April 1, 2002

AEP:NRC:2054-01
10 CFR 50.54(f)

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
RESPONSE TO
NUCLEAR REGULATORY COMMISSION BULLETIN 2002-01
REACTOR PRESSURE VESSEL HEAD DEGRADATION AND
REACTOR COOLANT PRESSURE BOUNDARY INTEGRITY

Reference: Nuclear Regulatory Commission Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," dated March 18, 2002

In the referenced bulletin, the Nuclear Regulatory Commission requested pressurized-water reactor licensees to provide information related to the integrity of the reactor coolant pressure boundary, including the reactor pressure vessel head, and the extent to which inspections have been undertaken to satisfy applicable regulatory requirements. The bulletin also requested that licensees provide the basis for concluding that their plants satisfy applicable regulatory requirements related to the structural integrity of the reactor coolant pressure boundary, and that future inspections will ensure continued compliance with applicable regulatory requirements. The bulletin requires that responses be provided within 15 days of the bulletin, 30 days following the completion of vessel head examinations, and 60 days of the date of the bulletin. This letter responds to the 15-day request.

Attachment 1 to this letter provides the information that was requested within 15 days of the date of the bulletin. Attachment 2 contains a list of commitments made in this letter.

Should you have any questions, please contact Mr. Gordon P. Arent, Manager of Regulatory Affairs, at (616) 697-5553.

Sincerely,

J. E. Pollock
Site Vice President

Attachments

/dmb

c: K. D. Curry, w/o attachments
J. E. Dyer
MDEQ - DW & RPD, w/o attachments
NRC Resident Inspector
R. Whale, w/o attachment

AFFIRMATION

I, Joseph E. Pollock, 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.

American Electric Power Service Corporation

J. E. Pollock
Site Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS ______ DAY OF _______________, 2002

____
Notary Public

My Commission Expires _____________________

bc: G. P. Arent, w/o attachments
G. F. Borlodan/C. R. Lane/K. R. Worthington
P. B. Cowan, w/o attachments
E. D. Forbis
D. J. Garner
R. W. Gaston, w/o attachments
S. A. Greenlee
S. B. Haggerty
D. W. Jenkins, w/o attachments
T. P. Noonan/M. R. Hill
J. E. Pollock, w/o attachments
M. W. Rencheck, w/o attachments
J. F. Stang, Jr. - NRC Washington, DC
T. R. Stephens

ATTACHMENT 1 TO AEP:NRC:2054-01

RESPONSE TO
NUCLEAR REGULATORY COMMISSION BULLETIN 2002-01
REACTOR PRESSURE VESSEL HEAD DEGRADATION AND
REACTOR COOLANT PRESSURE BOUNDARY INTEGRITY

In Bulletin 2002-01, the Nuclear Regulatory Commission (NRC) requested that pressurized-water reactor licensees provide information related to the integrity of the reactor coolant pressure boundary including the reactor pressure vessel head and the extent to which inspections have been undertaken to satisfy applicable regulatory requirements. The bulletin also requested that licensees provide the basis for concluding that their plants satisfy applicable regulatory requirements related to the structural integrity of the reactor coolant pressure boundary, and that future inspections will ensure continued compliance with applicable regulatory requirements. The bulletin requires that responses be provided within 15 days of the bulletin, 30 days following the completion of vessel head examinations, and 60 days of the date of the bulletin. This letter responds to the 15-day request.

The following provides the response to each specific information request that was required to be submitted within 15 days of the date of the bulletin.

Request 1.A

Provide a summary of the reactor pressure vessel head inspection and maintenance programs that have been implemented at your plant.

Response to 1.A

Generic Letter 88-05 Program

The requirements of Generic Letter (GL) 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Boundary Components in PWR Plants," have been incorporated into plant procedure 12-PMP-5030-001-001, "Boric Acid Corrosion of Ferritic Steel Components and Materials." Visual inspections are performed in accordance with the procedure during unit restart following each outage in which the reactor coolant system (RCS) is depressurized. The procedure establishes the guidelines for the identification, examination, and evaluation of boric acid-induced corrosion of ferritic steel components within the RCS pressure boundary.

Plant personnel who identify boric acid leakage that may affect ferritic steel components are required to write an action request that identifies the component, the type of leak (wet/dry), physical location, the portion of the component in contact with the boric acid, and other components that may be in contact with the boric acid. The boric acid program owner is required to examine the component for evidence of boric acid corrosion. If boric acid corrosion is confirmed, or is indeterminate, a VT-1 examination by a certified individual is required to determine the extent of the boric acid corrosion/wastage. A qualified engineer must then evaluate the current wastage and the projected amount of wastage and must make recommendations for corrective action.

In-service Inspection Program

The in-service inspection (ISI) visual inspection requirements to identify RCS leakage are set forth in procedure 12-QHP-5070-NDE-001, "Visual VT-2 Examination: RCS System Leakage Test." This procedure requires demonstration of the integrity of the RCS by a visual VT-2 examination during a system leakage test at normal system operating pressure and temperature conditions in Mode 3. The procedure satisfies the testing and documentation requirements set forth in the American Society of Mechanical Engineers Code, Section XI, Articles IWA-5000, IWB-2500, IWB-5000, and Code Cases N-498-1, N-416-1, and N-533. This system leakage test is performed prior to start-up following each refueling outage, and following the opening and closing of a component in the system.

During the VT-2 examination, the examiner is required to record leakage, evidence of leakage, or structural distress, the affected component, the leakage location, and the amount of leakage. All evidence of leakage or evidence of structural distress must be documented and forwarded to the in-service inspection program coordinator to evaluate for corrective action.

Augmented Inspections

Unit 1

In 1994, a visual inspection was performed on a portion of the Unit 1 reactor vessel head. Following the visual inspection and cleaning, a VT-1 examination was performed to assess the condition of this portion of the head. Additional details concerning this inspection are contained in the response to 1.C.

An inspection similar to the one performed on the Unit 2 reactor vessel head during the cycle 13 refueling outage will be performed on the Unit 1 reactor vessel head. This is described in the response to 1.D.

Unit 2

In response to NRC Bulletin 2001-01, "Circumferential Cracking Of Reactor Pressure Vessel Head Penetration Nozzles," examinations were performed on the Unit 2 reactor vessel head during the Unit 2 cycle 13 refueling outage completed in February 2002. For each of the control rod drive mechanism (CRDM) and thermocouple penetrations, one of the following was performed:

- A qualified visual examination.
- Surface examination, eddy current test (ECT) or liquid penetrant, of the wetted surfaces on and near the "J-groove" weld on the outside and inside diameter.
- Ultrasonic testing from the inside diameter of the penetration capable of detecting circumferential cracks on the outside diameter above and in the vicinity of the "J-groove" weld.

A 100% bare metal inspection of the upper surface of the head was also performed in conjunction with the CRDM and thermocouple penetration examinations.

Request 1.B

Provide an evaluation of the ability of your inspection and maintenance programs to identify degradation of the reactor pressure vessel head, including thinning, pitting, or other forms of degradation such as the degradation of the reactor pressure vessel head observed at Davis-Besse.

Response to 1.B

The inspection programs listed in the response to 1.A provide reasonable assurance that any significant leakage that would result in serious degradation of the reactor pressure vessel head would be detected prior to the occurrence of such damage.

These programs are effective in identifying boric acid that might leak from above the head to the insulation. Access ports permit viewing the top of the insulation, and any accumulation of boric acid would be visible. If any boric acid were found to have seeped onto the insulation, the panels would be removed to determine if any leaked onto the carbon steel head. Cleaning, examination, evaluation, and any required corrective action would then be taken in accordance with the GL 88-05 program. If a leak similar to that at Davis-Besse occurred in either unit, it is I&M's judgement that the flow from the leak would be sufficient to allow the corrosion products and boric acid to reach the outside of the insulation where they would be observed.

The effectiveness of identifying leakage that occurs above the reactor pressure vessel head insulation has been demonstrated by the discovery of three leaking Unit 1 canopy seals and their repair in 1994. Also, during the Unit 2 cycle 13 restart, a leaking thermocouple fitting and Cetna seal were discovered during the visual inspections, and were repaired.

The non-destructive examinations performed on Unit 2, and the non-destructive examinations to be performed on Unit 1 (see the response to 1.D), will provide a high level of confidence that through-wall cracks on the CRDM and thermocouple penetrations do not exist.

Request 1.C

Provide a description of any conditions identified (chemical deposits, head degradation) through the inspection and maintenance programs described in 1.A that could have led to degradation and the corrective actions taken to address such conditions.

Response to 1.C

Unit 1

A 100 percent bare metal inspection of the reactor vessel head has not been performed. However, in 1994, using a remote camera, a bare metal inspection of the 34 outermost penetrations was performed to detect evidence of RCS leakage. No leakage from the vessel head was detected during this examination. During the same outage, three canopy seal weld leaks that had been identified were repaired. Three panels of insulation were moved or removed for boric acid cleaning. The majority of the boric acid was cleaned, but minor amounts of dried boric acid crystals were allowed to remain on the head. Upon completion of cleaning, a bare metal VT-1 examination of these areas was performed using both direct visual and a remote camera system. The VT-1 examination was found to be acceptable. The three leaking canopy seal welds were repaired prior to returning Unit 1 to power.

Unit 2

There have been no identified leaks of boric acid or any other corrosive material that reached the reactor pressure vessel head or insulation since the 100 percent bare metal effective visual inspection that was conducted in January 2002. During the 2002 inspection, no indications of boron deposits (indicative of penetration leakage) were found on the penetrations. Additionally, in January and February 2002, a non-destructive examination of 100 percent of the CRDM and thermocouple penetrations was conducted in response to NRC Bulletin 2001-01. The methodologies employed during the inspection would not have been able to identify a potential cavity behind the nozzles.

However, the inspection revealed no through-wall leaks (a prerequisite to void formation) that could have challenged or are challenging the integrity of the reactor vessel head.

The results of the Unit 2 inspection were transmitted to the NRC in a letter from M. W. Rencheck to NRC Document Control Desk, dated March 28, 2002, (AEP:NRC:2054).

Request 1.D

Provide your schedule, plans, and basis for future inspections of the reactor pressure vessel head and penetration nozzles. This should include the inspection method(s), scope, frequency, qualification requirements, and acceptance criteria.

Response to 1.D

Indiana Michigan Power Company (I&M) will continue performing the GL 88-05 inspections and the ISI examinations for both units in accordance with the procedures described in the response to 1.A. The basis is that these inspections have been effective at detecting boric acid leaks from above the reactor head insulation.

I&M will perform a 100% bare metal visual inspection of the Unit 1 head under the insulation during the upcoming Unit 1 cycle 18 refueling outage. Certified personnel will perform the inspections.

I&M will perform one of the following inspections for each of the CRDM and thermocouple penetrations during the Unit 1 cycle 18 refueling outage:

- Surface examination, ECT or liquid penetrant, of the wetted surfaces on and near the "J-groove" weld on the outside and inside diameter, supplemented by ultrasonic testing as necessary for weld locations that are not accessible by eddy current probes.

- Ultrasonic testing from the inside diameter of the penetration capable of detecting circumferential cracks on the outside diameter above and in the vicinity of the J-groove weld.

The basis for the 100% bare metal visual inspection and the non-destructive examinations is to confirm that the vessel head has no indications of degradation, and that the CRDM and thermocouple penetrations have no unacceptable flaws.

Request 1.E

Provide your conclusion regarding whether there is reasonable assurance that regulatory requirements are currently being met. This discussion should also explain your basis for concluding that the inspections discussed in the response to 1.D will provide reasonable assurance that these regulatory requirements will continue to be met. Include the following specific information in this discussion:

(1) If your evaluation does not support the conclusion that there is reasonable assurance that regulatory requirements are being met, discuss your plans for plant shutdown and inspection.

(2) If your evaluation supports the conclusion that there is reasonable assurance that regulatory requirements are being met, provide your basis for concluding that all regulatory requirements discussed in the Applicable Regulatory Requirements section will continue to be met until the inspections are performed.

Response to 1.E

The Donald C. Cook Nuclear Plant (CNP) RCS pressure boundary was designed, fabricated, and constructed 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.

CNP complies with the provisions of 10 CFR 50.55a and has a quality assurance program that meets the requirements of 10 CFR 50, Appendix B. Additionally, Technical Specification 3.4.6.2.1 does not allow any pressure boundary leakage.

Unit 1 Evaluation

The inspection programs described in the response to 1.A, GL 88-05 and ISI, provide reasonable assurance that regulatory requirements are met. The 1994 inspection discussed in the response to 1.C provided evidence that there were no active leaks or degradation in a portion of the vessel head. In addition, the most recent boric acid walkdowns following a forced outage in September 2001 revealed no leakage from sources above the head.

CNP Unit 1 has been analyzed for its susceptibility ranking relative to the Oconee-3 Nuclear Plant, and penetration cracking is not expected to exist in Unit 1 due to its low susceptibility ranking. The plant-specific input data used in developing the susceptibility ranking were the average operating temperature of the reactor vessel head over the life of the plant and the number of effective full power years of operation. The susceptibility evaluation showed that it will take 30.3 EFPY of additional operation for CNP Unit 1 (from March 2001) to reach the same time-at-temperature as Oconee-3 at the time the leaking nozzles were discovered. This is based on the assumption that the average head temperatures remain the same for future years.

The continuation of the inspections performed as part of the GL 88-05 program and the ISI program provide assurance that the regulatory requirements will continue to be met. The planned Unit 1 visual inspections and non-destructive examinations will validate and verify that the inspection programs are effective.

Unit 2 Evaluation

The recent inspections conducted for NRC Bulletin 2001-01 confirmed that no through-wall cracks exist, and that no boric acid buildup existed on top of the head. The continuation of the inspections performed as part of the GL 88-05 program and the ISI program provide assurance that the regulatory requirements will continue to be met.

ATTACHMENT 2 TO AEP:NRC:2054-01

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
I&M will perform a 100% bare metal visual inspection of the Unit 1 reactor vessel head under the insulation.	Unit 1 cycle 18 refueling outage
I&M will perform one of the following inspections for each of the control rod drive mechanisms and thermocouple penetrations: • Surface examination, eddy current testing or liquid penetrant, of the wetted surfaces on and near the "J-groove" weld on the outside and inside diameter, supplemented by ultrasonic testing as necessary for weld locations that are not accessible by eddy current probes. • Ultrasonic testing from the inside diameter of the penetration capable of detecting circumferential cracks on the outside diameter above and in the vicinity of the J-groove weld.	Unit 1 cycle 18 refueling outage