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SUBJECT: Application for amend to License DPR-74,revising Tech Specs
re reactor vessel matl surveillance capsule reanalysis.

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AEP:NRC:0894L

Donald C. Cook Nuclear Plant Unit 2
Docket No. 50-316
License No. DPR-74
TECHNICAL SPECIFICATION CHANGE REQUEST
HEATUP AND COOLDOWN CURVES

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

Attn: T. E. Murley

October 25, 1989

Dear Dr. Murley:

RE: Our Letter AEP:NRC:0894K dated December 5, 1988

This letter and its attachments constitute an application for amendment to the Technical Specifications for Donald C. Cook Nuclear Plant Unit No. 2. Specifically, we request that the plant heatup and cooldown curves be changed to reflect the recent reactor vessel material surveillance capsule reanalyses performed by Westinghouse Electric Corporation based on Regulatory Guide (R/G) 1.99, Rev. 2. The reasons for the proposed change and our analysis concerning significant hazards considerations are contained in Attachment 1 to this letter. The proposed revised Technical Specification pages are contained in Attachment 2. The results of the capsule reanalyses performed by Westinghouse are contained in Attachment 3.

We believe that the proposed changes will not result in (1) a significant change in the types of effluents or a significant increase in the amounts of any effluent that may be released offsite, and (2) a significant increase in individual or cumulative occupational radiation exposure.

The revised heatup and cooldown curves in Attachment No. 2 have been evaluated against the criterion of 10 CFR 50, Appendix G, Paragraph IV.A.2. The proposed revised curves reflect the Appendix G criterion. The impact of this T/S change request on our cold overpressurization limits has been evaluated. This evaluation indicates that the existing limits remain adequate.

A001

In order to implement the revised heatup and cooldown curves in an orderly fashion, we are proposing to institute these new limits during the refueling window (Mode 6) of the upcoming Cycle 7-8 refueling outage, prior to the unit returning to Mode 5. We presently anticipate the outage to occur as early as July 1, 1990. Therefore, we request that your review of these changes be completed by June 1, 1990.

These proposed Technical Specifications have been reviewed by the Plant Nuclear Safety Review Committee and the Nuclear Safety and Design Review Committee.

In compliance with the requirements of 10 CFR 50.91(b)(1), copies of this letter and its attachments have been transmitted to Mr. R. C. Callen of the Michigan Public Service Commission and to the Michigan Department of Public Health.

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

eh

Attachments

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

ATTACHMENT NO. 1

TO

AEP:NRC:0894L

This letter and its attachments constitute an application for amendment to the Technical Specifications (T/Ss) for Donald C. Cook Nuclear Plant Unit 2. Submittal of these changes was identified as a follow-up action for Unit 2 in our response to Generic Letter (GL) 88-11 (AEP:NRC:0894K), dated December 5, 1988 for implementing Regulatory Guide (R/G) 1.99, Revision 2. Specifically, that letter committed to submitting revised 10 CFR 50 Appendix G heatup/cooldown pressure-temperature (P-T) limit curves and, if necessary, revised setpoints for the low temperature overpressure protection (LTOP) system. Therefore, we are submitting revised P-T curves (and associated T/S pages) for Unit 2 which have been developed using the methodology of Revision 2 to R/G 1.99 for predicting the effect of neutron radiation on reactor vessel materials. As stated in letter AEP:NRC:0894K, revised P-T curves for Unit 1 will be submitted following analysis of capsule U, presently scheduled for completion in April of 1990.

As stated in the response to GL 88-11, initial evaluation of the impact of the revised R/G was based on an analysis by Southwest Research Institute (SwRI), in which the previous controlling materials for both Unit 1 and Unit 2 were reviewed. Since that time, Westinghouse Electric Corporation was requested to re-evaluate the impact of the R/G for all Unit 2 reactor vessel materials, and to develop heatup/cooldown curves for the limiting material. As a result, a new controlling material was identified. The Westinghouse report that identifies the new controlling material and provides the new P-T limit curves is contained in Attachment 3.

Separately, Westinghouse evaluated the impact of the new P-T curves on the LTOP system setpoints. This evaluation determined that the current setpoints provide adequate protection against exceeding the new P-T limits during anticipated overpressure transients while operating at low temperatures.

Implementation of the noted T/S changes completes the proposed actions identified in letter AEP:NRC:0894K, for Donald C. Cook Nuclear Plant Unit 2. Future analyses which require an estimate of the embrittlement of reactor vessel beltline materials will be performed using the methods presented in Revision 2 to R/G 1.99.

Technical Specification Revisions

The T/S changes found in Attachment 2 are based on implementation of R/G 1.99, Revision 2. The following T/S pages are impacted by this change:

Unit No. 2:	Page 3/4 4-24	Page B 3/4 4-6
	Page 3/4 4-25	Page B 3/4 4-7
	Page 3/4 4-26	Page B 3/4 4-8
		Page B 3/4 4-9
		Page B 3/4 4-9a
		Page B 3/4 4-10

Page 3/4 4-24

The maximum heat-up rate was changed from 100°F/hour to 60°F/hour since the reduced rate is more indicative of achievable rates, and provides an expanded window for reactor coolant pump operation. This revised maximum heat-up rate is also reflected in Figure 3.4-2 (Page 3/4 4-25).

Pages 3/4 4-25, 26

Figures 3.4-2 and 3.4-3 are the heatup and cooldown P-T limit curves for Unit 2. These curves are composite curves reflecting the adjusted reference temperature of the most limiting material at the end of 12 effective full power years (EFPY).

These P-T curves were developed based on the following considerations:

1. Intermediate shell plate C5556-2 is the new limiting material with a copper and nickel content of 0.15% and 0.57%, respectively.

2. The maximum heat up rate was changed from 100°F/hour to 60°F/hour, as noted above for page 3/4 4-24.

3. The projected fluence is as noted in Table 5.1 of the Southwest Research Institute report, "Reactor Vessel Material Surveillance Program for Donald C. Cook Unit No. 2: Analysis of Capsule X," dated May 1987. This report was submitted as an attachment to letter AEP:NRC:0894I, dated June 26, 1987.

4. Radiation induced changes in the reference nil ductility temperature are predicted based on the methodology presented in U.S. Nuclear Regulatory Commission R/G 1.99, Revision 2.

The P-T curves reflect a new limiting material, intermediate shell plate C5556-2. As noted in prior capsule analyses, the intermediate shell plates were projected to control the adjusted value of RT_{NDT} through the design life of the pressure vessel. Intermediate shell plates C5521-2 (RT_{NDT} = 38°F, %P = 0.013, %Cu = 0.14, %Ni = 0.58) and C5556-2 (RT_{NDT} = 58°F, %P = 0.014, and %Cu = 0.15, %Ni = 0.57) are equivalent materials in accordance with ASTM E185-73, Annex A1. However, Westinghouse selected only plate C5521-2 for the original surveillance program because it had the lower Charpy V-notch upper shelf energy as directed by Figure A1 of ASTM E 185-73. Previous vessel RT_{NDT} predictions for Unit 2 surveillance material (plate C5521-2) were based on an unirradiated RT_{NDT} of 58°F for C5556-2, because it was more conservative to do so.

Under the provisions of the revised R/G, plate C5556-2 now becomes the controlling material. This results from the combination of the higher RT_{NDT} (58°F for C5556-2 versus 38°F for C5521-2) and the increased margin that must be included for plate material C5556-2 through application of Position 1 of the R/G (surveillance data not available). The adjusted reference temperature at 1/4T for this material is projected to be 178°F and 207°F for 12 and 32 EFPY, respectively. This compares to reported values of 165°F and 195°F in letter AEP:NRC:0894K. While the adjusted 1/4T reference temperature of 207°F exceeds the screening criteria of 200°F for new plants, this value is based on fluence received through the design lifetime (32 EFPY). We expect that the actual EFPY accumulated through the end of operating license will be less than the 32 EFPY design lifetime. We will be updating our P-T curves after the next surveillance capsule (Capsule U) is removed to reflect cumulative fluence for the EFPYs we expect the plant to accumulate through expiration of the operating license.

<u>Page B3/4 4-6 through B 3/4 4-10</u>

The Bases for the P-T limits were modified to reflect the use of Revision 2 to R/G 1.99 for predicting the effect of radiation on the embrittlement of reactor vessel materials. As noted for Pages 3/4 4-24 and 3/4 4-25, the maximum heatup rate has been revised from 100°F to 60°F to reflect a more realistic heatup rate and to provide an expanded window for reactor coolant pump operation.

Figure B 3/4/4-1, "Fast Neutron Fluence (E > 1MeV) As a Function of Full Power Service Life," was deleted since fluence is measured and projected based on surveillance capsule dosimetry.

Figure B 3/4.4-2 "Effect of Fluence and Copper Content on Shift of RT_{NDT} for Reactor Vessels Exposed to 550°F Temperature," was deleted since the methodology of R/G 1.99, Revision 2 is used to predict the shift of RT_{NDT}.

Table B 3/4.4-1 was revised to include nickel content vessel materials and to delete the phosphorus content since the revised R/G is sensitive to nickel (and copper) content, rather than phosphorus. Additionally, supplemental data identified while researching the nickel content of the vessel materials has been included for some materials. This table was also condensed to a single page.

Per 10 CFR 50.92, a proposed amendment will not involve significant hazards consideration if the proposed amendment does not:

(1) involve a significant increase in the probability or consequences of a previously evaluated accident,

(2) create the possibility of a new or different kind of accident from any previously analyzed or evaluated, or

(3) involve a significant reduction in a margin of safety.

<u>Criterion 1</u>

The changes to the P-T curves are in the conservative direction. The new curves were generated based on the latest NRC guidance, Rev. 2 to R/G 1.99.. Therefore, we conclude that the changes will not involve a significant increase in the probability or consequences of a previously evaluated accident, nor will the changes involve a significant reduction in a margin of safety.

<u>Criterion 2</u>

The changes do not involve any physical modifications to the plant. The changes will involve changes to plant operations; however, these changes are in the conservative direction, placing more stringent requirements on heatup/cooldown operations. Therefore, the changes should not create the possibility of a new or different kind of accident from any previously analyzed or evaluated.

Therefore, the change should not create the possibility of a new or different kind of accident from any previously analyzed or evaluated.

Criterion 3

See Criterion 1 above.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing examples (48 FR 14879) of amendments considered not likely to involve significant hazards consideration. The second of these examples refers to changes that constitute additional limitations, restrictions, or controls not presently included in the T/Ss. As discussed above, the changes impose more stringent requirements on heatup/cooldown operations. For this reason, we believe the example cited is relevant and conclude that the changes should not require significant hazards consideration.

ATTACHMENT NO. 2

TO

AEP:NRC:0894L