INDIANA & MICHIGAN ELECTRIC COMPANY

P.O. BOX 16631
COLUMBUS, OHIO 43216

April 12, 1985
AEP:NRC:0906E

Donald C. Cook Nuclear Plant
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74
ADDITIONAL INFORMATION CONCERNING NRC
INSPECTION REPORT 50-315/84-13(DRS); 50-316/84-15(DRS)

Mr. James G. Keppler
U.S. Nuclear Regulatory Commission
Region III
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Dear Mr. Keppler:

This letter provides additional information concerning two issues addressed in the subject report. The first issue involves the implementation of activities related to a 1979 Interpretation (XI-01-79-18) of ASME Code, Section XI-1977. The second issue involves an assessment of our valve program to determine program modifications, which would be needed to assure adequate tracking of valve failure rates and to assure an adequate valve trending program is in place. Our letter AEP:NRC:0906B, dated January 30, 1985 indicated that a written description of the results of the assessment would be provided by April 1, 1985. Based on a discussion with Mr. P. Wohld of your staff on March 26, 1985, we agreed to provide by mid-April, 1985 a description of the results of the assessment and a description of how we will implement Interpretation XI-01-79-18.

Issue 1

Based on discussions with your staff, Interpretation XI-01-79-18, states that surveillance, of accessible valves with remote indication, are to be performed with an observer near the valves. The observer, along with another observer near the remote indicators, would assure agreement between valve movement and the remote indicators.

Per our agreement with your staff on March 27, 1985, the following describes the activities we will implement with regard to Interpretation XI-01-79-18 and the ISI valve program as discussed with the Office of Nuclear Reactor Regulation on March 5-7, 1985.

- Units 1 and 2 procedures will be revised by the end of the Unit 2 refueling outage (approximately February, 1986) to address the above. This will be done by revising our procedures to comply with Paragraph IWV-3300 of the ASME B&PV Code, Section XI-1980 Edition, plus Addendum through Winter 1981.

- o Unit 2 accessible valves equipped with remote indication will be surveilled per the revised procedures by the end of the Unit 2 refueling outage.

- o Those Unit 1 valves, which have no other check (such as a partial or full flow test) to indicate they will perform their safety function when needed, will be identified. The remote indications for the Unit 1 valves in this category will be verified per the 1970 Code by the end of the current Unit 1 refueling outage (approximately late-July, 1985) or the valves will be tested.

- o Unit 1 accessible valves equipped with remote indication will be surveilled per the revised procedures, during the next scheduled surveillance following the issuance of the revised procedure. These valves will be surveilled by the end of the next Unit 1 refueling outage (approximately October, 1986).

- o Should a problem be discovered during the surveillance of an accessible Unit 2 valve, the commensurate Unit 1 valve will be evaluated when available to determine whether remote indication should be verified.

Issue 2

The following is a description of the valve program modifications that have or will be implemented as a result of the aforementioned assessment, concerning valve failure rates and valve trending. Also provided are the target dates for completing the modifications:

- o Procedure No. 12-QHP-5070ISI.014 dated April 1, 1985 delineates (1) when valves are to be placed on increased frequency, (2) the criteria for determining a valve failure, and (3) provides criteria for valve trending.

- o In support of a trending program which is responsive to valve degradation, two aids will be developed. The first aid consists of a set of curves (graphs), which will be used to assure that valve degradation does not continue beyond the established limit. The second aid is a matrix, which identifies like valves that could be susceptible to a common failure. These aids are targeted to be completed by December 31, 1985.

- o To ensure that unacceptable valve degradation or valve failures are properly considered for trending and tracking, the appropriate plant department (ISI section) has been added to the distribution list for Condition Reports that document valve failures and unacceptable degradation.

- o Procedures are in place to assure that the respective Condition Reports are reviewed by cognizant personnel to identify the cause of the valve failures or unacceptable valve degradation, and the appropriate corrective actions.

o Data accumulated as part of the ISI valve trending program will be used as a tool in scheduling timely maintenance and replacement.

This document has been prepared following Corporate procedures which incorporate a reasonable set of controls to insure its accuracy and completeness prior to signature by the undersigned.

Very truly yours,

M. P. Alexich
Vice President

RBK 4/12/85

cm

cc: John E. Dolan
W. G. Smith, Jr. - Bridgman
R. C. Callen
G. Bruchmann
G. Charnoff
NRC Resident Inspector - Bridgman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

April 11, 1985

Docket Nos. 50-315
and 50-316

Mr. John Dolan, Vice President
Indiana and Michigan Electric Company
c/o American Electric Power Service Corporation
1 Riverside Plaza
Columbus, Ohio 43216

Dear Mr. Dolan:

In response to the Indiana and Michigan Electric Company letter dated November 28, 1984, we have completed our review of your method of performing the monthly pump test required by Technical Specification 4.7.1.2.a.2.b for the turbine driven pump for D. C. Cook, Units 1 and 2. The subject specification requires that the turbine driven pump be demonstrated operable every 31 days by verifying that "the steam turbine-driven pump develops a discharge pressure of $\geq$1285 psig at a flow of $\geq$ 700 gpm when the secondary steam pressure is greater than 310 psig." The plant procedures for performing this test provide for adjusting the turbine speed to 4350 rpm and establishing a flow of 700 gpm. The pump discharge pressure is then recorded and compared to the 1285 psig (corrected for temperature) minimum requirement of the specification.

Region 3 (refer to Inspection Reports 50-315/84-13 and 50-316/84-15) expressed a concern that the pressure, when corrected for temperature, allows the test to be found acceptable when the pressure is less than that stated in the surveillance requirements (1285 psig). While we agree with the licensee's philosophy for correcting the discharge pressure for temperature and with the correction factors identified in the procedures, we question the methods identified in the procedure for measuring the pumped fluid temperatures. The IMEC procedures for determining condensate temperature call for placing a handheld pyrometer against the suction piping. We do not believe this is an accurate method of determining condensate temperature. Further, since the correction factors involved with the range of temperatures expected in the condensate tank (40°F/+ 1 psig - 100°F/-8 psig) are small, and may be no greater than inherent instrument errors and reading inaccuracies, the correction factor obtained by this method appears unwarranted and unnecessary.

However, we do agree that a temperature correction factor for determining the required discharge pressure during the surveillance testing procedure does not invalidate the test results on a technical basis. Based on our review, we conclude that IMEC should take one of two actions:

1. Revise the specification to identify that the 1285 psig acceptance pressure is based on a fluid temperature of 60°F and develop a more accurate method of determining the temperature of the pump fluid, or

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2. Delete the correction factor from the surveillance procedure.

If you have any further questions on this matter, please let us know.

Sincerely,

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

cc:
W. Shafer, RIII
B. Jurgensen, RI

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

APR 11 1985

Docket No.: 50-341

MEMORANDUM FOR: Richard L. Spessard, Director
Division of Reactor Safety
Region III

FROM: Hugh L. Thompson, Jr., Director
Division of Licensing
Office of Nuclear Reactor Regulation

SUBJECT: RESPONSE TO REQUEST FOR TECHNICAL ASSISTANCE REGARDING MAXIMUM STROKE TIME TESTING FOR IST OF VALVES

We have reviewed the information submitted in your request for technical assistance dated November 14, 1984 regarding testing of the maximum stroke time as part of the in-service testing (IST) program at the Fermi-2 facility. Our basic position on this request is that the applicant has committed to comply with the requirements of the ASME Code and has not requested specific relief from the applicable portion of the ASME Code. Our response is directed towards the third concern outlined in your letter (i.e., the acceptability of baseline data established for valve testing in accordance with Section XI of the ASME Code) since the first two concerns were previously resolved.

Acceptability of Baseline Data Established for Valve Testing per Section XI

With respect to the applicant's procedures for measuring valve stroke times, as described in your letter dated November 14, 1984, the staff agrees that these procedures are not in accordance with the requirements of Section XI, Subsection IWV-3417 of the ASME Code (the Code). The use of such procedures would require prior written relief by the staff from the specific requirements of the Code.

The specific applicable Code requirements are:

IWV-3417 Corrective Action

(a) If, for power operated valves, an increase in stroke time of 25% or more from the previous test for valves with full-stroke times greater than 10 sec or 50% or more for valves with full-stroke times less than or equal to 10 sec is observed, test frequency shall be increased to once each month until corrective action is taken, at which time the original test frequency shall be resumed. In any case, any abnormality or erratic action shall be reported. (Emphasis added).

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Contact: M. Lynch, 492-7050

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(b) If a valve fails to exhibit the required change of valve stem or disk position or exceeds its specified limiting value of full-stroke time by this testing, then corrective action shall be initiated immediately. If the condition is not, or cannot be, corrected within 24 hours, the valve shall be declared inoperative. When corrective action is required as a result of tests made during cold shutdown, the condition shall be corrected before startup. A retest showing acceptable operation shall be run following any required corrective action before the valve is returned to service.

As cited above, each in-service test valve stroke time is required to be compared to the previous in-service test valve stroke time and is not related in any way to the design or purchase specification for a valve. Additionally, the staff does not interpret a corrective action to be the acceptance of the new stroke time measured on the first monthly test. When a valve has exceeded this criterion on one in-service test, the monthly frequency must be maintained until maintenance is performed on the valve so that it will not become inoperable.

It appears that the applicant's practice for establishing maximum limiting stroke times for valves is also inconsistent with the staff's interpretation of the Code. Subsection IWV is specifically a "component" test code and, therefore, requires that the owner specify the maximum limiting stroke times for each power operated valve (IWV-3413). It is the staff's position that these limiting values of full stroke time are required to be based on reasonable engineering judgement of component (valve) operability, not minimum system requirements. System (or component) response time limitations, as stated in the applicant's FSAR or in the plant Technical Specifications, are also time limitations placed on each subcomponent of that system (or component). However, the staff's position is that these response time limitations should rarely take precedence over a component-oriented limiting valve stroke time.

Inasmuch as the IST program requirements become applicable when Detroit Edison declares that the Fermi-2 facility has gone "commercial," you should bring this matter to its attention so that it can be properly resolved.

Frank J. Miraglia

for Hugh L. Thompson, Jr., Director
Division of Licensing
Office of Nuclear Reactor Regulation

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

October 19, 1984

MEMORANDUM FOR: R. L. Spessard, Director
Division of Reactor Safety, Region III

FROM: Darrell G. Eisenhut, Director
Division of Licensing, NRR

SUBJECT: RELIEF REQUESTS FROM LEAK TESTING REQUIREMENTS AS STATED IN SECTION XI, SUBSECTION IWV-3420 OF THE ASME CODE
TIA 84-62

REFERENCE: R. L. Spessard memorandum to D. G. Eisenhut dated July 19, 1984; Request for Technical Assistance - Relief Requests from Leak Testing Requirements as Stated in Section XI, Subsection IWV-3420 of the ASME Code (AITS F03043684)

Your July 19, 1984 memo noted that the Commission has granted relief from leak rate testing requirements of IWV-3420 for containment isolation valves and permitted 10 CFR Part 50, Appendix J, type C testing as an alternative. This practice has led to two questions:

1. Does granting such relief exempt licensees from specifying discreet or weighted leak rates for Category A valves addressed by the relief request?

2. Does granting such relief exempt licensees from leak rate analysis and corrective action requirements as stated in IWV-3426 and 3427, respectively as well as those requirements stated in IWV-3420 through IWV-3425?

As requested, we have reviewed the questions and the implications of the granting of exemptions from Section XI, IWV-3420 of the ASME Code. Section XI of the ASME Code requires individual testing for each component in the IST program, including individual acceptance criteria. Containment Isolation Valves (CIVs) are required to be individually included in the IST program because of their accident mitigation service requirements. However, since licensees are required to perform leak rate testing of CIVs in accordance with 10 CFR Part 50, Appendix J, NRR has routinely granted relief from the leak rate test requirements of the ASME Code for these components. For cases where this relief is granted the staff requires that the licensee still meet the Analysis of Leak Rates and Corrective Action requirements of the Code, paragraphs IWV-3426 and IWV-3427 of the 1980 Edition, respectively.

The staff believes that a "weighted" approach is the most appropriate method of assigning allowable leak rates. This method is based on the existence of a linear relationship between valve sizes with respect to allowable leakage (i.e., a 6" valve would be allowed twice the leakage of a 3" valve). Additionally, when the allowable leak rates are added up for all type C tested CIVs, the total should not exceed 0.6 L_A. This allows a certain amount of

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flexibility since the $0.6L_A$ value specified by Appendix J is the maximum allowed for the combined cumulative leak rates of type C tested CIVs and containment penetrations as determined by type B testing.

This completes NRR review pursuant to TIA 84-62.

Frank J. Miraglia
for Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

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