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Docket Nos. 50-329/330
MS 12-12

MEMORANDUM FOR: S. A. Varga, Chief
Light Water Reactors Branch No. 4
Division of Project Management

FROM: S. S. Pawlicki, Chief
Materials Engineering Branch

SUBJECT: CONSUMERS POWER COMPANY (CPCo), MIDLAND PLANT, UNITS 1 AND 2

Plant Name: Midland Plant
Suppliers: Babcock and Wilcox; Bechtel
Licensing Stage: OL
Docket Numbers: 50-329/330
Responsible Branch and Project Manager: LWR 4; D. Hood
Reviewer: M. L. Boyle
Description of Task: Q-2 Supplement
Review Status: Information Required

In a memorandum from S. S. Pawlicki to S. A. Varga, dated January 17, 1979, the Materials Integrity Section, Materials Engineering Branch, Division of Systems Safety, identified four review areas that required additional information from CPCo before the safety evaluation for the Midland Plant could be completed. MTEB received Revision 18 to the Midland Plant FSAR on March 2, 1979 and we have evaluated the information contained in this revision. We have reached a conclusion that two of the areas identified in our memorandum are now adequately described in the FSAR. These areas are prediction of radiation damage to RV materials and effects of atypical weld material on pressure-temperature limits.

We have also concluded that the following two areas require additional information before we may complete our review of the materials integrity of the Midland Plant.

1. Non-Compliance with Appendices G and H, 10 CFR 50

Certain areas of non-compliance with these regulations have been identified by MTEB and by CPCo (response to Question 121.17). The response to this question, and the pertinent FSAR Sections, have proved to contain insufficient information to fully justify the issuance of any exceptions to the regulations.

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We require CPCo to provide the information identified in the attached questions so that we can complete our review of this item.

2. Preservice and Inservice Inspection Programs, 10 CFR 50.55a

CPCo submitted proposed preservice and inservice inspection programs, CPCo letter dated October 5, 1978. These programs do not reference the edition and addenda of Section XI of the ASME Code required by 10 CFR 50.55a, nor the edition and addenda that will be required by the proposed change to this regulation (FEDERAL REGISTER, January 18, 1979).

Therefore, in order to complete our review of the preservice program, we require CPCo to submit a revised preservice inspection program that complies with the requirements of 10 CFR 50.55a. Since the inservice inspection program is required to comply with the edition and addenda of Section XI of the ASME Code that is in effect six months prior to the date of commercial operation, we do not require the details of the inservice inspection program to complete our SER review.

Upon receipt of adequate responses to the attached questions, we will prepare our SER input and submit it to DPM.

Original signed by
S. S. Pawlicki

S. S. Pawlicki, Chief
Materials Engineering Branch
Division of Systems Safety
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc w/encl:
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cc w/o encl:
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Distribution:
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RE 1.1-1

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OFFICE	DSS:MTEB	DSS:MTEB	DSS:MTEB			
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DATE	3/19/79	3/19/79	3/22/79			

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 CPCo 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 NRC 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. CPCo 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 de require mechanical testing of materials to be used througho the reactor coolant pressure boundary, not only beltline regi 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 para aph, 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. Sufficient information has been provided in the FSAR and other Babcock and Wilcox topical reports, and with the CPCo commitment to fully implement the recommendations of Regulatory Guide 1.99, we require no additional information in this area.

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.