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SUBJECT: Forwards assessment of ref temp pressurized thermal shock per requirements of 10CFR50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events."

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INDIANA & MICHIGAN ELECTRIC COMPANY

P.O. BOX 16631
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January 22, 1986
AEP:NRC:0561A

Donald C. Cook Nuclear Plant Unit Nos. 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74
10 CFR 50.61 - PRESSURIZED THERMAL SHOCK RULE

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Denton:

This submittal is made pursuant to the requirements of 10 CFR Part 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events." The attachment to this letter contains the projected Reference Temperature (RTPTS) for the reactor vessel materials which was calculated by the method noted in 10 CFR 50, Section 50.61(b)(2) and the bases for these projections.

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

Very truly yours,

M. P. Alexich
Vice President

1/22/86

cm

Attachment

cc: John E. Dolan
W. G. Smith, Jr. - Bridgman
R. C. Callen
G. Bruchmann
G. Charnoff
NRC Resident Inspector - Bridgman

Add:

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ATTACHMENT TO AEP:NRC:0561A

REFERENCE TEMPERATURE, PRESSURIZED THERMAL SHOCK (RTPTS) ASSESSMENT FOR D. C. COOK UNIT NOS. 1 AND 2

BASES FOR RTPTS PROJECTIONS

The RTPTS calculations for both units were performed in accordance with the guidance provided in 10 CFR 50.61, as published in the Federal Register, Vol. 50, No. 141, dated July 23, 1985.

MATERIAL ANALYSES

The copper and nickel content values used in the calculations were obtained from data submitted to AEP by the Westinghouse Electric Corporation. Copies of the data have previously been submitted to the NRC by the following correspondence:

Unit 1

Indiana & Michigan Electric Co. letter dated November 7, 1977, Docket No. 50-315, DPR No. 58

Letter, AEP:NRC:0894C, dated July 3, 1985

Unit 2

Unit 2 FSAR, Appendix Q, answer to question 121.2, Amendment 1977, July, 1977

INITIAL REFERENCE TEMPERATURE

The values of Initial Reference Temperature for the plate materials were obtained from the results of actual fracture toughness tests performed on the unirradiated plate materials. As such, these values were considered to be "measured" values.

No fracture toughness tests were performed on the unirradiated weld material, therefore the "generic mean" values given in 10 CFR 50.61 were used.

FLUENCE VALUES

The fluence received to date was based upon the time of effective full-power operation up through December 1, 1985. For Unit 1 that is 6.81 Effective Full-Power Years (EFPYs) and for Unit 2 that is 5.09 EFPYs.

The fluence received up through the expiration of the operating license was based upon the projected maximum time of effective full-power operation that can be achieved up through the last complete fuel cycle prior to the expiration date of the operating license. Both units' operating licenses

expire on March 25, 2009 (i.e., 40 years from the date of receipt of the Construction Permit). The projection of full-power operation beyond December 1, 1985 was based on an 80% capacity factor, 60-day refueling outages, and no forced outages. Unit 1 is projected to complete fuel Cycle 22 and accumulate 22.89 EFPYs of operation, whereas Unit 2 is projected to complete fuel Cycle 19 and accumulate 21.38 EFPYs of operation.

Unit 1

The peak flux value was obtained from the Southwest Research Institute (SwRI) Final Report on the "Unit No. 1 Analysis of Capsule Y." Capsule Y is the most recent surveillance capsule removed as part of the Unit 1 reactor vessel material surveillance program. The peak flux incident on the inside surface of the reactor vessel shell was determined by SwRI through analyses of the capsule Y dosimetery. Capsule Y was removed between core Cycles 6 and 7.

The Unit 1 core loading pattern was changed to a low-leakage design for Cycles 8 and 9. Subsequent core loadings are expected to continue with the low-leakage loading patterns. Since capsule Y was removed after Cycle 6 and the core loading design philosophy changed after Cycle 7, the peak flux value obtained from the SwRI analysis of capsule Y is not completely appropriate for projecting fluence values beyond Cycle 7. However, as additional surveillance capsules are removed from Unit 1, they will provide a means to analyze the effects that the core loading design changes have had on the peak flux. The corresponding effect on the RTPTS values will also be evaluated as those capsules are removed. The next capsule removal from Unit 1 is scheduled to occur when the plant accumulates approximately 9.0 EFPYs of operation.

A change in core loading patterns similar to the one made in Unit 1 was instituted in Unit 2, and analyses of the surveillance capsule removed subsequent to the design change revealed a decrease in peak flux with the low-leakage core loading pattern. Since the changes in the core loading patterns that occurred at both units were somewhat similar, the peak flux value for Unit 1 is expected to decrease for the cycles after core Cycle 7. In light of this, the peak flux value obtained from the Unit 1 analysis of capsule Y was considered conservative for cycles beyond Cycle 7, and was therefore used in projecting the fluence values.

Unit 2

Capsule Y was removed from Unit 2 after core Cycle 3 and is the latest surveillance capsule to be removed as part of the Unit 2 reactor vessel material surveillance program. Since low-leakage loading patterns were instituted at Unit 2 after Cycle 1, the SwRI analysis of capsule Y had to consider these changes. The Cycle 2 core loading pattern did not differ significantly from Cycle 1 with respect to the power in the peripheral assemblies. Since the peripheral assemblies have the greatest effect on the neutron leakage to the vessel shell,

core Cycle 2 was considered to be the same as Cycle 1 with respect to peak flux to the vessel shell. The Cycle 3 loading pattern differed from Cycles 1 and 2, resulting in lowered power in corner peripheral assemblies. This reduction in power in the corner assemblies is projected to have an effect on the flux to the vessel shell. The SwRI flux density calculation took this into consideration, and as such two values of peak flux were determined; one for core Cycles 1 and 2, and an approximately 20% lower value for core Cycle 3.

The fluence values used in the RTPTS calculations for Unit 2 were based upon the original core loading pattern flux value for the 2.00 EFPYs of operation during Cycles 1 and 2 and the lower Cycle 3 low-leakage pattern flux value for all subsequent cycles.

As additional surveillance capsules are removed from Unit 2, they will provide a means to analyze the effects that the post Cycle 3 core loading design changes have had on the peak flux. The corresponding effect on the RTPTS values will also be evaluated at that time. The next capsule removal from Unit 2 is scheduled to occur after the present fuel cycle (Cycle 5) is completed. This should occur in the early part of 1986, and the capsule removed at that time will have been exposed to approximately 5.3 EFPYs of plant operation.

Core loading patterns subsequent to Cycle 3 have evolved into an even more effective low-leakage loading scheme, which we believe will further reduce the peak flux to the reactor vessel shell. The peak flux value obtained from the analysis of capsule Y for Cycle 3 is therefore considered conservative for projecting fluence values beyond Cycle 3.

Based on the above core loading assumptions and the calculated RTPTS values noted in the attached table, it is our conclusion that reactor vessel materials for the Donald C. Cook Plant will not exceed the RTPTS screening criterion of 10 CFR 50.61 prior to expiration of the operating licenses. However, the information submitted herein will be updated whenever changes in core loadings surveillance measurements or other information indicate a significant change in the projected values.

The SwRI reports for both units were submitted to the Commission by letter AEP:NRC:0894A, dated July 20, 1984.

RT_{PTS} Projections

D. C. Cook Unit No. 1

Reactor Vessel Beltline Material*	RT_{PTS} To Date 6.81 EFPYs	RT_{PTS} @ Expiration of Operating License 22.89 EFPYs
Plate B4406-1	127°F	149°F
Plate B4406-2	164°F	192°F
Plate B4406-3	164°F	192°F
Plate B4407-1	121°F	148°F
Plate B4407-2	97°F	120°F
Plate B4407-3	127°F	153°F
All Axial Welds	82°F	113°F
Circumferential Weld 9-442	181°F	251°F

D. C. Cook Unit No. 2

Reactor Vessel Beltline Material*	RT_{PTS}, To Date 5.09 EFPYs	RT_{PTS} @ Expiration of Operating License 21.38 EFPYs
Plate C5521-2	146°F	172°F
Plate C5556-2	170°F	199°F
Plate C5540-2	75°F	96°F
Plate C5592-1	88°F	115°F
All Beltline Region Welds	30°F	43°F

* see attached sketch depicting the arrangements of the reactor vessel beltline region materials.

D.C. COOK PLANT

Arrangement of Reactor Vessel Beltline Region Materials