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AEP:NRC:0894K
Reg. Guide 1.99, Rev. 2

Donald C. Cook Nuclear Plant Units 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74
GENERIC LETTER 88-11, "NRC POSITION ON RADIATION EMBRITTLEMENT OF REACTOR VESSEL MATERIALS AND ITS IMPACT ON PLANT OPERATIONS"

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

Attn: T. E. Murley

December 5, 1988

Dear Dr. Murley:

This submittal and its attachments are in response to NRC Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations," dated July 12, 1988. The Generic Letter requires the use of Revision 2 to Regulatory Guide 1.99 to predict the effect of neutron radiation embrittlement on the reactor vessel materials as required by paragraph V.A of 10 CFR Part 50, Appendix G.

The attachments provide the results of the analysis as applicable to the Donald C. Cook Nuclear Plant. Attachment No. 1 gives a summary of the evaluation as well as planned actions/schedules, and Attachment No. 2 is the analysis report from our consultant, Southwest Research Institute (SwRI).

Among the findings reported in Attachment No. 2, SwRI has concluded that the Cook Unit 2 reactor vessel materials in the beltline region are projected to retain sufficient toughness to meet the current requirements of 10 CFR 50 Appendix G for the duration of the design life of the unit, 32 EFPY. For Unit 1, the plot of upper shelf energy decrease versus fluence, indicates adequate toughness for the surveillance capsule specimens of the controlling weld material through 32 EFPY.

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

Sincerely,

M. P. Alexich
Vice President

MPA/eh

Attachments

cc: D. H. Williams, Jr. (w/o attachments)
W. G. Smith, Jr. - Bridgman (w/ attachments)
R. C. Callen (w/o attachments)
G. Charnoff (w/o attachments)
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A. B. Davis - Region III (w/ attachments)
NRC Resident Inspector - Bridgman (w/ attachments)

A. Introduction

In order to support the effort to respond to Generic Letter (GL) 88-11 for the Donald C. Cook Nuclear Plant, Southwest Research Institute (SwRI) was contracted to utilize methods presented in Revision 2 to Regulatory Guide (RG) 1.99 to (1) calculate revised reference temperatures, (2) prepare revised Pressure-Temperature curves, and (3) provide a comparison with the results of previous analyses performed under Revision 1 of the RG. The purpose of this attachment is to summarize the results of the SwRI analysis as well as our proposed actions and schedule for implementing these actions.

B. Adjusted Reference Temperature

SwRI calculated new values of Adjusted Reference Temperature (ART) by applying methodologies presented in Revision 2 of RG 1.99 to data from previously removed material surveillance specimens through Capsules Y and X, for Units 1 and 2, respectively. The results of these calculations were compared with those performed under Revision 1 of the RG. The results for the controlling materials is contained in the following table.

SUMMARY OF ART VALUES

		Initial RT_{NDT} (°F)	Revision 1 (°F) 12 EFPYs	Revision 1 (°F) 32 EFPYs	Revision 2 (°F) 12 EFPYs	Revision 2 (°F) 32 EFPYs
Unit 1						
Weld	0 T	0	293	478	267	325
Metal	1/4 T	0	234	373	236	297
	3/4 T	0	117	186	176	232
Unit 2						
Plate	0 T	58	159	198	181	211
C5521-2	1/4 T	58	146	163	165	195
	3/4 T	58	102	130	136	164

Unit 1 ART

The plot of upper shelf energy decrease versus fluence, as shown in Figure 9 of Attachment 2, indicates adequate toughness for the surveillance capsule specimen of the controlling material, weld metal, through 32 EFPYs. Additionally, the Unit 1 ART of 325°F at the inner vessel wall (0 T) for 32 effective full power years

(EFPYs) indicates that a margin of approximately 225°F will be maintained between the operating temperature (about 550°F), and the limiting temperature based on toughness. As noted in Revision 1 to the RG, a margin of 200°F is considered to permit safe management of system transients.

In addition to the SwRI analysis, we evaluated the impact of core loading patterns of a low leakage design. The Unit 1 core loading pattern was changed to a low leakage design for Cycle 8 in an effort to reduce peak flux. Subsequent core loadings are expected to continue with low leakage loading schemes. Since capsule Y was removed after Cycle 7, and the core loading design philosophy changed beginning with Cycle 8, the peak flux value obtained from the SwRI analysis of Capsule Y is not completely appropriate for projecting fluence values beyond Cycle 7. However, since the intent of the low leakage core is to reduce peak flux, the Unit 1 analysis of Capsule Y is considered conservative for this evaluation. When the next surveillance capsule (Capsule U) is removed at about 9 EFPYs for Unit 1, it will provide a means to confirm the effects that the core loading design changes are expected to have on the peak flux. The ability to meet Appendix G Fracture Toughness requirements will be re-evaluated at that time, utilizing the new fluence data. In accordance with our Technical Specification material surveillance program, Capsule U is scheduled to be removed for Unit 1 at the end of the current fuel cycle, which is anticipated to occur in the Spring of 1989.

The RG 1.99, Revision 2 values for ART were calculated based on fluence received through 32 EFPYs. Both units' operating licenses expire on March 25, 2009. A projection through expiration of the licenses, using an 80% capacity factor, 60 day refueling outages, and no forced outages indicates that Unit 1 will complete fuel Cycle 22 and accumulate 22.89 EFPYs of operation. This value was included in a submittal dated January 22, 1986 in compliance with 10 CFR 50.61 (Pressurized Thermal Shock). We have chosen to retain 32 EFPY as a basis for the design lifetime calculations in response to Generic Letter 88-11 since this number has been used in past submittals regarding 10 CFR 50, Appendix G. We believe that this approach is necessary in order to provide a meaningful comparison of the impact of the calculation methods provided by Revision 2 of RG 1.99. However, following the next refueling outage, we will be updating our Appendix G Pressure-Temperature

curves to be more indicative of the EFPYs we expect to accumulate through expiration of the plant operating license.

Unit 2 ART

ART values for Unit 2 calculated using RG 1.99, Revision 2 indicate that the reactor vessel beltline materials are projected to retain sufficient toughness to meet the current requirements of 10 CFR 50, Appendix G, throughout the design life of the unit. This is based on a calculated ART at the 1/4 T position in the vessel wall for 32 EFPYs of 195°F, which is less than the screening criteria of 200°F for new plants. Also, the plot of upper shelf energy decrease versus fluence, as shown in Figure 10 of Attachment 2, indicates adequate toughness for the plate metal through 32 EFPY. Finally, the Unit 2 ART of 211°F at the inner vessel wall for 32 EFPYs indicates that a margin of approximately 329°F will be maintained between the operating temperature and the limiting temperature based on toughness.

As was the case for Unit 1, the Revision 2 values for Unit 2 ART were calculated based on fluence received through 32 EFPYs. We conservatively expect that Unit 2 will now only complete fuel Cycle 19 and accumulate 21.38 EFPYs of operation at the time the plant operating license expires. However, 32 EFPYs was retained as the design lifetime in order to allow a meaningful comparison with the results of past calculations on this issue. We will be updating our Pressure-Temperature curves after the next surveillance capsule (Capsule U) is removed to reflect the EFPYs we expect the plant to accumulate through expiration of the plant operating license.

C. Pressure-Temperature Limits

Unit 1

Reactor Coolant System Pressure-Temperature limitations for the first 32 and 12 EFPYs, calculated using methods presented in RG 1.99, Revision 2, are shown in Figures 1, 2, 5, and 6 of Attachment 2. Although these revised curves incorporate conservatism in fluence calculations as noted in the discussion of ART results, a comparison of the 12 EFPY curve with previous results utilizing methodologies presented in Revision 1 to the RG indicates that the use of Revision 2 results in a shift of the Pressure-Temperature curves. Under the new methodology, the horizontal asymptote for the 60°F/HR heatup curve at 12 EFPYs shifted downward from about 440 psig to about 340 psig. This decrease results in a reduction in operational flexibility during unit startups in order to maintain appropriate margin between operating points and the Heatup Pressure-Temperature curve. This arises since a Reactor Coolant System pressure of 325 psig is required as a prerequisite for reactor coolant pump starts, due to controlled leakage seal differential pressure requirements. Heatup and cooldown procedures will have to address the narrower permissible band between the maximum pressure of 340 psig and the minimum pressure required to operate the reactor coolant pumps, which is 325 psig. Additionally, this may require the use of Pressure-Temperature curves based on slower heatup/cooldown rates at the lower end of the Pressure-Temperature limits.

Unit 2

Revised Reactor Coolant System Heatup and Cooldown limitations for Unit 2 are presented in Figures 3, 4, 7, and 8 of the attachment. Although reductions in the Pressure-Temperature curves will decrease the allowable operating band, this decrease will not present unacceptable difficulties in the short term. As fluence increases toward the end of the plant life, it is projected that, although these bands will narrow, operational flexibility will not be impacted.

D. Low Temperature Overpressure Protection

The Pressure-Temperature curves are inputs to the mass and heat input transient analyses which determine the Low Temperature Overpressure Protection (LTOP) pressure and temperature parameters. Therefore, a review of our present LTOP enable temperature and setpoint pressure analyses was performed to determine if they contain sufficient margin to bound the results which would be obtained for new analyses using the RG 1.99 Revision 2 Pressure-Temperature curves for 12 EFPYs. Based on this review, the mass and heat input analyses should be revised to incorporate the new Pressure-Temperature curves.

E. Planned Actions and Proposed Schedule

Unit 1

We propose the following actions and schedule to comply with the provisions of GL 88-11 for Cook Nuclear Plant, Unit 1:

1) Withdraw and analyze Capsule U utilizing RG 1.99 Revision 2 methodologies, and submit a report to the Commission within one year of Capsule withdrawal in accordance with the

provisions of 10 CFR 50, Appendix H and our Technical Specification material surveillance program. Tentative submittal date is June 30, 1990 based on Capsule withdrawal by June 30, 1989.

2) Using the analysis of Capsule U, evaluate the impact on the LTOP system. Prepare and submit Technical Specification revisions for both the LTOP system and the Appendix G Pressure-Temperature curves as needed. Scheduled submittal date is approximately 120 days after the report on Capsule U is complete, or October 31, 1990.

3) Following Commission approval, implement plant procedure and Technical Specification revisions. Scheduled completion date is 60 days after commission approval.

Unit 2

We propose the following actions and schedule to comply with the provisions of GL 88-11 for Cook Nuclear Plant, Unit 2:

1) Proceed with the formal analysis of the impact on our LTOP system using the results contained in Attachment 2, and submit a Technical Specification revision request for both the LTOP system and the Appendix G Pressure-Temperature curves as needed. Scheduled submittal date is approximately 180 days following the date of this letter, or about May 31, 1989.

2) Following Commission approval, implement plant procedure and Technical Specification revisions. Scheduled completion date is 60 days after commission approval.

ATTACHMENT NO. 2
TO AEP:NRC:0894K