REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:8005130234 DOC.DATE: 80/05/09 NOTARIZED: NO DOCKET #
FACIL:50-315 Donald C. Cook Nuclear Power Plant, Unit 1, Indiana & 05000315
50-316 Donald C. Cook Nuclear Power Plant, Unit 2, Indiana & 05000316

AUTH.NAME AUTHOR AFFILIATION
DOLAN,J.E. Indiana & Michigan Electric Co.
RECIP.NAME RECIPIENT AFFILIATION
DENTON,H.R. Office of Nuclear Reactor Regulation

SUBJECT: Submits Tech Specs Change 1 re extension to surveillance test interval re ice condenser lower inlet door testing & Change 2 re reactor surveillance capsule withdrawal schedule.Forwards proposed changes to Tech Specs & fee.

DISTRIBUTION CODE: A001S COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 6+4
TITLE: General Distribution for after Issuance of Operating Lic

NOTES: W/CHECK $2,800.00, I&E - 3 CYS ALL MATL.

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
ACTION:	05 BC ORB #1	7	7			
INTERNAL:	01 REG FILE	1	1	02 NRC PDR	1	1
	12 I&E	2	2	15 CORE PERF BR	1	1
	17 ENGR BR	1	1	18 REAC SFTY BR	1	1
	19 PLANT SYS BR	1	1	20 EEB	1	1
	21 EFLT TRT SYS	1	1	EPB-DOR	1	1
	OELD	1	0	STS GROUP LEADR	1	1
EXTERNAL:	03 LPDR	1	1	04 NSIC	1	1
	23 ACRS	16	16			

MAY 14 1980

APP 2

60

TOTAL NUMBER OF COPIES REQUIRED: LTTR 38 ENCL 37

8005130234
P

INDIANA & MICHIGAN ELECTRIC COMPANY

P. O. BOX 18
BOWLING GREEN STATION
NEW YORK, N. Y. 10004

May 9, 1980
AEP:NRC:00121

Donald C. Cook Nuclear Plant Unit Nos. 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74

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

Dear Mr. Denton:

This letter requests changes to the Appendix 'A' Technical Specifications for the Donald C. Cook Nuclear Plant Unit Nos. 1 and 2. Attachment 'A' to this letter contains the description and review of each change. Attachment 'B' contains the appropriate revised pages.

The proposed Technical Specification changes contained herein have been reviewed and approved by the Plant Nuclear Safety Review Committee and will be reviewed shortly by the AEPSC Nuclear Safety and Design Review Committee. The PNSRC review indicates that in no instance will the proposed Technical Specification change adversely affect the health and safety of the public.

Change No. 1 is considered to be a Class II Amendment as per the provisions of 10 CFR 170.22. Change No. 2 is considered to be a Class II Amendment and a Class I Amendment. Accordingly, attached is a check in the amount of $2,800.00.

Your prompt attention to this matter is requested.

Very truly yours,

John E. Dolan
John E. Dolan
Vice President

A001
s
1/1
w/check:
$2800.00

JED/emc
cc: (Attached)

cc: R. C. Callen
G. Charnoff
R. S. Hunter
R. W. Jurgensen
D. V. Shaller - Bridgman

ATTACHMENT 'A' TO AEP:NRC:00121

CHANGE NO. 1 - SPECIFICATION 4.6.5.3.1

This change serves to request a 'one-time' extension to the surveillance test interval of Unit No.2 Technical Specification No. 4.6.5.3.1 regarding Ice Condenser Lower Inlet Door Testing.

The present surveillance interval, including the 'Grace Period' allowed by Specification No. 4.0.2, will end on June 8, 1980. As previously discussed with the NRC Staff, I&MECo. intends to bring Unit No. 2 off line during the forthcoming Unit No. 1 refueling outage to complete modifications to the Auxiliary Feedwater System. This Unit No. 2 outage is tentatively scheduled to begin in late June or early July, 1980. It is requested that an extension to the surveillance test interval of Specification 4.6.5.3.1 be granted until July 20, 1980. This extension will avoid an unnecessary unit shutdown with the corresponding challenge to safety systems and fuel thermal stresses. The required lower inlet door surveillance would be completed prior to startup from the aforementioned Unit No. 2 outage.

Extension of the lower inlet door surveillance test interval will have no adverse effect on the ability of the Cook Plant to safely mitigate the consequences of a hypothetical accident. No significant icing has been experienced with the Unit No. 2 lower inlet doors and there is no reason to expect any icing during the extended surveillance interval.

Previous surveillance testing of the lower inlet doors has clearly demonstrated that the doors would indeed open as designed during a hypothetical accident. A summary of the previous tests is shown in Table 1 below.

TABLE 1

UNIT NO. 2 LOWER INLET DOOR SURVEILLANCE HISTORY

TEST DATE	COMMENTS
March 1978	All doors found acceptable
May 1978	All doors found acceptable
July 1978	All doors found acceptable
November 1978	One door out of forty eight found unacceptable (*)
May 1979	All doors found acceptable
October 1979	All doors found acceptable

* The right door in Bay 9 was found to require a greater opening force than allowed by the Technical Specifications. All succeeding surveillance tests on this door have been successful.

CHANGE NO. 2 - TABLE 4.4-5 (UNIT NOS. 1 AND 2)

Technical Specification Tables 4.4-5 of Units 1 and 2 have been revised to reflect recent changes in the reactor vessel surveillance capsule withdrawal schedule which will allow for the collection of data from high lead factor capsules in a more meaningful time frame of reactor service life. The proposed revised withdrawal schedule has been recommended to us by Westinghouse Electric Corporation and by Southwest Research Institute, our inservice inspection consultant. The recalculated lead factors as supplied to us by Westinghouse are higher than the previously reported factors also generated by Westinghouse. The changes marked on pages B 3/4 4-11 (Unit 1) and B 3/4 4-10 (Unit 2) correct a previous typographical error.

ATTACHMENT 'B' TO AEP:NRC:00121

TABLE 4.4-5

REACTOR VESSEL MATERIAL IRRADIATION SURVEILLANCE SCHEDULE

SPECIMEN	REMOVAL INTERVAL
1. Capsule T	1.25 EFPY
2. Capsule X	3 EFPY
3. Capsule Y	5 EFPY
4. Capsule U	9 EFPY
5. Capsule S	32 EFPY
6. Capsules V, W, Z	Standby

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E>1 Mev) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence and copper content of the material in question, can be predicted using Figures B 3/4.4-1 and B 3/4.4-2. The heatup and cooldown limit curves, Figures 3.4-2 and 3.4-3, include predicted adjustments for this shift in RT_{NDT} at the end of 12 EFPY, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-5 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on pressurizer heatup and cooldown and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

3/4.4.10 STRUCTURAL INTEGRITY

The required inspection programs for the Reactor Coolant System components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant. To the extent applicable, the inspection program for the Reactor Coolant System components is in compliance with Section XI of

TABLE 4.4-5

REACTOR VESSEL MATERIAL IRRADIATION SURVEILLANCE SCHEDULE

	SPECIMEN	REMOVAL INTERVAL
1.	CAPSULE T	1 EFPY
2.	CAPSULE Y	3 EFPY
3.	CAPSULE X	5 EFPY
4.	CAPSULE U	9 EFPY
5.	CAPSULE S	32 EFPY
6.	CAPSULES V, W, Z	STANDBY

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-5 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on pressurizer heatup and cooldown and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

3/4.4.10 STRUCTURAL INTEGRITY

The inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant. To the extent applicable, the inspection program for these components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code.