APR 0 2 1986

Docket Nos.: 50-315
and 50-316

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

Dear Mr. Dolan:

Subject: Response to TMI Action Item II.D.1 on Relief and Safety Valve Testing

In our review of the Indiana and Michigan Electric Company responses to TMI Action Item II.D.1 on Relief and Safety Valve Testing, we have developed with our contractor, EG&G, additional questions for clarification. It is requested that IMEC provide the necessary responses on a mutually agreeable schedule after first discussing the questions with this office. We suggest a telephone conference call at the earliest convenience so that the review may be completed. The conference call should be arranged thru the licensing project manager, Dave Wigginton.

Sincerely,

Original signed by:
D. Hood

for B. J. Youngblood, Director
PWR Project Directorate #4
Division of PWR Licensing-A

Enclosure: As stated

cc: See next page

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Mr. John Dolan
Indiana and Michigan Electric Company

Donald C. Cook Nuclear Plant

cc:
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UNRESOLVED QUESTIONS ON TMI ACTION NUREG-0737
Item II.D.1 for D. C. Cook Units 1 and 2

1. Safety Valve Inlet Pressure Drop

The licensee gave the pressure drop values through the inlet piping as 14.4 psi for the plant specific safety valve and 15.8 psi for the applicable EPRI test valve (see question 6, Reference 1). These pressure drop values appear to be extremely low, when compared with the pressure drop values listed in Table B-3 of Reference 2 for various EPRI valve inlet configurations. Please verify the pressure drop values given previously. Present a recalculation of the total pressure drops through the inlet piping of D. C. Cook Units 1 and 2 safety valves and the applicable EPRI inlet piping arrangement. The total pressure drop should include both the frictional and accoustic wave components evaluated under steam conditions.

2. Safety Valve Backpressure

EPRI Test No. 1411, which was selected as a representative test of the D. C. Cook safety valve, showed that the maximum backpressure developed at the outlet of the test valve was 245 psia. The backpressure for the D. C. Cook safety valve was not given in the submittal. Please provide the predicted value of the maximum backpressure developed at the outlet of the D. C. Cook Units 1 and 2 safety valves in order to verify whether the plant specific pressure drop is bounded by the EPRI test.

3. PORV Control Circuitry

In response to the question on the qualification of the PORV control circuitry (Reference 1), the licensee stated that their procedures used in the design, procurement, test and maintenance of the circuitry were considered sufficient to ensure qualification. No specific information required for the qualification was presented. To satisfy the requirements of NUREG-0737, Item II.D.1, the licensee must provide evidence supported by test to demonstrate that the PORV associated control circuitry can be operated under all normal and accident conditions. Alternatively, it would also be acceptable to NRC, if the licensing requirements of 10 CFR 50.49 for electric equipment qualification is satisfied. Therefore, verify whether the D. C. Cook PORV control circuitry has been included in the 10 CFR 50.49 review. Additional qualification of the PORV control circuitry is not needed, if the control circuitry has already been reviewed and approved under 10 CFR 50.49. Otherwise provide adequate information as required.

4. Safety Valve and PORV Piping Stress Analysis

   a. Reference 1 indicates that details of the piping stress analysis for PORV discharge cases are contained in the Teledyne Reports TR-5364-1 and TR-5364-2. Our record shows that the above reports have not been submitted for review by the licensee. Please provide copies of the above reports so that the piping analysis pertinent to PORV discharge can be reviewed.

   b. In the thermal hydraulic analysis of the piping discharge conditions, the computer codes REPIPE/SAP2SAP were used to calculate the fluid force time histories from RELAP5 analysis results. Provide verification of the REPIPE/SAP2SAP codes by showing that these codes produce accurate results for similar problems.

   c. The licensee presented a verification of the structural analysis code, TMRSAP, using assumed force time histories in the form of ramp functions to represent the fluid forces exerted on the piping model (Question 16, Reference 1). In order to demonstrate that the computer code produces accurate results for the valve discharge problem analyzed, similar piping model and fluid forces must be used. The ramp shape force time histories used in the verification problem does not realistically represent the type of fluid forces found in a safety valve/PORV discharge event. Therefore, present a verification of the TMRSAP code using fluid forces representative of the valve discharge conditions.

d. The piping stress analysis presented by the licensee did not include a faulted service condition which called for the combination of SSE (Safe Shutdown Earthquake) and the safety valve/PORV discharge loads as suggested in Reference 2. The licensee contended that this load combination was not considered, because it was not a part of the original design basis as evidenced by the D. C. Cook FSAR, Table 2.9-2, Part B and C. It further stated that the likelihood of a SSE earthquake concurrently occurring during a SRV event seemed highly remote and was not included in the C2 emergency load combination (Reference 1). However, in the "Staff Recommendations on Event Combinations" presented on page 1-1 of Reference 3, the NRC staff do not deem it unduly conservative to combine the piping of dynamic loads such as water hammer, safety relief valve discharge, etc. with postulated earthquakes. Accordingly, a faulted condition considering the combination of SSE and valve discharge cannot be ignored without proper justification. Therefore, provide evidence which justifies that this load combination need not be considered.

5. Piping Support Stresses

The Teledyne stress report indicates that a number of pipe supports (8 in Unit 1 and 24 in Unit 2) exceeded their design loads. Subsequent reanalysis performed by the licensee concluded that, with the exception of a few supports which required modifications, the rest of the supports were found to be adequate to resist the revised loads. Since the licensee did not provide any detail concerning the reanalysis, a review of the support adequacy cannot be performed. Therefore provide the following information required for the evaluation.

a. Identify the governing code and standards used for support design.

b. List the load combination equations used including all the operation conditions as suggested in the EPRI Guide for Plant Specific Evaluations (Reference 2). Also specify the allowable stress limit applicable to each condition.

c. Provide a comparison of the worst case support stresses (or loads) with the applicable allowable stress limits (or allowable loads) and identify the load combinations associated with the worst stresses. These data should be presented for the questionable supports indicated in the Teledyne stress report.

6. Valve Accelerations

The licensee stated in Reference 4 that the Teledyne analysis showed that the maximum vertical acceleration of the D. C. Cook safety valve resulting from a piping discharge transient exceeded the design allowable of 2 g by less than 1 g. The licensee contended that these valves were successfully tested in the Westinghouse seismic tests to 4 g in the vertical direction. Therefore, the predicted acceleration for the D. C. Cook valves, although exceeding the design allowable, would still be acceptable. However, the licensee did not provide any details of the test and its findings. Since NUREG-0737, Item II.D.1 requires that the valve operability must be demonstrated for operating and transient conditions, the operability of the valves under the predicted g loadings must be assured. Therefore, provide a discussion of the seismic test performed by Westinghouse to demonstrate that the operability of the safety valve will not be impaired under the predicted acceleration level.

REFERENCES

1. M. P. Alexich, Indiana & Michigan Electric Co., letter to M. R. Denton, NRC, "Response to NRC, Request for Additional Information Regarding Testing of the Pressurizer Safety and Relief Valves, Donald C. Cook Nuclear Plant, Units 1 and 2," AEP:NRC:0585H, July 15, 1985.

2. EPRI PWR Safety and Relief Valve Test Program Guide for Application of Valve Test Program Results to Plant-Specific Evaluations, Revision 2, Interim Report, July 1982.

3. U.S. Nuclear Regulatory Commission, Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, Evaluation of Other Loads and Load Combinations, NUREG-1061, Volume 4, December 1984.

4. M. P. Alexich, Indiana & Michigan Electric Co., letter to H. R. Denton, NRC, "Donald C. Cook Nuclear Plant Unit 1 and 2, NUREG-0737, Item II.D.1, PWR Relief and Safety Valve Test Program," AEP:NRC:0585D, December 15, 1983.