January 17, 2003

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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
REQUEST FOR ADDITIONAL INFORMATION REGARDING NUCLEAR
REGULATORY COMMISSION BULLETIN 2002-01, "REACTOR
PRESSURE VESSEL HEAD DEGRADATION AND REACTOR COOLANT
PRESSURE BOUNDARY INTEGRITY"
(TAC NOS. MB4540 AND MB4541)

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, Sixty Day Response To Nuclear Regulatory Commission Bulletin 2002-01 Reactor Pressure Vessel Head Degradation And Reactor Coolant Pressure Boundary Integrity," submittal AEP:NRC:2054-02, dated May 17, 2002

3. Letter from John F. Stang, Nuclear Regulatory Commission, to A. C. Bakken III, Indiana Michigan Power Company, "Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," 60-Day Response for Donald C. Cook Nuclear Plant Units 1 and 2 – Request for Additional Information (TAC Nos. MB4540 and MB4541)," dated November 19, 2002

In Reference 1, the Nuclear Regulatory Commission (NRC) 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. Indiana Michigan Power Company, the licensee for Donald C. Cook Nuclear Plant Units 1 and 2, provided the requested information for both units in Reference 2.

In Reference 3, the NRC requested additional information regarding the inspections and evaluations for reactor coolant system components exposed to boric acid. The attachment to this letter responds to that request.

This letter contains no new commitments. Should you have any questions, please contact Mr. Brian A. McIntyre, Manager of Regulatory Affairs, at (269) 697-5806.

Sincerely,

J. E. Pollock
Site Vice President

RV/rdw

Attachment

cc: K. D. Curry, w/o attachment
J. E. Dyer
J. T. King, MPSC, w/o attachment
MDEQ – DW & RPD, w/o attachment
NRC Resident Inspector
J. F. Stang, Jr. – NRC Washington, DC

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
Site Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS ______ DAY OF _______________, 2003

____
Notary Public

My Commission Expires ____________________ bc:

D. C. Baker
A. C. Bakken III, w/o attachment
G. F. Borlodan/C. R. Lane/R. E. Hall
M. J. Finissi, w/o attachment
D. J. Garner
S. A. Greenlee
D. W. Jenkins, w/o attachment
J. A. Kobyra, w/o attachment
B.A. McIntyre, w/o attachment
J. E. Newmiller
T. P. Noonan/M. R. Hill
J. E. Pollock, w/o attachment
D.J. Poupard
S. E. Saad/K. A. Muller
T. R. Satyan-Sharma
M. K. Scarpello
T. K. Woods, w/o attachment

ATTACHMENT TO AEP:NRC:3054

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

Nuclear Regulatory Commission (NRC) Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," 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. Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant (CNP) Units 1 and 2, provided the requested information for both units in a letter dated May 17, 2002.

In a letter dated November 19, 2002, the NRC requested additional information regarding the inspections and evaluations for reactor coolant system components exposed to boric acid. The requested information is provided below.

NRC QUESTION 1

Provide detailed information on, and the technical basis for, the inspection techniques, scope, extent of coverage, and frequency of inspections, personnel qualifications, and degree of insulation removal for examination of Alloy 600 pressure boundary material and dissimilar metal Alloy 82/182 welds and connections in the reactor coolant pressure boundary (RCPB). Include specific discussion of inspection of locations where reactor coolant leaks have the potential to come in contact with and degrade the subject material (e.g. reactor pressure vessel (RPV) bottom head.)

RESPONSE:

Visual inspection of RCPB components and other boric acid corrosion-susceptible components is performed in accordance with CNP's boric acid inspection procedure. The scope and extent of coverage of boric acid inspection includes the entire RCPB, which encompasses Alloy 600 and Alloy 82/182 dissimilar weld pressure retaining materials. For inspections conducted inside containment, the boric acid inspection procedure lists susceptible ferritic steel components and potential leak sources that could impact ferritic steel components.

The inspection frequencies for systems and components in contact with boric acid that are not accessed during power operations are a) the initial containment inspection upon any unit shutdown, b) when RCPB pressure is returned to approximately 300 pounds per square inch gauge (psig) upon unit restart, c) when RCPB pressure is returned to approximately 1000 psig upon unit restart, and d) full RCPB pressure prior to unit restart. The item (a) inspection may not be performed if it was performed within the previous 60 days. Item (c) may be waived by senior management at any time. These inspections are performed by teams led by Operations personnel. When evidence of wastage is detected or is indeterminate, the material condition of the affected ferritic steel components is examined by a qualified VT-1 inspector. Engineering personnel who perform boric acid wastage evaluations, assessing the material condition of the degraded component or system, are qualified to an I&M INPO accredited Engineering Support Personnel (ESP) Training Program Position Specific Guide.

Readily accessible sections of insulation installed on the RPV upper head will be removed for boric acid inspections performed under cold shutdown condidtions. Typically, no other insulation removal is performed prior to performing boric acid inspections, as sufficient time will have elapsed for boric acid leakage to be observable through insulation joints and seams. Although insulation is not removed from the RPV bottom head during visual inspection, any leakage would be readily evident through the joints in the segmented insulation panels, and any leakage from bottom–mounted penetrations could not pool nor flow upward under the insulation. Hence, there is no potential to disperse over the inverted hemispherical bottom head.

Additionally, every refueling outage, RCPB system leakage testing is conducted by VT-2 certified personnel in accordance with Section XI of the American Society of Mechanical Engineers (ASME) Code. Also, a 100 percent bare metal visual examination of the RPV upper head will be performed using robotic inspection equipment/manual guide probes.

The inspection of other systems and components in contact with boric acid that are accessed during power operations is conducted by plant personnel (Operations, Engineering, Radiation Protection, Maintenance) during normal rounds and work activities. None of the RCPB is typically accessed during power operation. Any discoveries of boric acid leakage are integrated into the boric acid corrosion control program.

NRC QUESTION 2

Provide the technical basis for determining whether or not insulation is to be removed to examine <u>all</u> locations where conditions exist that could cause high concentrations of boric acid on pressure boundary surfaces or locations that are susceptible to primary water stress corrosion cracking (Alloy 600 base metal and dissimilar metal Alloy 82/182 welds). Identify the type of insulation for each component examined, as well as any limitations to removal of insulation. Also include in your response actions involving removal of insulation required by your procedures to identify the source of leakage when relevant conditions (e.g. rust stains, boric acid stains, or boric acid deposits) are found.

RESPONSE:

The insulation installed on the RCPB consists of panels that are mechanically fastened and can be removed. This insulation is typically metallic-reflective type insulation. Scaffolding would be required to reach some locations, and some areas, such as the lower reactor vessel head, would have higher radiation exposure rates. I&M does not consider these to be an impediment to insulation removal.

Accessible sections of insulation installed on the RPV upper head will be unbuckled and removed to provide a better view of the RPV upper head for boric acid inspections performed under cold shutdown conditions.

Except as noted above, no insulation is removed prior to performing boric acid inspections as sufficient time will have elapsed for boric acid leakage to be observable through insulation joints and seams. The technical bases for not removing insulation are as follows:

- Reactor operation at pressure and temperature has been shown by the experience at the V. C. Summer Plant (where a 0.3 gallon per minute leak deposited 100 to 200 pounds of boric acid in the area surrounding the leak) to provide adequate time for even very small leaks to migrate through the insulation and become visible.
- This approach is consistent with the guidance provided by Section XI of the ASME Boiler and Pressure Vessel Code where a four-hour hold time is required prior to performing VT-2 examinations of insulated systems.
- The practical experience which serves as the technical basis for the ASME Code provides reasonable assurance that any leakage resulting from primary water stress corrosion cracking in insulated systems will be readily identifiable following the prescribed hold time allowed by the ASME Code.

During the performance of the depressurized walkdown of the RCPB for ASME Code, Section XI inservice inspection system leakage test, insulation is removed from all bolted connections in accordance with Code Case N-533. The RCPB is then re-examined at the pressure and temperature corresponding to 100 percent rated reactor power prior to returning the unit to service. It should be noted that I&M has requested approval of an alternative to use Code Case N-616 for bolted connection inspections. This request, which was transmitted by submittal AEP:NRC:2055-05, dated September 12, 2002, would eliminate the requirement to remove insulation if the bolting material is resistant to boric acid corrosion.

I&M procedures require that the source of leakage be identified and evaluated. Insulation that interferes with performing the evaluation of active leakage would have to be removed to perform the evaluation.

NRC QUESTION 3

Describe the technical basis for the extent and frequency of walkdowns and the method for evaluating the potential for leakage in <u>inaccessible areas</u>. In addition, describe the degree of inaccessibility, and identify any leakage detection systems that are being used to detect potential leakage from components in inaccessible areas.

RESPONSE:

There are no inaccessible areas inside containment during shutdown. However, many high radiation areas in the containment are not accessible during operation. The RCPB visual examinations are performed during reactor shutdowns. It is I&M's judgement that any RCPB leakage would deposit boric acid at locations where it could be observed using the existing inspection procedures. Leakage from inaccessible areas at power should be identified as part of the RCPB leak monitoring required by the technical specifications.

The methods used to detect RCPB leakage at power are:

- The containment atmosphere particulate radioactivity monitor.
- Water inventory balances.
- The containment sump level and flow monitor.
- The containment humidity monitor.
- The containment atmosphere gaseous radioactivity monitor.

RCPB leakage is monitored as a part of technical specification compliance. This information is recorded and routinely reviewed by management in daily status meetings.

I&M considers the visual inspections that are performed at least once per refueling cycle adequate for identifying leakage paths and taking timely corrective action.

NRC QUESTION 4

Describe the evaluations that would be conducted upon discovery of leakage from mechanical joints (e.g. bolted connections) to demonstrate that continued operation with the observed leakage is acceptable. Also describe the acceptance criteria that were established to make such a determination. Provide the technical basis used to establish the acceptance criteria. In addition,

a. If observed leakage is determined to be acceptable for continued operation, describe what inspection/monitoring actions are taken to trend/evaluate changes in leakage, or
b. If observed leakage is not determined to be acceptable, describe what corrective actions are taken to address the leakage.

RESPONSE:

The discovery of active boric acid leakage from mechanical joints where the leakage affects, or potentially affects, ferritic steel components prompts the initiation of a Boric Acid Inspection Checklist. When evidence of wastage is detected or is indeterminate, the material condition of the affected ferritic steel components is examined by a qualified VT-1 inspector. An engineer, qualified to an I&M INPO accredited Engineering Support Personnel (ESP) Training Program Position Specific Guide, performs an evaluation of the boric acid leakage rate and the material condition of the affected parts and components. Utilizing the guidance provided by the 1989 edition of the ASME Code, Section XI, IWA-5250 and the Electric Power Research Institute (EPRI) Boric Acid Corrosion Guidebook, Report Number 1000975, Revision 1, the assessing engineer determines whether repair or replacement activities will be performed or if continued operation with the observed leakage is acceptable. The assessing engineer evaluation is documented in Part III of the Boric Acid Inspection Checklist, which considers the following:

- Materials affected or potentially affected by the boric acid leakage.
- Environmental conditions that may affect the boric acid residue.
- Boric acid leakage rate and source.
- Affected systems' functions.
- Leakage history of the affected component or part.
- Visual evidence of corrosion/wastage.
- Estimated elapsed time of exposure to boric acid.
- Estimated amount of time of continued operation.
- Potential consequences of failure.
- Material replacement recommendations.
- Long-term action recommendations.
- Continued operation inspection frequency.
- Impact on other systems/components.

The preferred boric acid corrosion management resolution is to effect proper repair/replacement, rectifying the active leakage condition and correcting any material condition concerns. In order to justify continued operation, the critical functions of the affected component are considered, and the guidance provided by Section 8 of the EPRI Boric Acid Corrosion Guidebook is used. The material condition is evaluated to ensure that affected components meet applicable Code design requirements, and continue to meet applicable Code design requirements for the justified period of continued operation. Also, an evaluation shall be performed of the potential impact of the affected components failure on the design and licensing bases should the predicted remaining service life be proven to be non-conservative. The requirements that are considered to justify continued operation are as follows:

- Boric acid leakage source and flow rate shall be completely understood.
- The boric acid leakage path must be identified and all ferritic steel parts in the leakage path shall be assessed.
- The amount of existing degradation shall be characterized either by cleaning, disassembling and examining the affected parts or by estimating degradation based on the volume of boric acid crystals, the material involved and the corrosion environment.
- Once the current extent of leakage and degradation has been characterized, an estimate is made for the extent of degradation expected until the next planned inspection.
- The maximum corrosion rate for the predicted corrosion environment is conservatively established considering any possible mitigating circumstances.

- The predicted degradation, until the next planned inspection, is compared to applicable ASME Code requirements to ensure the minimum acceptable material condition is met or exceeded.
- Once the technical justification has been established showing the material condition of the degraded component will remain within Code allowables until the next planned inspection, additional evidence shall be presented to demonstrate that the safe operation of the plant can be maintained even if the justification for continued operation proves to be non-conservative. Additional evidence includes, as a minimum, establishing the frequency for periodic supplemental inspections until the degraded component can be repaired or replaced. Also, PRA, UFSAR accident analysis, and leak-before-break behavior may be utilized to demonstrate acceptable performance.

Boric acid leakage concerns, identified during periodic system pressure testing, are evaluated in accordance with the requirements of the CNP Inservice Inspection Program. The requirements for Inservice Inspection Program corrective actions are in conformance with the requirements of the 1989 edition of the ASME Code, Section XI, Paragraph IWA-5250 and an approved Code relief that was granted in a letter from G. H. Marcus (NRC) to E. E. Fitzpatrick (I&M), dated January 16, 1997. The Code relief allows evaluation of a leaking bolted connection rather than the Code required bolt removal and examination.

NRC QUESTION 5

Explain the capabilities of your program to detect the low levels of reactor coolant pressure boundary leakage that may result from through-wall cracking in the bottom reactor pressure vessel head incore instrumentation nozzles. Low levels of leakage may call into question reliance on visual detection techniques or installed leakage detection instrumentation, but has the potential for causing boric acid corrosion. The NRC has had a concern with the bottom reactor pressure vessel head incore instrumentation nozzles because of the high consequences associated with loss of integrity of the bottom head nozzles. Describe how your program would evaluate evidence of possible leakage in this instance. In addition, explain how your program addresses leakage that may impact components that are in the leak path.

RESPONSE:

During the boric acid inspection conducted each shutdown, the condition of the area under the RPV lower head is observed. I&M has concluded that the fluid that would flow from any cracks that developed during the previous operating cycle would be discovered during these inspections.

The methods to identify leakage that develops during power operation are provided in the response to NRC Question 3.

As there are no inaccessible areas inside containment during shutdown, I&M considers the visual inspections that are performed at least once per refueling cycle adequate for identifying leakage paths and taking timely corrective action. The impact on components in the leak path would be evaluated using existing plant procedures.

NRC QUESTION 6

Explain the capabilities of your program to detect the low levels of reactor coolant pressure boundary leakage that may result from through-wall cracking in certain components and configurations for other small diameter nozzles. Low levels of leakage may call into question reliance on visual detection techniques or installed leakage detection instrumentation, but has the potential for causing boric acid corrosion. Describe how your program would evaluate evidence of possible leakage in this instance. In addition, explain how your program addresses leakage that may impact components that are in the leak path.

RESPONSE:

During the boric acid inspections conducted during each refueling outage, the condition of areas surrounding small diameter piping is observed. I&M has concluded that evidence of any leaks that developed during the previous cycle of operation would be observed during these inspections.

The methods to identify any leakage that would develop during power operation are provided in the response to NRC Question 3

It is I&M's judgement that any leakage from small diameter nozzles would be detected during the performance of visual inspections, and the impact of the leak path on components in the leak path would be assessed in accordance with existing procedures.

NRC QUESTION 7

Explain how any aspects of your program (e.g. insulation removal, inaccessible areas, low levels of leakage, evaluation of relevant conditions) make use of susceptibility models or consequence models.

RESPONSE:

I&M has a susceptibility evaluation of the reactor vessel head penetrations to primary water stress corrosion cracking. Since I&M is performing a 100 percent visual inspection of the reactor vessel head during outages, this evaluation has not been factored into the boric acid corrosion control program. I&M has not performed any additional susceptibility evaluations for use in the program.

NRC QUESTION 8

Provide a summary of recommendations made by your reactor vendor on visual inspections of nozzles with Alloy 600/82/182 material, actions you have taken or plan to take regarding vendor recommendations, and the basis for any recommendations that are not followed.

RESPONSE:

Based on input from the Westinghouse Owner's Group in a letter dated December 13, 2002, Westinghouse noted a search of the communications sent out to utilities on this subject has not found any letters of recommendation on visual inspections. Additionally, I&M has been unable to identify any CNP specific reactor vendor communications regarding Alloy 600/82/182 visual inspections.

NRC QUESTION 9

Provide the basis for concluding that the inspections and evaluations described in your responses to the above questions comply with your plant Technical Specifications and Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55(a), which incorporates Section XI of the American Society of Mechanical Engineers (ASME) Code by reference. Specifically, address how your boric acid corrosion control program complies with ASME Section XI, Paragraph IWA-5250 (b) on corrective actions. Include a description of the procedures used to implement the corrective actions.

RESPONSE:

The Cook Nuclear Plant Boric Acid Corrosion Control Program complies with corrective action requirements delineated by the 1989 edition of the ASME Code, Section XI, Paragraph IWA-5250 (b). Plant procedures provide the requirements for the identification, examination, evaluation, and correction of active boric acid leakage conditions, as follows:

- I&M has established procedures for the identification, examination, evaluation, and corrective action to preclude recurrence of boric acid induced corrosion of ferritic steel components as part of the boric acid inspection program. These guidelines are applicable when the potential exists to degrade ferritic steel components within the RCPB and other plant systems due to contact with borated water.

- I&M has established actions to be taken to address the potentially corrosive effects of RCPB leakage at less than Technical Specification 3.4.6.2 limits. There are four boric acid corrosion control program considerations in the CNP program as follows:

- Determination of principal locations where leaks less than Technical Specification limits have the potential to cause degradation.
- Procedures for locating small coolant leaks.
- Methods for conducting examinations and prompt engineering evaluations, as required.
- Prompt corrective actions to prevent recurrence of boric acid corrosion such as maintenance activities and design changes (i.e. replacement with non-susceptible materials).

- The boric acid inspection and evaluation procedure specifically identifies ASME Code, Section XI, Paragraph IWA-5250 (b) as the basis for developing the required corrective actions needed to effectively resolve active boric acid leakage.

- CNP procedures specify the requirements for repair or replacement of ASME Class components to resolve active boric acid leakage condition.

- CNP procedures require evaluation, examination and corrective action per the requirements of the 1989 edition of the ASME Code, Section XI, IWA -5250 (a) (2) and a Code relief that was granted in a letter from G. H. Marcus (NRC) to E. E. Fitzpatrick (I&M), dated January 16, 1997. The Code relief allows the evaluation of a leaking bolted connection rather than the Code required bolt removal and examination.

The guidance provided by the 1989 edition of Section XI of the ASME Code, the Code relief that allows bolt evaluation rather than removal and examination, and the EPRI Boric Acid Corrosion Guidebook are utilized by the CNP Boric Acid Corrosion Control Program. This ensures the effective identification and control of active boric acid corrosion conditions. Therefore, I&M considers that the inspections and evaluations described in the responses to the preceding questions demonstrate compliance with CNP Technical Specifications and Title 10 of the Code of Federal Regulations (10 CFR), Section 50.55(a), which incorporates Section XI of the ASME Code by reference.