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

MAR 26 1979

Docket Nos.: 50-329
50-330

Consumers Power Company
ATTN: Mr. S. H. Howell
Vice President
212 West Michigan Avenue
Jackson, Michigan 49201

Gentlemen:

SUBJECT: CORRECTION TO LETTER OF MARCH 22, 1979

The enclosed requests for additional information on the FSAR for Midland Plant, Units 1 & 2, were inadvertently omitted in my letter of March 22, 1979 and were transmitted by telecopier to Mr. Jim Zabritski of your organization on March 23, 1979. Please add these enclosed requests to those forwarded previously.

Sincerely,

Steven A. Varga

Steven A. Varga, Chief
Light Water Reactors Branch No.4
Division of Project Management

Enclosures:
Requests 121.23
through 121.32

cc: See next page

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Consumers Power Company

ccs:
Michael I. Miller, Esq.
Isham, Lincoln & Beale
Suite 4200
One First National Plaza
Chicago, Illinois 60670

Judd L. Bacon, Esq.
Consumers Power Company
212 West Michigan Avenue
Jackson, Michigan 49201

Mr. Paul A. Perry
Secretary
Consumers Power Company
212 W. Michigan Avenue
Jackson, Michigan 49201

Myron M. Cherry, Esq.
One IBM Plaza
Chicago, Illinois 60611

Mary Sinclair
5711 Summerset Drive
Midland, Michigan 48640

Frank J. Kelley, Esq.
Attorney General
State of Michigan Environmental
  Protection Division
720 Law Building
Lansing, Michigan 48913

Mr. Wendell Marshall
Route 10
Midland, Michigan 48640

Grant J. Merritt, Esq.
Thompson, Nielsen, Klaverkamp & James
4444 IDS Center
80 South Eighth Street
Minneapolis, Minnesota 55402

121.0 MATERIALS ENGINEERING BRANCH - MATERIALS INTEGRITY SECTION

121.23
(5.2.4)
RSP

The preservice inspection program for the Midland Plant, for ASME Code Class 1, 2, and 3 components (letter, S. H. Howell to R. S. Boyd, CPCo Serial 5930, October 5, 1978, submitted in response to NRC Questions 121.1, 121.3 and 121.14) is not adequate. The proposed preservice inspection program does not reference the edition and addenda of Section XI of the ASME Code required by 10 CFR Part 50, Section 50.55a, nor the edition and addenda of the ASME Code that will be required by the proposed change to this regulation (FEDERAL REGISTER, Vol. 44, No. 13, January 18, 1979, pp. 3719-3721).

It is our position that you submit a preservice inspection program for the Midland Plant that complies with 10 CFR Part 50, Section 50.55a.

121.24
(5.4.2)
(16.0)
RSP

The proposed steam generator inspection program contained in the Midland Plant FSAR and Technical Specifications, is not acceptable.

Response to Question 121.3 indicates that a preservice eddy current and ultrasonic examination and inspection of the steam generators has been conducted. Provide a description of the preservice inspection and a summary of the inspection results (FSAR Section 5.4.2 and Technical Specification Section 16.3/4.4.5).

It is our position that Section 16.3/4.4.5 (applicability and bases) of the Midland Plant Technical Specification be revised to be consistent with the corresponding section of NUREG-0103, "Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors."

121.25
(5.2)

Midland Plant FSAR Table 5.2.3 indicates that SA-533 Grade B Class 1 plate material is used in the fabrication of NSS-12 and NSS-13 reactor vessels. Your response to NRC Question 121.10 indicated no use of this material. Clarify this discrepancy.

121.26
(5.2)
(5.3)

Table 5.2-1 of the Midland Plant FSAR indicates that components of the reactor coolant pressure boundary were ordered and constructed to editions and addenda of the ASME Code that were effective prior to the issuance of 10 CFR Part 50, Appendix G. Section 5.3.1.5 of the FSAR discusses the difference in fracture toughness requirements between the ASME Code and 10 CFR 50, Appendix G. Table 5.3-2 of the FSAR lists fracture toughness test results for the materials of the reactor beltline region and lists the estimated fracture toughness for materials in other areas of the reactor vessel and other components in the reactor coolant pressure boundary. Babcock and Wilcox Topical

Report BAW-10046A, "Methods of Compliance with Fracture Toughness and Operational Requirements of 10 CFR Part 50, Appendix G," was used to provide the estimated fracture toughness values in Table 5.3-2.

Sections III and IX of the ASME Code require mechanical testing of materials to be used throughout the reactor coolant pressure boundary, not only beltline region materials. To demonstrate compliance with Appendix G to 10 CFR Part 50 and to demonstrate applicability of the estimations of BAW-10046A, supply the results of the ASME Code required tests (i.e., test required, ASME Code paragraph, yield stress, ultimate tensile stress, impact energy, lateral expansion, test temperatures) for all of the ferritic materials used in the reactor coolant pressure boundary.

Identify any of the reactor coolant pressure boundary material test results that were obtained prior to developing standard documentation to demonstrate personnel competency in materials testing (Paragraph II.B.4, Appendix G to 10 CFR Part 50, item of non-compliance identified in CPCo response to NRC Question 121.17).

121.27 (5.3) Paragraph IV.A.4 of Appendix G, 10 CFR Part 50, requires that all bolting over one inch nominal diameter meet a minimum of 25 mils lateral expansion and 45 foot-pounds as determined by Charpy V-notch tests. Section 5.3.1.7 of the Midland Plant FSAR presents tensile strength and Charpy V-notch energy data for the reactor vessel fasteners only.

Confirm that the reactor vessel fasteners are the only bolting over one inch nominal diameter, or supply the required test results (and acceptance standards used if different from Appendix G) for any other bolting material in this size classification. As specified by Appendix G to 10 CFR Part 50, bolting includes bolts, nuts and washers.

Identify any of the bolting material test results that were obtained prior to developing standard documentation to demonstrate personnel competency in materials testing (Paragraph II.B.4, Appendix G of 10 CFR Part 50, item of non-compliance identified in CPCo response to NRC Question 121.17).

121.28 (5.3) Babcock and Wilcox Topical Report BAW-10056A, "Radiation Embrittlement Sensitivity of Reactor Pressure Vessel Steels," dated August 1973, is referenced in Section 5.3.1 of the Midland Plant FSAR. This report presents background information and materials test results that were used to formulate radiation damage curves. In July 1975, the NRC issued Regulatory

Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," (Revision 1 issued April 1977) which presents radiation damage curves acceptable to the NRC staff.

In response to NRC Questions 121.5, 121.12, 121.18 and 121.21, CPCo has committed to fully implement the recommendations of this regulatory guide for the Midland Plant.

Consequently, Babcock and Wilcox Topical Report BAW-10056A is not applicable to the Midland Plant licensing review and reference to this topical report in the FSAR should be deleted.

121.29 (5.3) (16.0) To demonstrate compliance with Appendix H to 10 CFR Part 50, include in the Midland Plant FSAR and Technical Specifications a table that provides the following information for each surveillance specimen capsule:

(1) The actual surveillance materials in each capsule.
(2) The test specimen type(s) made from each material.

121.30 (5.3) (16.0) Revise Table 5.3-7 of the Midland Plant FSAR to show the following for each surveillance specimen capsule:

(1) Proposed loading schedule of capsules into the reactor vessels.

(2) Indicate the specific surveillance capsules that will be placed in the locations identified in Figure 5.3-6.

(3) Proposed time of capsule withdrawal (calendar years and effective full power years).

Incorporate this table into the Technical Specifications for the Midland Plant (Table 4.4-5).

121.31 (5.3) Babcock and Wilcox Topical Report BAW-10100A, "Reactor Vessel Material Surveillance Program, Compliance with 10 CFR 50, Appendix H, for Oconee Class Reactors," dated February 1975, is referenced in Section 5.3.1 of the Midland Plant FSAR. This report presents discussions on surveillance specimen capsules, surveillance specimen holder tubes, neutron flux lead factor, radiation damage, holder tube mounting locations, surveillance specimen types and number.

Due to operating problems experienced by the surveillance specimen capsules and holder tubes, Babcock and Wilcox has redesigned the holder tubes and changed the mounting locations resulting in different neutron flux lead factors. The capsule itself has been redesigned to hold different surveillance specimen types and quantities. Also, as discussed in NRC Question 121.28, the radiation damage curves as presented in this topical report, and in BAW-10056A, are no longer used in the Midland Plant FSAR.

Consequently, Babcock and Wilcox Topical Report BAW-10100A is not applicable to the Midland Plant licensing review and reference to this topical report in the FSAR should be deleted.

121.32
(16.0)

Figure 4-1, "Fast Neutron Fluence (E > 1 MeV) as a Function of Full Power Service Life," Figure 4-2, "Effect of Fluence and Copper on Shift of RT_{NDT} for Reactor Vessel Steels Exposed to 550 F Temperature," and Table 4-1, "Reactor Vessel Toughness," of the Midland Plant Technical Specifications have been left blank. Supply this information.