Indiana Michigan
Power Company
Cook Nuclear Plant
500 Circle Drive
Buchanan, MI 49107
616-465-5901

INDIANA
MICHIGAN
POWER

May 10, 2002

AEP:NRC:2054-02
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
SIXTY DAY RESPONSE TO
NUCLEAR REGULATORY COMMISSION BULLETIN 2002-01,
"REACTOR PRESSURE VESSEL HEAD DEGRADATION AND
REACTOR COOLANT PRESSURE BOUNDARY INTEGRITY"

Reference: 1. Nuclear Regulatory Commission Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," dated March 18, 2002.

2. Letter from J. E. Pollock, Indiana Michigan Power Company, to Nuclear Regulatory Commission Document Control Desk, "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," submittal AEP:NRC:2054-01, dated April 1, 2002.

In Reference 1, the Nuclear Regulatory Commission requested 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 and 60

A095

days of the date of the bulletin and 30 days after restart following the completion of vessel head examinations. Indiana Michigan Power Company provided information regarding the reactor pressure vessel head in Reference 2. This letter responds to the 60-day request to provide the basis for concluding that Donald C. Cook Nuclear Plant's (CNP) inspection program provides reasonable assurance of compliance with the regulatory requirements applicable to CNP regarding the remainder of the reactor coolant pressure boundary's structural integrity.

Attachment 1 to this letter provides the information that was requested within 60 days of the date of the bulletin.

There are no new commitments 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

/jen

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 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

J E Pollock

J. E. Pollock
Site Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 10 DAY OF MAY, 2002

Jennifer A Kernosky
Notary Public

My Commission Expires 5/24/05

ATTACHMENT TO AEP:NRC:2054-02

SIXTY-DAY 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 and 60 days of the date of the bulletin and 30 days after restart following the completion of vessel head examinations. Indiana Michigan Power Company provided the information regarding the reactor vessel head in a letter from J. E. Pollock to NRC Document Control Desk, dated April 1, 2002. This letter responds to the 60-day request to provide the basis for concluding that Donald C. Cook Nuclear Plant's (CNP) inspection plan provides reasonable assurance of compliance with the regulatory requirements applicable to CNP regarding the remainder of the reactor coolant pressure boundary's structural integrity.

The following summarizes the programs at CNP to detect reactor coolant pressure boundary leakage, and the actions taken when leakage is discovered.

Generic Letter 88-05 Program

The requirements of Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Boundary Components in PWR Plants," have been incorporated into plant procedure 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. Inspections are also performed during initial containment entry following a unit shut down. The procedure establishes the guidelines for the identification, examination, and evaluation of boric acid-induced corrosion of ferritic steel components within the reactor coolant pressure boundary.

Plant personnel who identify boric acid leakage or accumulation 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 ferritic steel is potentially impacted. If boric acid corrosion is confirmed, or is indeterminate, a visual examination (VT-1) 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, if present, and the projected amount of wastage to make recommendations for corrective action. Insulation providing an obstruction to performing this evaluation is removed first.

Operations Walkdowns

During unit startups, teams perform walkdowns of the containment and the systems contained within. One of the tasks performed during these walkdowns is inspecting for leaks and boric acid accumulation. The walkdowns are performed at approximately 300 psig (for the Mode 5 to Mode 4 change), at approximately 1000 psig, and again at full pressure and temperature. To enhance the walkdown's effectiveness, personnel performing the walkdowns are provided information identifying issues documented from the most recent walkdown. These previously identified issues are then re-inspected to verify that they have been satisfactorily resolved.

Walkdowns are also performed during the initial containment entry following a unit shutdown while RCS temperature and pressure are near normal operating values. Operations Department personnel perform the walkdown, and the findings are reviewed by a qualified engineer and an Operations Department representative for possible problems. The corrective action is dependent on the issues identified. If any issue is identified that could possibly impact carbon steel components, additional attention is placed on that issue. If the issue is not a boric acid corrosion concern, then the magnitude of the leakage is considered for repair before restart or repair at the next available window in accordance with the established work control process.

Inservice Inspection Program

The inservice inspection (ISI) requirements to identify RCS leakage are provided 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 (ASME), 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 startup following each refueling outage, and/or following the opening and re-closing of a component within the RCS following a repair or replacement.

During the VT-2 examination, the examiner is required to record evidence of leakage or structural distress, the affected component, the leakage location, and the amount of leakage. This information is forwarded to the ISI program coordinator for corrective action evaluation.

Code Case N-533, "Alternative Requirements for VT-2 Visual Examination of Class 1 Insulated Pressure-Retaining Bolted Connections, Section XI, Division 1," has been approved for use on all Class 1, 2, and 3 systems at CNP. Code Case N-533 is an alternative to ASME Section XI, Article IWA-5242(a) and allows the performance of a system pressure test and VT-2 examination at operating pressure and temperature without having to remove insulation during startup after each refueling outage. The use of Code Case N-533 was approved with the stipulation that a cold depressurized walkdown of each bolted connection with the insulation removed would be performed during each refueling outage, and, following a 4-hour hold time, a VT-2 examination would be performed at normal operating conditions with the insulation in place. Evidence of any leakage at a bolted connection would be evaluated in accordance with ASME Section XI, Article IWA-5250. This code provision stipulated that if leakage occurred at a bolted connection during a system pressure test, the bolting would be removed, a VT-3 visual examination performed for evidence of corrosion, and an evaluation performed in accordance with ASME Section XI, Article IWA-3100. Relief from this provision of ASME Section XI, Article IWA-5250, was also approved. The relief allows an evaluation of the bolted connection to determine the susceptibility of the bolting material to corrosion and the potential for failure. This action was in lieu of removing the bolting for examination. The above code reliefs are implemented by 12-QHP-5070-NDE-001, "Visual VT-2 Examination: RCS System Leakage Test," and 12-QHP-5070-NDE-002, "Visual VT-2 Examinations: Inservice and Repair/Replacements."

Per 12-QHP-5070-NDE-001 and in accordance with ASME Section XI, Articles IWB-5000 and IWB-2500 and Code Case N-533, each refueling outage the insulation is removed from all Class 1 pressure retaining system bolted connections and a cold depressurized walkdown conducted. During startup, a walkdown of the Class 1 bolted connections with the insulation in place at operating pressure and temperature is conducted. During these cold and hot walkdowns, each pressure boundary connection is examined for evidence of leakage and the results are documented. If leakage is suspected or discovered at a bolted connection, prior to removing bolts for a VT-3 examination, an engineering evaluation is performed that considers the following:

- History of leakage
- Location of leakage
- Bolting material
- Evidence of corrosion with the connection assembled
- Corrosiveness of the internal fluid
- Available information for similar bolting materials in similar environments
- Condition of other components in the vicinity that could also be degraded due to leakage

If the evaluation concludes that there has been degradation of bolting or is inconclusive in determining bolting degradation, then the required action is to remove the bolt closest to the source of leakage and perform a VT-3 examination. If the removed bolt has evidence of degradation, then all remaining bolting in the connection is removed and VT-3 examined.

A VT-2 examination may be performed on other components without removing the insulation. If evidence of leakage is observed, the insulation is removed to identify the source of the leak.

Conclusion

The CNP boric acid inspection and ISI programs provide the capability to detect boric acid leakage from the RCS. When leakage is detected, actions are taken to assess the impact of the leakage and to define any corrective actions required to maintain reactor coolant pressure boundary integrity. This provides reasonable assurance of compliance with the applicable CNP regulatory requirements regarding the reactor coolant pressure boundary integrity.