Indiana Michigan Power Company
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INDIANA MICHIGAN POWER

October 12, 2001

C1001-08

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
REVISED RESPONSE TO
NUCLEAR REGULATORY COMMISSION (NRC) BULLETIN 2001-01:
CIRCUMFERENTIAL CRACKING OF REACTOR PRESSURE VESSEL
HEAD PENETRATION NOZZLES
(TAC numbers MB2624 and MB2625)

Reference:
1) Letter from M. W. Rencheck (I&M) to NRC Document Control Desk, "Nuclear Regulatory Commission (NRC) Bulletin 2001-01: Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles, (TAC numbers MB2624 and MB2625)" C0801-20, dated September 4, 2001.

2) Letter from A. Christopher Bakken III (I&M) to NRC Document Control Desk, "Proposed Alternative to the Provisions of the ASME OM Code, Part 1, for Pressure Relief Device Testing," C0901-02, dated September 21, 2001.

This letter revises I&M's response to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," to correct an administrative error.

I&M provided its response to NRC Bulletin 2001-01 in Reference 1. The response to Request 2.c on Page 7 of Attachment 1 to Reference 1 erroneously indicated that the next Unit 2 refueling outage would occur in 2001. The response to Request 2.c has been revised to eliminate reference to the year 2001.

A088

The original submittal responded to the required paragraphs for operation beyond December 31, 2001. As such, no other changes are required.

At the time Reference 1 was submitted, Donald C. Cook Nuclear Plant (CNP) Units 1 and 2 had recently entered an extended unscheduled outage. As a result of uncertainties associated with the duration of the outage, I&M was considering postponing the Unit 2 refueling outage, which had been scheduled to begin on November 3, 2001. Accordingly, I&M intended that reference to a specific date for the next Unit 2 refueling outage be eliminated from Reference 1. However, due to an administrative oversight, reference to the 2001 outage was inadvertently retained in the response to Request 2.c.

The postponement of the Unit 2 refueling outage was verbally communicated to the NRC staff. Additionally, the new outage start date, January 2002, was identified to the NRC in a proposed alternative to the ASME Code submitted on September 21, 2001 (Reference 2).

Attachment 1 to this letter provides the revised response to NRC Bulletin 2001-01. Page 7 contains the changed portion of the text, indicated by a revision bar in the right margin. Attachment 2 contains a list of commitments made in this letter.

Should you have any questions, please contact Mr. Ronald W. Gaston, Manager of Regulatory Affairs, at (616) 697-5020.

Sincerely,

M. W. Rencheck
Vice President Nuclear Engineering

/jen

c: J. E. Dyer
MDEQ – DW & RPD, w/o attachment
NRC Resident Inspector
R. Whale, w/o attachment

AFFIRMATION

I, Michael W. Rencheck, being duly sworn, state that I am 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

MW Rencheck

M. W. Rencheck
Vice President Nuclear Engineering

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 12th DAY OF October, 2001

Julie E. Newmiller
Notary Public

My Commission Expires 8-22-04

JULIE E. NEWMILLER
Notary Public, Berrien County, MI
My Commission Expires Aug 22, 2004

ATTACHMENT 1 TO C1001-08

NUCLEAR REGULATORY COMMISSION (NRC) BULLETIN 2001-01:
CIRCUMFERENTIAL CRACKING
OF REACTOR PRESSURE VESSEL HEAD PENETRATION

References:

1) Electric Power Research Institute – Materials Reliability Program Report MRP-44, Part 2, titled, "PWR Materials Reliability Program, Interim Alloy 600 Safety Assessments for US PWR Plants (MRP-44)," dated May 2001.

2) Electric Power Research Institute – Materials Reliability Program Report MRP-48, titled "PWR Material Reliability Program Response to NRC Bulletin 2001-01, MRP-48," dated August 2001.

3) Nuclear Energy Institute Letter, "NRC Staff Questions on EPRI Interim Report TP-1001491, Part 2, Section 4.0, Comment No. 2," dated July 31, 2001.

4) NRC Generic Letter 97-01, "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations," dated April 1, 1997.

5) Westinghouse WCAP-14118, Revision 1, "Structural Integrity Evaluation of Reactor Vessel Upper Head Penetrations to Support Continued Operation: D. C. Cook Unit 2," dated October 1994.

6) Letter, M. Reinhart, NRC, to E. E. Fitzpatrick, I&M, "D. C. Cook, Unit 1 and 2, Reactor Vessel Head Penetration, Alternate Weld Repair Method," dated April 9, 1996.

In Bulletin 2001-01, the NRC requested that addressees provide information related to the structural integrity of the reactor pressure vessel head penetration (VHP) nozzles for their respective facilities, including the extent of VHP nozzle leakage and cracking that has been found to date, the inspections and repairs that have been undertaken to satisfy applicable regulatory requirements, and the basis for concluding that their plans for future inspections will ensure compliance with applicable regulatory requirements.

The following provides the response to each specific information request contained in the Bulletin.

<u>Request 1.a</u>

Provide plant-specific susceptibility ranking for your plant(s) (including all data used to determine each ranking) using the PWSCC susceptibility model described in Appendix B to the MRP-44, Part 2, report.

Response to 1.a

Donald C. Cook Nuclear Plant (CNP) has been analyzed for its susceptibility ranking relative to Oconee-3 Nuclear Plant using the time-at-temperature model described in Reference 1. The susceptibility ranking model is further explained in a second MRP Report, Reference 2. Plant-specific input data used in developing the susceptibility ranking were average operating temperature of the reactor vessel head over the life of the plant and number of effective full power years (EFPY) of operation. The plant-specific information used in the evaluation is listed below and was submitted to the NRC via a Nuclear Energy Institute (NEI) letter, Reference 3.

	Average Head Temperature Range Over Plant Life	Accumulated EFPY (Normalized to 600 degrees)
CNP, Unit 1	575-592 F	9.5 EFPY
CNP, Unit 2	596-601 F	13.0 EFPY

The susceptibility evaluation noted in Reference 1 showed that:

1) It will take 30.3 EFPY of additional operation for CNP Unit 1 (from March 2001) and 8.5 EFPY of additional operation for CNP Unit 2 (from March 2001) to reach the same time-at-temperature as Oconee-3 at the time the leaking nozzles were discovered (March 2001). This is based on the assumption that the average head temperatures remain the same for future years.

2) The histogram grouping noted in Reference 1 for both CNP units are, for Unit 1 – Group 7 (30-50 EFPY), and for Unit 2 – Group 3 (6-10 EFPY).

Also, since a crack was discovered and repaired in CNP Unit 2 in the 1994-96 time period, Unit 2 will be in the NRC category of, "Previously experienced either leakage from or cracking in VHP nozzles."

The plant-specific information on head temperature and the EFPY of operation used in the above evaluation were derived from the information in the responses to Generic Letter 97-01, "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations," dated April 1, 1997, Reference 4. Currently, efforts are underway to re-evaluate and/or revalidate this plant-specific information as noted in MRP-48, Section 2.3.

Request 1.b

Provide a description of the VHP nozzles in your plant(s), including the number, type, inside and outside diameter, materials of construction, and the minimum distance between VHP nozzles.

Response to 1.b

The following table provides a description of the vessel head penetrations.

Unit 1

Type	Number	Material	Weld Material	Outside Diameter	Inside Diameter	Minimum Spacing
CRDM / Thermocouple	73/6	SB-167	ENiCrFe-3 (W86182)	4.000 inches	2.75 inches	11.973 inches
Head Vent	1	SB-167	ENiCrFe-3 (W86182)	1.05 inches	0.612 inches	8.466 inches

Unit 2

Type	Number	Material	Weld Material	Outside Diameter	Inside Diameter	Minimum Spacing
CRDM / Thermocouple	73/5	SB-166	ENiCrFe-3 (W86182)	4.000 inches	2.75 inches	11.973 inches
Head Vent	1	SB-167	ENiCrFe-3 (W86182)	1.05 inches	0.612 inches	8.466 inches

Note: The head vents are now used for the reactor vessel level instrumentation. A CRDM penetration (part length) has been converted to provide the reactor head vent as requested by NUREG 0737.

Request 1.c

Provide a description of the RPV head insulation type and configuration.

Response to 1.c

The insulation used for the reactor head is metallic reflective. It is 3 inches thick and made of Type 304 stainless steel. The configuration is as follows. The outer circumference covers the outer row of VHPs. This is connected to vertical panels that extend up 17 inches. This then connects to a set of flat panels that cover the remaining VHPs. There is a 1.5 inch clearance between the underside of the insulation and the top of the head.

Request 1.d

Provide a description of the VHP nozzle and RPV head inspections (type, scope, qualification requirements, and acceptance criteria) that have been performed at your plant(s) in the past 4 years, and the findings. Include a description of any limitations (insulation or other impediments) to accessibility of the bare metal of the RPV head for visual examinations.

Response to 1.d

There were no inspections in the past 4 years. However, inspections were performed in 1994. CNP Unit 2 was in an extended outage from September 1997 to June 2000, and Unit 1 from September 1997 to January 2001. This limited the amount of EFPY of operation from 1994 to present. See response to request 2.1 and 2.b.

The only potential limitations for access to perform the visual examination of the VHPs is the insulation. If necessary, I&M intends to modify the insulation to allow access for remote visual examination and expects that all VHPs will be accessible for remote visual examination.

Request 1.e

Provide a description of the configuration of the missile shield, the CRDM housings and their support/restraint system, and all components, structures, and cabling from the top of the RPV head up to the missile shield. Include the elevations of these items relative to the bottom of the missile shield.

Response to 1.e

The Missile Shield

The missile shield is a set of removable concrete structures that also act as the divider barrier between the upper and lower containment. These blocks are approximately 5 feet above the seismic support structure. They are 4 feet thick and are made of reinforced concrete. Their design function is to divide the upper containment from the lower containment. This divider barrier forces the Loss-of-Coolant-Accident steam through the ice condenser. The blocks are also designed to resist a missile from a rod ejection event.

The Control Rod Drive Mechanisms

The Control Rod Drive Mechanism (CRDM) is the assembly that positions the control rods in accordance with signals from the rod control system. The assembly consists of the pressure tube, which is attached to the VHP nozzle. Inside the pressure tube is the control rod drive shaft. This shaft is latched to the control rods to control reactor power. Surrounding the pressure tube is the electromagnetic coil stack. This provides the motive force to move the control rods. The CRDM assembly is screwed on the VHP and is seal-welded to prevent leakage. The CRDM housings are constructed from Type 304 stainless steel tubing, and Types CF8 and F8 cast stainless steel.

The Seismic Support Structure

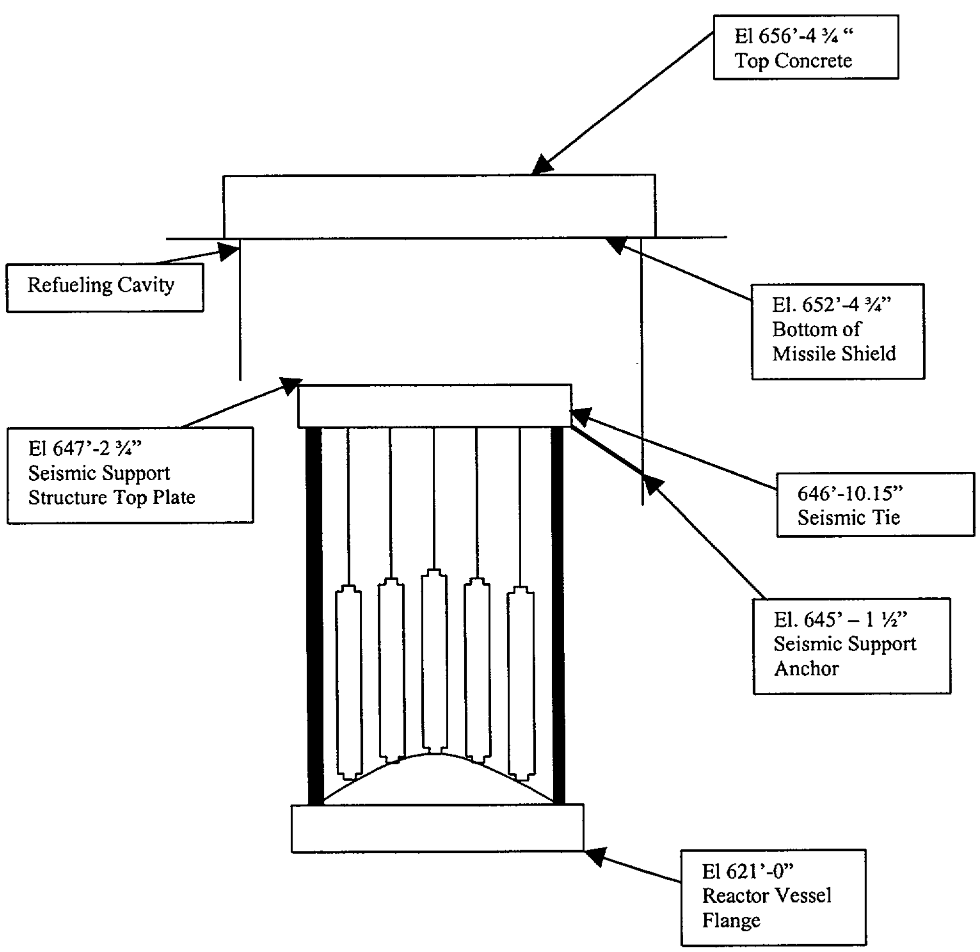
The seismic support structure is located approximately 25 feet above the reactor vessel flange. It provides lateral stability for the CRDM housings as well as access for the interconnecting cables.

The CRDM housings are laterally supported by 3/4-inch plates that are secured to the seismic support structure. The seismic support structure itself is anchored to the refueling cavity wall by a series of struts. The seismic support structure also supports the cable pans that route the power and instrumentation cables to their connector panels. These panels are located on the side of the seismic support structure. The panels host a series of connectors that allow the cables to be disconnected during refueling.

Ventilation and Radiation Shields

Encircling the reactor head and extending approximately 90 inches is the shroud. This shroud provides the ducting for the head ventilation system. The air is fed into the shroud by external ductwork, which is attached to the shroud. Outside the shroud is a radiation shield that extends approximately 8 feet up from the reactor head. This shield reduces the radiation dose of the workers during refueling preparations.

The following sketch provides the relative elevation of the various components.



Request 2.a

If your plant has previously experienced either leakage from or cracking in VHP nozzles, addressees are requested to provide the following information:

a. A description of the extent of VHP nozzle leakage and cracking detected at your plant, including the number, location, size, and nature of each crack detected;

Response to 2.a

No VHP leakage has been detected for CNP Units 1 & 2. However, during the CNP Unit 2 1994 Cycle 10 refueling outage, an eddy current testing (ECT) examination was performed on 71 of the 78 VHPs (the reactor vessel head vent, which is now used for reactor vessel level indication, was not examined). The ECT was qualified in accordance with the EPRI Vendor Qualification Program in place during that time. The results of the ECT examination showed indications in one penetration, number 75. Three indications were found with lengths of 9mm, 16mm, and 45mm. These indications were axial in orientation and were closely spaced. The 3 indications were located on the high side of the penetration near the 160-degree location. The orientation of the penetration can be defined as having a high (180 degree) and low (0 degree) side with respect to the reactor vessel head. In addition, the upper extent of the 45mm indication was near the "J-groove" weld, but the flaw was mostly below the weld.

An Ultrasonic examination (UT) was performed on penetration number 75, covering the 90 to 270 degree locations. The UT examination confirmed the location of an indication between 153 and 160 degrees, as well as the maximum length of 45mm. The UT results showed a maximum depth of 6.8mm for the 45mm indication. The two smaller indications did not show up separately on the UT scan because they were too shallow (< 1mm) or because of their proximity to the larger indication.

A flaw evaluation was completed for the indications in penetration number 75, and the methodology is documented in Reference 5. In this evaluation, the maximum depth of the indication, 6.8mm (as measured by UT) was used, with a length of 45mm. A flaw evaluation chart was developed in Reference 5 for indications in this location. Since the indication has been characterized as 6.8mm deep, and the penetration wall thickness is 16mm, the indication is 42.5 percent of the wall at its deepest point. The flaw evaluation results showed that the NEI acceptance criteria (75 percent through-wall) would not be violated for the next 18-month fuel cycle. A justification for continued operation was provided to the NRC and approved in a safety evaluation report. The flaw was subsequently repaired in 1996, see response to 2.b.

Request 2.b

A description of the additional or supplemental inspections (type, scope, qualification requirements, and acceptance criteria), repairs, and other corrective actions you have taken in response to identified cracking to satisfy applicable regulatory requirements;

Response to 2.b

During the CNP Unit 2 1996 Cycle 11 refueling outage, the 5 outer VHPs, including penetration number 75, were re-inspected using the same ECT examination technique used in 1994. Re-inspection of penetration number 75 identified no significant flaw growth, and no additional indications were identified in the other four outer penetrations. In addition, penetration number 75 was repaired by embedding the flaw using an alternate repair method approved by the NRC in Reference 6. This technique partially removes the flaw and a weld overlay is applied.

Request 2.c

Your plans for future inspections (type, scope, qualification requirements, and acceptance criteria) and the schedule.

Response to 2.c

CNP intends to perform a remote visual examination of all accessible VHPs under the reactor vessel head insulation during the next Unit 2 refueling outage. VT-2 certified personnel will be used, and the personnel will be briefed on the recent events and experiences at Oconee and Arkansas Nuclear One, including the need to specifically look for small quantities of boric acid around VHPs.

In addition to the visual examination, CNP plans to perform an ECT examination of the CRDM and Thermocouple Housing VHP base material near the susceptible weld area (outside diameter under the head and inside diameter) and "J-groove" welds. Relevant indications detected will be investigated using UT to size and characterize depth, length, and orientation (axial or circumferential). ECT and UT procedures will be demonstrated using calibration blocks. Due to access limitations at the thermocouple penetrations, complete eddy current examination of the housing base material and the "J-groove" weld may not be possible. Therefore, an alternate ultrasonic technique may be used for detection in this location, as well as sizing and characterization of any observed indications. Further, CNP will re-examine the embedded flaw in Unit 2's penetration number 75 using a liquid penetrant technique to verify that there is no surface indication open to the primary water environment.

All detected flaws will be evaluated for acceptability using the criteria contained in the vendor's flaw data handbook currently under development. The handbook will contain predetermined evaluations for flaws dependent on size, location, and orientation that will permit determination of the way the flaw may be dispositioned (e.g., no action, justify continued operation, or repair before return to service). This will be a revision to information contained in WCAP-14118. The flaw acceptance criteria contained in WCAP-14118 has been developed through an industry initiative coordinated by NUMARC (later becoming the NEI criteria). Such criteria are normally found in Section XI of the American Society of Mechanical Engineers (ASME) Code; however, Section XI delineates no requirements for examination of the VHP base metal. Therefore, ASME Code acceptance criteria are not available. If any flaw exceeds the acceptance criteria of the vendor's handbook, it will be evaluated for continued operation or repaired prior to the unit's return to service.

The scope of enhanced examinations beyond the next Unit 2 refueling outage has not been determined. This will be determined based, in part, on the results of the examinations discussed above and other industry experience and advancements.

Request 2.d

Your basis for concluding that the inspections identified in 2.c will assure that regulatory requirements are met (see Applicable Regulatory Requirements section). Include the following specific information in this discussion:

(1) If your future inspection plans do not include performing inspections before December 31, 2001, provide your basis for concluding that the regulatory requirements discussed in the Applicable Regulatory Requirements section will continue to be met until the inspections are performed.

(2) If your future inspection plans do not include volumetric examination of all VHP nozzles, provide your basis for concluding that the regulatory requirements discussed in the Applicable Regulatory Requirements section will be satisfied.

Response to 2.d

The CNP Reactor Coolant System (RCS) pressure boundary was designed, fabricated, and constructed so as to have an exceedingly low probability of gross rupture or significant uncontrolled leakage throughout its design lifetime. The RCS pressure boundary was designed and is operated to reduce to an acceptable level the probability of a rapidly propagating crack type failure.

No VHP leakage has been detected for CNP Units 1 & 2. However, during the CNP Unit 2 1994 Cycle 10 refueling outage, an eddy current testing (ECT) examination was performed on 71 of the 78 VHPs. The results of the ECT examination showed indications in one penetration, number 75 (see response to request 2.a).

During the CNP Unit 2 1996 Cycle 11 refueling outage, the 5 outer VHPs, including penetration number 75, were re-inspected using the same ECT examination technique used in 1994. Re-inspection of penetration number 75 identified no significant flaw growth, and no additional indications were identified in the other four outer penetrations. In addition, penetration number 75 was repaired by embedding the flaw using an alternate repair method approved by the NRC in Reference 6. This technique partially removes the flaw and a weld overlay is applied (see response to request 2.b).

I&M believes that operation until the next Unit 2 RFO does not significantly increase the probability of CNP Unit 2 experiencing a rapidly propagating crack type failure. This is based on: 1) the severity of the indication identified in 1994; 2) the fact that no significant propagation occurred during the period between the 1994 and 1996 inspections; 3) the repairs made in 1996, and; 4) the fact that Unit 2 was in an extended outage from September 1997 to June 2000, which limited the amount of EFPY of operation from 1994 to present.

CNP is required to comply with the provisions of 10 CFR 50.55a and to have a quality assurance program that meets the requirements of 10 CFR 50, Appendix B, and Technical Specification 3.4.6.2.1, which does not allow any pressure boundary leakage.

With respect to VHP cracking, Section XI of the ASME Code delineates no requirements for examination of the Alloy 600 VHPs. The only inspection required is of the bi-metallic weld joining the Alloy 600 penetration to the stainless steel head adapter. The VHP inspection will be performed using approved procedures and qualified personnel, in conjunction with the existing corrective action program. No through-wall cracks or cracks that could rapidly propagate are allowed.

The examination to be performed for CNP Unit 2 will be ECT. PWSCC originated cracks developed on the OD, above the "J-grove" weld, must be initiated from through wall cracks propagating from surfaces in contact with primary coolant. Therefore, ECT of the wetted inside and outside surfaces of the nozzle and the "J-groove" weld will detect any cracks open to the surface. Any through-wall cracks will be repaired, and other indications will be dispositioned using predetermined acceptance criteria and the plant's corrective action process. The unit will not be returned to service with reactor coolant boundary leakage. Therefore, the technical specifications and the regulations will continue to be met.

Request 4.a

4. If the susceptibility ranking for your plant is greater than 5 EFPY and less than 30 EFPY of ONS2, addressees are requested to provide the following information:

   a. your plans for future inspections (type, scope, qualification requirements, and acceptance criteria) and the schedule.

Response to 4.a

For Unit 2, see response to information request 2.c.

Because the Unit 1 susceptibility ranking places it greater than 30 EFPY, no response to this request is required. In 1994, the outer portion of the penetrations was examined by remote visual cameras. Although some boric acid deposits were found, these were attributed to past leakage from canopy seal welds. No leakage was attributed to penetrations. However, during the spring 2002 refueling outage, CNP intends to conduct a remote visual examination. This will be done because the remote visual inspection performed in 1994 did not reach all of the penetrations, and because the evaluation of boric acid deposits was not to today's standards. If any leakage is detected, ECT/UT examination will be performed on the suspected penetrations(s). All flaws will be evaluated, and any that do not meet the acceptance criteria contained in the vendor's flaw data handbook (currently under development) will be evaluated for continued operation or repaired in accordance with Section XI of the ASME Code or using an alternate repair method approved by the NRC.

Also for Unit 1, the capability to perform ECT/UT examinations similar to that for Unit 2 will be available. The decision to perform ECT/UT examinations will be dependent on the results obtained for Unit 2, other industry results attained prior to the Unit 1 outage, and the visual examinations results obtained for Unit 1.

Request 4.b

Provide your basis for concluding that the inspections identified in 4.a. will assure that regulatory requirements are met (see Applicable Regulatory Requirements section). Include the following specific information in this discussion:

(1) If your future inspection plans do not include a qualified visual examination at the next scheduled refueling outage, provide your basis for concluding that the regulatory requirements discussed in the Applicable Regulatory Requirements section will continue to be met until the inspections are performed

(2) The corrective actions that will be taken, including alternative inspection methods (for example, volumetric examination), if leakage is detected.

Response to 4.b

Examinations are scheduled for the next Unit 2 refueling outage, and will include visual, ECT, and UT examinations (as discussed in the response to request 2.c and 2.d).

CNP intends to conduct a visual examination of Unit 1 during the next refueling outage. If any leakage is detected, ECT/UT examination will be performed on the suspected penetrations(s) (as discussed in 2.c and 4.a).

Certified personnel will perform these examinations, and the examinations will be performed using procedures that have been developed in accordance with the CNP or vendor's quality assurance program. All flaws will be evaluated, and any that do not meet the acceptance criteria contained in the vendor's flaw data handbook (previously discussed under response to request 2.c) will be evaluated for continued operation or repaired in accordance with Section XI of the ASME Code or using an alternate repair method approved by the NRC.

These provisions provide reasonable assurance that the VHP reactor coolant pressure boundary is not breached, and will assure that the applicable regulatory requirements are met.

ATTACHMENT 2 TO C1001-08

COMMITMENTS

The following table identifies those actions committed to by Indiana Michigan Power Company (I&M) in this document. Any other actions discussed in this submittal represent intended or planned actions by I&M. They are described to the Nuclear Regulatory Commission (NRC) for the NRC's information and are not regulatory commitments.

Commitment	Due Date
CNP will perform a remote visual examination of all accessible VHPs under the reactor vessel head insulation during the next refueling outage. VT-2 certified personnel will be used and the personnel will be briefed on the recent events and experiences at Oconee and Arkansas Nuclear One, including the need to specifically look for small quantities of boric acid around VHPs.	Unit 2 – Next RFO
CNP will perform eddy current testing (ECT) examination of the VHP base material near the susceptible weld area and "J-groove" welds. Relevant indications detected will be investigated using an ultrasonic technique to size and characterize their depth, length, and orientation.	Unit 2 – Next RFO
ECT procedures will be demonstrated using calibration blocks. A demonstrated ultrasonic technique will be used for detection, as well as sizing and characterization, of any observed indications.	Unit 2 – Next RFO
Certified personnel will perform the examinations using procedures that have been developed in accordance with CNP or vendor's quality assurance program.	Unit 2 – Next RFO
CNP will re-examine an embedded flaw in Unit 2's penetration number 75 (repair performed in 1996) using a liquid penetrant technique to verify that there are no surface indications open to the primary water environment.	Unit 2 – Next RFO
All detected flaws will be evaluated for acceptability using the criteria contained in the vendor's flaw data handbook currently under development. This will be a revision to information contained in WCAP-14118.	Unit 2 – Next RFO
If any flaw exceeds the acceptance criteria of the vendor's handbook, it will be dispositioned for continued operation or repaired prior to returning the unit to service.	Unit 2 – Next RFO