questions, please contact Mr. James M. Petro, Jr., Regulatory Affairs Manager, at (269) 466-2489.

Sincerely,

Quinton A. Lies
Q. Shane Lies for Ray Hruby

Raymond A. Hruby, Jr.
Vice President - Site Support Services

RSP/rdw

Enclosure: Relief Request ISIR-31

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NRR

c: T. A. Beltz – NRC Washington, DC
J. T. King – MPSC
S. M. Krawec, Ft. Wayne AEP
MDEQ – WHMD/RPS
NRC Resident Inspector
M. A. Satorius – NRC Region III

10 CFR 50.55a Relief Request Number ISIR-31

**Proposed Alternative**
**In accordance with 10 CFR 50.55a(a)(3)(ii)**

--Hardship or Unusual Difficulty
Without a Compensating Increase in Level of Quality or Safety--

1. **<u>American Society of Mechanical Engineers (ASME) Code Component Affected</u>**
Code Component Affected: 14"-1-RC-5-IRS
Description: Unit 1 Pressurizer Surge Line Nozzle Inner Radius Sections
Item Number: B-3.120
Code Class: 1

2. **<u>Applicable Code Edition and Addenda</u>**
ASME Boiler and Pressure Vessel Code (ASME Code) Section XI, 1989 Edition, with no Addenda.

3. **<u>Applicable Code Requirement</u>**
Table IWB-2500-1, Examination Category B-D, Inspection Program B, Item No. B3.120 requires volumetric examination of pressurizer nozzle inner radius section.

Code of Federal Regulations, Paragraph 10 CFR 50.55a(b)(2)(xxi) permits enhanced visual examination of the interior surface instead of ultrasonic examination of the exterior surface as stated below.

> (xxi) Table IWB-2500-1 examination requirements. "(A) The provisions of Table IWB-2500-1, Examination Category B-D, Full Penetration Welded Nozzles in Vessels, Items B3.40 and B3.60 (Inspection Program A) and Items B3.120 and B3.140 (Inspection Program B) of the 1998 Edition must be applied when using the 1999 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section. A visual examination with magnification that has a resolution sensitivity to detect a 1-mil width wire or crack, utilizing the allowable flaw length criteria in Table IWB-3512-1, 1997 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section, with a limiting assumption on the flaw aspect ratio (i.e., a/l = 0.5), may be performed instead of an ultrasonic examination."

4. **<u>Reason for Request</u>**
In accordance with 10 CFR 50.55a(a)(3)(ii), an alternative is proposed on the basis that compliance with the code requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. The difficulty associated with access to either the exterior surface for the volumetric examination or to the interior surface for the enhanced visual examination is described below.

The Donald C. Cook Nuclear Plant (CNP) Unit 1 pressurizer surge nozzle is integrally cast within the pressurizer lower head. To perform the volumetric examination of the surge nozzle inner radius section, the outside surface of the lower head must be accessible. This surface is made accessible by removing the insulation surrounding the surge nozzle. The design of this insulation requires disconnecting the 78 heater cables from the immersion heaters prior to removing the insulation. Each cable consists of two wires, each mechanically connected to the heater. Care must be taken while disconnecting and reconnecting to ensure the ceramic terminal blocks connecting the wires to the heater pins are not damaged. If damaged, an unbrazing/brazing evolution would be required to replace the blocks. The dose estimate for this examination assumes that no ceramic terminal blocks would require replacement. Another concern is the presence of asbestos in the cable jackets. Special monitoring and material control would be necessary due to the presence of friable asbestos. Though airborne radioactive contamination is not typically a concern in this area, respirators would be required to address the potential asbestos exposure. Additional coveralls, suitable for asbestos work, would be required over the anti-contamination clothing, causing a heat stress concern.

The dose estimate below is based on a radiological survey obtained on performing a similar evolution during U2C18, Unit 2 spring 2009 refueling outage. The hours estimated to perform the activities involved in disconnecting and reconnecting the cables are based on similar efforts performed during the Unit 2 refueling outage (U2C18) when cables and insulation were removed for a visual examination. Therefore, the estimated times used in the dose estimate are believed to be reasonably accurate.

The dose estimate for cable disconnection/reconnection, insulation removal/ reinstallation, surface preparation, and ultrasonic examination is 2.495 person-Rem. Indiana and Michigan Power Company (I&M) expect similar levels for Unit 1 based on plant configuration and shutdown conditions. The dose estimate is based on a radiological survey conducted with the insulation installed and Unit 2 experience. Once the insulation is removed the dose rates documented in the survey would likely increase. Temporary shielding at this location is not practical since the radiation source is the component surface to be examined.

DOSE ESTIMATE FOR THE PRESSURIZER SURGE NOZZLE INNER RADIUS SECTION VOLUMETRIC EXAMINATION

| **Activity** | **Crew Size** | **Time** | **Dose rate** | **Estimate** |
|---|---|---|---|---|
| | | Hours (hr) | Effective Dose Rate milliRem/hour | (mRem) |
| **ELECTRICIANS:** | | | | |
| Disconnect heater cables | 2 | 2.3 | 31 mR/hr | 142 |
| Reconnect heater cables | 2 | 6 | 31mR/hr | 372 |
| Task Total | | | | 514 |
| **INSULATORS:** | | | | |
| Remove insulation | 2 | 2.9 | 48.1 mR/hr | 279 |
| Reinstall insulation | 2 | 5.7 | 48.1 mR/hr | 546 |
| Remove convection covers | 2 | 3.6 | 31 mR/hr | 226 |
| Reinstall convection covers | 2 | 8.8 | 31 mR/hr | 546 |
| Task Total | | | | 1,597 |
| **BOILERMAKERS:** | | | | |
| Weld preparation | 2 | 2 | 48.1 mR/hr | 192 |
| Task Total | 2 | 2 | | 192 |
| **EXAMINERS:** | | | | |
| Perform exam | 2 | 2 | 48.1 mR/hr | 192 |
| Task Total | 2 | 2 | | 192 |
| | | | | |
| Job Total: | | | | 2,495 |

A remote visual examination from the inside of the pressurizer was evaluated as an alternative to the ultrasonic examination. A screen is located at the surge line nozzle and baffle plates in the lower section of the pressurizer which restricts access to the area of interest. The surge nozzle contains a thermal sleeve that would significantly limit the examination. The heater support structure creates interference for remote equipment and creates a potential entanglement. The distance from the manway to the surge nozzle area is approximately 40 feet, making positioning adjustments of the remote camera difficult. Because of these limitations, a remote visual examination from the inside of the pressurizer is not considered a viable alternative.

The following describes the basis for concluding that there is no compensating increase in the level of quality and safety. There are no credible failure mechanisms other than fatigue for the surge nozzle. Corrosion degradation protection is provided by the combination of the austenitic stainless steel cladding of the surge nozzle inner radius and by the chemistry controls on the reactor coolant system. Strict chemistry standards are maintained to ensure a non-corrosive environment. Oxygen, chloride, fluoride, and other contaminant concentrations are maintained below the thresholds known to be conducive to stress corrosion cracking. Since the surge nozzle is cast, the typical failure mechanisms associated with weld material do not apply to this examination. Erosion and erosion/corrosion degradation is not credible at this location since the austenitic stainless steel cladding resists this mechanism. There is relatively low fluid velocity in the surge nozzle and reactor coolant chemistry minimizes the amount of particles in the fluid that could potentially cause erosion.

I&M has considered fatigue degradation in this area due to the potential thermal cycling caused by the insurge and outsurges of the reactor coolant flow. Since the surge nozzle is cast with the bottom head, there is no nozzle to vessel weld. The inner radius is believed to be less susceptible to fatigue problems than a nozzle to vessel weld. Initiation of fatigue cracking may have equal potential at the inner radius as compared to a nozzle to vessel weld, but a pre-existing flaw is less likely in the inner radius casting than at a nozzle to vessel weld due to the manufacturing process. For this hypothetical flaw to be present, it had to have been overlooked by the shop nondestructive examinations, which include surface, ultrasonic, and radiographic examination. Inservice fatigue crack growth for such a flaw would be small since the pressurizer is hot during the insurges and outsurges resulting in relatively high fracture toughness of the material. A thermal sleeve installed in the surge nozzle provides a measure of protection from the affects of fluid temperature changes. Examinations are performed on the nozzle to safe-end weld, which is within 18 inches of the inner radius.

This area was examined in the first and second ten-year inservice inspection interval and no indications were reported. The other five top-head pressurizer inner radius sections have been successfully examined. No recordable indications were noted on these examinations. CNP Units 1 and 2 have over 50 years of combined operation, with no leakage in the surge nozzle area. The CNP Unit 2 surge nozzle inner radius section was ultrasonically examined in the interval and no recordable indications were found. The CNP Unit 1 surge nozzle inner radius section was ultrasonically examined in the first and second interval. No evidence for need-of-repair was noted on these examinations.

There are several methods available to detect a leak should a throughwall leak occur. Listed below are some examples:

- Control room operators perform Operation Surveillance Test "Reactor Coolant System Water Inventory Balance," every 72 hours with the plant operating at steady state conditions. Leakage would be discovered by this test.

- Containment airborne radiation monitors continuously sample the containment atmosphere and alarm in the control room if a leak were to occur. This detection method is dependent on the size of the leak, reactor coolant activity, and containment background activity.

- Leakage would cause an increase in containment pressure, temperature and humidity, which are indicated in the control room. Additionally, high containment pressure produces an alarm in the control room.

- Substantial leakage would collect in the containment sump. The sump level is indicated and alarmed in the control room.

The radiation exposure associated with this examination is considered a hardship. If performed, the examination would not significantly increase the level of quality and safety due to the low probability of the presence of a flaw in this area and available leak detection methods as discussed above.

5. **Proposed Alternative and Basis for Use**

As an alternative to the applicable code examination requirements cited above, visual examination (VT-2) of the pressurizer surge nozzle external surface area with the insulation installed will be completed in conjunction with the boric acid walk-down, performed every shutdown. Also, this area is included and documented in the Mode 3 walk-down of the reactor coolant system boundary, performed during each startup following refueling outages, as required by Table IWB-2500-1, Examination Category B-P, Item No. B15.10. Both of these activities are performed by qualified VT-2 examiners. These examinations are augmented by the leakage detection methods noted above.

6. **Duration of Proposed Alternative**

The proposed alternative is applicable to the third and fourth ten-year inservice inspection interval for CNP Unit 1.

7. **Precedent**

Letter from Nuclear Regulatory Commission (NRC), to Kewaunee Nuclear Power Plant, "Kewaunee Nuclear Power Plant – Fourth 10-Year Inservice Inspection Interval Program Requests for Relief," dated February 18, 2005, ADAMS Accession Number ML050350225.

Letter from NRC, to Virginia Electric and Power Company, "North Anna Power Station, Unit 2 – ASME Section XI Inservice Inspection (ISI) Program Third 10-Year Interval Requests for Relief," dated June 12, 2002, ADAMS Accession Number ML021630245.