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SUBJECT: Responds to GL 92-01,rev 1 re reactor vessel structural integrity.Supporting info.

See Report

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NOTES: See WCAP-8047P
WCAP-11902
WCAP-12483
WCAP-8512

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AEP:NRC:1173

Donald C. Cook Nuclear Plant Units 1 and 2
Docket No. 50-315 and 50-316
License No. DPR-58 and DPR-74
GENERIC LETTER 92-01, REVISION 1
REACTOR VESSEL STRUCTURAL.INTEGRITY

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Attn: T. E. Murley

July 13, 1992

Dear Dr. Murley:

The purpose of this letter is to respond to Generic Letter 92-01, Revision 1. This Generic Letter was issued to ensure that licensees and permit holders are complying with 10 CFR 50.60 and 10 CFR 50.61, and are fulfilling commitments made in response to Generic Letter 88-11. The reponses to the specific questions raised in Generic Letter 92-01 for the Donald C. Cook Nuclear Plant, Units 1 and 2, are provided in the enclosure to this letter.

This letter is submitted pursuant to 10 CFR 50.54(f) and, as such, an oath of affirmation is enclosed.

Sincerely,

E E Fitzpatrick

E. E. Fitzpatrick
Vice President

dag

Enclosure w/Attachments

Charge: NRC PDR Ltr. Encl. 1 1NP
NSIC 1 1NP
A028 1/1

cc: D. H. Williams, Jr.
A. A. Blind - Bridgman
J. R. Padgett
G. Charnoff
NFEM Section Chief
A. B. Davis - Region III
NRC Resident Inspector - Bridgman

STATE OF OHIO)
COUNTY OF FRANKLIN)

E. E. Fitzpatrick, being duly sworn, deposes and says that he is the Vice President of licensee Indiana Michigan Power Company, that he has read the forgoing Response to Generic Letter 92-01 and knows the contents thereof; and that said contents are true to the best of his knowledge and belief.

E E Fitzpatrick

Subscribed and sworn to before me this 13th

day of July, 1992.

NOTARY PUBLIC
Commission expires 3-9-96

ENCLOSURE TO AEP:NRC:1173

RESPONSES TO GENERIC LETTER 92-01 QUESTIONS

Question 1

"Certain addressees are requested to provide the following information regarding Appendix H to CFR Part 50:

> Addressees who do not have a surveillance program meeting ASTM E 185-73, -79, or -82 and who do not have an integrated surveillance program approved by the NRC (see Enclosure 2), are requested to describe actions taken or to be taken to ensure compliance with Appendix H to 10 CFR Part 50. Addressees who plan to revise the surveillance program to meet Appendix H to 10 CFR Part 50 are requested to indicate when the revised program will be submitted to the NRC staff for review. If the surveillance program is not to be revised to meet Appendix H to 10 CFR Part 50, addressees are requested to indicate when they plan to request an exemption from Appendix H to 10 CFR Part 50 under 10 CFR 50.60(b)."

Response to Question 1

Unit 1

The Unit 1 Reactor Vessel Material Surveillance Program was developed in accordance with ASTM E-185-70, and is documented in Westinghouse WCAP-8047 (Section 1, page 1-1). This report documents the pre-irradiation testing of the reactor vessel materials and compliance with Appendix G of ASME Code Section III. We believe that the surveillance program noted in WCAP-8047 is in compliance with Appendix H to CFR part 50. A copy of this report is included as Attachment 1.

Unit 2

The Unit 2 Reactor Vessel Material Surveillance Program was developed in accordance with ASTM E-185-73, and is documented in Westinghouse WCAP-8512 (Section 1, page 1-1). This report documents the pre-irradiation testing of the reactor vessel materials and compliance with Appendix G of ASME Code Section III. We believe that the surveillance program noted in WCAP-8512 is in compliance with Appendix H to 10CFR part 50. A copy of this report is included as Attachment 2.

Question 2.a

"Certain addressees are requested to provide the following information regarding Appendix G to 10 CFR Part 50:

Addressees of plants for which the Charpy upper shelf energy is predicted to be less than 50 foot-pounds at the end of their licenses using the guidance in Paragraphs C.1.2 or C.2.2 in Regulatory Guide 1.99, Revision 2, are requested to provide to the NRC the Charpy upper shelf energy predicted for December 16, 1991, and for the end of their current license for the limiting beltline weld and the plate or forging and are requested to describe the actions taken pursuant to Paragraphs IV.A.1 or V.C of Appendix G to 10 CFR Part 50."

Response to Question 2.a

Unit 1

As identified in the Unit 1 Capsule U Analysis-WCAP 12483 (Section 1, page 1; Section 5.2, Tables 5-1 through 5-6 and Figures 5-1 to 5-5), all materials exhibit an upper shelf energy level greater than 50 ft-lbs for plant operation through 32 effective full power years (EFPY).

WCAP 12483 was submitted to the NRC with our letter AEP:NRC:0894M dated June 20, 1990 and is included as Attachment 3.

Unit 2

As identified in Unit 2 Capsule X Analysis-SWRI Report 06-8888 (Section 1, page 2; Section 4.3, page 36, Tables 4.19-4.22 and Figures 5 through 8), all materials exhibited an average Charpy upper shelf energy greater than 50 ft-lbs at a fluence of 1.002 x 10^{19} n/cm^2 for 12 EFPY. The upper shelf energy of 50 ft-lbs is not expected to be exceeded based on 32 EFPY fluence projections (page 52). SWRI Report No. 06-8888 was submitted to the NRC with our letter AEP:NRC:0894I dated June 11, 1987 and is included as Attachment 4.

Unit 2 Capsule U was removed in May 1992, and will be analyzed. Results of this analysis will include updated Charpy projections to 32 EFPY.

Question 2.b.(1)

"Addressees whose reactor vessels were constructed to an ASME Code earlier than the Summer 1972 Addenda of the 1971 Edition are requested to describe the consideration given to the following material properties in their evaluations performed pursuant to 10 CFR 50.61 and Paragraph III.A of 10 CFR Part 50, Appendix G.

The results from all Charpy and drop weight tests for all unirradiated beltline materials, the unirradiated reference temperature for each beltline material, and the method of determining the unirradiated reference temperature from the Charpy and drop weight test."

Response to Question 2.b.(1)

Unit 1

The Cook Nuclear Plant Unit 1 reactor vessel was constructed to the 1965 Edition through 1966 Winter Addenda, Code Cases 1332-2, 1358, 1339-2, 1335, 1359-1, 1338-3, and 1336 of the ASME Boiler and Pressure Vessel Code, Section III, Class A. A copy of the applicable UFSAR pages is included as Attachment 5.

The results of the Charpy and drop weight tests for all unirradiated beltline materials and the method of determining their unirradiated reference temperature is presented in WCAP 8047 (Section 3.0), "Reactor Vessel Material Surveillance Program." A copy of this report is included as Attachment 1.

Unit 2

The Cook Nuclear Plant Unit 2 reactor vessel was constructed to the 1968 Edition through 1968 Summer Addenda, Code Case 1335-4 of the ASME Boiler and Pressure Vessel Code, Section III, Class A. A copy of the applicable UFSAR pages is included as Attachment 5.

The results of the Charpy and drop weight tests for all unirradiated beltline materials and the method of determining the unirradiated reference temperature is presented in WCAP 8512 (Section 3.0), "Reactor Vessel Material Surveillance Program," Reference 2. A copy of this report is included as Attachment 2.

Question 2.b.(2)

"The heat treatment received by all beltline and surveillance materials."

Response to Question 2.b.(2)

Unit 1

The heat treatment received by all beltline materials is identified in a letter from J. Tillinghast to the NRC dated November 7, 1977. A copy of this letter is included as Attachment 6.

The heat treatment received by all surveillance materials is identified in WCAP-8047 (Appendix A), and a letter from J. Tillinghast to the NRC dated November 7, 1977. Copies of these documents are included as Attachments 1 and 6, respectively.

Unit 2

The heat treatment received by all beltline materials is identified in Unit 2 FSAR Appendix Q Amendment 77 dated July 1977. A copy of this document is included as Attachment 7.

The heat treatment received by all surveillance materials is identified in the WCAP 8512 (Appendix A), and Unit 2 FSAR Appendix Q Amendment 77 dated July 1977. Copies of these documents are included as Attachments 2 and 7, respectively.

Question 2.b.(3)

"The heat number for each beltline plate or forging and the heat number of wire and flux lot number used to fabricate each beltline weld."

Response to Question 2.b.(3)

Unit 1

The heat number for each beltline plate and the heat number of wire and flux lot number used to fabricate each beltline weld are presented in a letter from J. Tillinghast to the NRC dated November 7, 1977. A copy of this document is included as Attachment 6. Additional supporting information was previously provided in our letter AEP:NRC:0097C dated July 3, 1979 which responded to I&E Bulletin 78-12, "Atypical Weld Material in Reactor Pressure Vessel Welds". A copy of this document is included as Attachment 8.

Unit 2

The heat number for each beltline plate and the heat number of wire and flux lot number used to fabricate each beltline weld are presented in Unit 2 FSAR Appendix Q Amendment 77 dated July 1977. A copy of this document is included as Attachment 7. Additionally, supporting information was provided in our letter AEP:NRC:0097 dated June 1, 1979 which responded to I&E Bulletin 78-12, "Atypical Weld Material in Reactor Pressure Vessel Welds." A copy of this document is included as Attachment 9.

Question 2.b.(4)

"The heat number for each surveillance plate or forging and the heat number of wire and flux lot number used to fabricate the surveillance weld."

Response to Question 2.b.(4)

Unit 1

The heat number for each surveillance plate and heat number of wire and flux lot number used to fabricate the surveillance welds are presented in WCAP 8047 (Appendix A), a letter from J. Tillinghast to the NRC dated November 7, 1977, and a letter from Westinghouse to AEPSC dated June 14, 1985. Copies of these documents are included as Attachments 1, 6 and 10, respectively.

Unit 2

The heat number for each surveillance plate and heat number of wire and flux lot number used to fabricate the surveillance welds are presented in WCAP 8512 (Appendix A), and Unit 2 FSAR Appendix Q Amendment 77, dated July 1977. Copies of these documents are included as Attachments 2 and 7, respectively.

Question 2.b.(5)

"The chemical composition, in particular the weight in percent of copper, nickel, phosphorous, and sulfur for each beltline and surveillance material."

Response to Question 2.b.(5)

Unit 1

The chemical composition for each beltline material is identified in a letter from J. Tillinghast to the NRC dated November 7, 1977. A copy of this document is included as Attachment 6. Additional supporting information was previously provided in our letter AEP:NRC:0097C dated July 3, 1979 which responded to I&E Bulletin 78-12, "Atypical Weld Material in Reactor Pressure Vessel Welds". A copy of this document is included as Attachment 8.

The chemical composition of the surveillance material is identified in WCAP 8047 (Appendix A), and in a letter from J. Tillinghast to the NRC dated November 7, 1977. Copies of these documents are included as Attachments 1 and 6, respectively.

Unit 2

The chemical composition for each beltline material is identified in Unit 2 FSAR Appendix Q Amendment 77, dated July 1977. A copy of this document is included as Attachment 7. Additionally, supporting information was provided in our letter AEP:NRC:0097 dated June 1, 1979 which responded to I&E Bulletin 78-12, "Atypical Weld Material in Reactor Pressure Vessel Welds." A copy of this document is included as Attachment 9.

The chemical composition of the surveillance material is identified in WCAP 8512 (Appendix A), and the Unit 2 FSAR Appendix Q Amendment 77 dated July 1977. Copies of these documents are included as Attachments 2 and 7, respectively.

Question 2.b.(6)

"The heat number of the wire used for determining the weld metal chemical composition if different than Item (3) above."

Response to Question 2.b.(6)

Unit 1

As identified in Attachment 6, the circumferential girth seam weld between the intermediate and lower shell (limiting weld) was fabricated with weld wire heat number IP3571 and Linde 1092 flux lot number 3958. The chemical analysis for this wire/flux combination has been identified in a letter from Westinghouse to AEPSC dated June 14, 1985. This is included here as Attachment 10. This letter was submitted to the NRC with our submittal AEP:NRC:0894C. Additional supporting information was previously provided in our letter AEP:NRC:0097C dated July 3, 1979 which responded to I&E Bulletin 78-12, "Atypical Weld Material in Reactor Pressure Vessel Welds". A copy of this document is included as Attachment 8.

Unit 2

See response to Question 2.b.(3)

Question 3.a

"Addressees are requested to provide the following information regarding commitments made to respond to GL 88-11:

How the embrittlement effects of operating at an irradiation temperature (cold leg or recirculation suction temperature) below 525 °F were considered. In particular licensees are requested to describe consideration given to determining the effect of lower irradiation temperature on the reference temperature and on the Charpy upper shelf energy."

Response to Question 3.a

Unit 1

Cook Nuclear Plant Unit 1 has only operated at an irradiation temperature (cold leg) below 525°F for Cycles 11 and 12. The approximate cold leg temperature for Cycles 11 and 12 was 518°F and was the result of our Reduced Temperature and Pressure (RTP) program that was initiated after Cycle 10. The RTP program was initiated to reduce the operating temperatures and pressures with the intent of reducing steam generator U-tube stress corrosion cracking. The RTP program and associated technical specification changes were submitted to the NRC with our letter AEP:NRC:1067 dated October 14, 1988, and were approved in an NRC Safety Evaluation Report dated June 9, 1989. A copy of the Safety Evaluation Report is included as Attachment 11.

As identified in Section 3.10.1.2 of WCAP 11902, an evaluation of the rerating impact on reactor vessel integrity for neutron embrittlement was performed and judged not to have any significant impact on 10CFR Part 50, Appendix G requirements. This is in part due to the use of the modified low leakage core designs in Unit 1 since Cycle 8. As such, the Unit 1 fluence projections for end of vessel life have decreased from earlier predicted values. A copy of the applicable WCAP 11902 pages are included as Attachment 12.

Unit 2

Cook Nuclear Plant Unit 2 does not operate at an irradiation temperature (cold leg) below 525°F. The Unit 2 cold leg temperature is 541°F, as identified in the FSAR. A copy of the applicable FSAR page is included as Attachment 13. Additionally, modified low leakage cores have been used in Unit 2 from the second fuel cycle. As a result, the Unit 2 fluence projections for end of vessel life have remained well within earlier projections.

Question 3.b

"How their surveillance results on the predicted amount of embrittlement were considered."

Response to Question 3.b

Unit 1

AEPSC's response to Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations," was submitted to the NRC in our letter AEP:NRC:0894K dated December 5, 1988. This submittal was prepared utilizing the results from previously analyzed capsules and identified actions to be taken. For instance, Capsule U was pulled and analyzed in 1990 and the results submitted to the NRC in our letter AEP:NRC:0894M dated June 20, 1990. Revised heatup and cooldown curves and LTOP calculations were submitted to the NRC in our submittal letter numbers AEP:NRC:0894O, AEP:NRC:0894Q and AEP:NRC:0894R. Additionally, based on the results of the Unit 1 Capsule U analysis, a PTS evaluation was performed. Copies of AEPSC's Capsule U analysis, response to Generic Letter 88-11, and the PTS evaluation are included as Attachments 3, 14, and 15, respectively.

Unit 2

As identified for Unit 1, AEPSC's response to Generic Letter 88-11 is based on results from previously analyzed capsules. The PTS evaluation for Unit 2 was approved by the NRC in an SER dated March 27, 1987. Unit 2 Capsule U was removed in May 1992 and will be analyzed within the next year. Results of this analysis will be used to update a Unit 2 RT_{PTS} evaluation.

Question 3.c

"If a measured increase in reference temperature exceeds the mean-plus-two standard deviations predicted by Regulatory Guide 1.99, Revision 2, or if a measured decrease in Charpy upper shelf energy exceeds the value predicted using the guidance in Paragraph C.1.2 in Regulatory Guide 1.99, Revision 2, the licensee is requested to report the information and describe the effect of the surveillance results on the adjusted reference temperature and Charpy upper shelf energy for each beltline material as predicted for December 16, 1991, and for the end of its current license."

Response to Question 3.c

Unit 1

No data from the Cook Nuclear Plant reactor vessel material surveillance program exceeds the mean-plus-two standard deviation bound predicted by Regulatory Guide 1.99 Rev. 2. Refer to Attachment 14 (specifically refer to pages 3 and 4 of Attachment 2 contained within Attachment 14).

Unit 2

No Cook Nuclear Plant reactor vessel material surveillance program exceeds the mean-plus-two standard deviation bound predicted by Regulatory Guide 1.99 Rev. 2. Refer to Attachment 14 (specifically refer to pages 3 and 4 of Attachment 2 contained within Attachment 14).