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SUBJECT: Provides results of offsite dose calculation for reactor
 coolant pump locked rotor event for facility Cycle 8.

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AEP:NRC:1071K

Donald C. Cook Nuclear Plant Unit 2
docket No. 50-316
License No. DPR-74
OFFSITE DOSE CALCULATION FOR THE REACTOR COOLANT
PUMP LOCKED ROTOR EVENT FOR UNIT 2 CYCLE 8

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

Attn: T. E. Murley.

July 23, 1990

Dear Dr. Murley:

The purpose of this letter is to transmit the results of the offsite dose calculation for the reactor coolant pump locked rotor event for Unit 2 Cycle 8.

The license basis analysis of locked rotor event included in Appendix B, Section B.3.5.2 of Attachment 4 of our submittal AEP:NRC:1071E does not assume fuel failure based on DNB. However, Condition 9 of the SER for WCAP 10444-P-A, "Reference Core Report, Vantage 5 Fuel Assembly," states "With regard to the RCS pump shaft seizure accident, the fuel failure criterion should be the 95/95 DNBR limit. The mechanistic method mentioned in WCAP-20444 is not acceptable." Such an analysis showing 11% rods in failure was submitted as Attachment 3 to our supplementary submittal AEP:NRC:1071I. Our fuel contractor, Westinghouse Electric Corporation has also calculated the expected offsite dose resulting from the failure of 11% of the fuel rods. At the request of the staff, we are submitting a discussion of this calculation as an attachment to this letter. A preliminary copy of this discussion was telecopied to Timothy Colburn, Cook Nuclear Plant Project Manager, on July 18, 1990.

In a discussion with the staff on July 16, 1990, concern was expressed regarding our references to the "Short-term Containment Analysis" and the "LOCA Containment Integrity" on page 34 of Attachment 4 to our submittal AEP:NRC:1071E. For the convenience of the staff, we identified previously issued SERs addressing these analyses. The first of these was attached to a letter dated

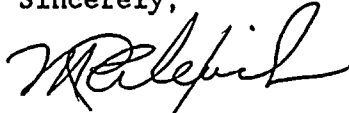
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January 30, 1989 from John F. Stang of the NRC staff to Mr. Milton P. Alexich, Vice President of Indiana Michigan Power Company, regarding a "Change to Licensing Basis for SI and RHR Crossties." This SER approves the LOCA containment integrity analysis for both units. The second SER was attached to Amendment No. 126 to Facility Operating License No. DPR-58, Cook Nuclear Plant Unit 1. The staff concluded "that the D. C. Cook Nuclear Plant's design basis pertaining to containment short term response... is adequate for RTP operation and therefore is acceptable." The SER specifies Unit 1 in its discussion. However, as indicated in WCAP 11902, (Attachment 2 to our submittal AEP:NRC:1067), on pg 3.4-2, "The analysis addressed the Cook Nuclear Plants [sic] Units 1 and 2, at rerated conditions assuming an NSSS power level of 3600 MWt, for a range of conditions which bound [the reduced temperature and pressure conditions]" The analysis also bounds the operating conditions proposed for Unit 2 Cycle 8. The assumptions also bound the conditions documented in Attachment 4 to AEP:NRC:1071E for Unit 2. We also note that as far as they impact this analysis, the Unit 1 and Unit 2 containments are identical. One of these SERs and pertinent portions of the other were telecopied to the staff on July 18, 1990 for the convenience of the reviewer.

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

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

ATTACHMENT TO AEP:NRC:1071K
LOCKED ROTOR DOSE ANALYSIS FOR
DONALD C. COOK NUCLEAR PLANT
UNIT 2 CYCLES 8 AND 9

Reactor Coolant Pump Locked Rotor Offsite Radiation Dose Analysis

A conservative analysis of the potential radiological consequences of a reactor coolant pump locked rotor event has been performed.

The salient assumptions used to calculate the activity releases and offsite doses follow.

1. Prior to the accident, the primary coolant iodine concentration is assumed to equal the Technical Specification transient limit for full power operation - 60 uCi/gram of dose equivalent I-131 (Technical Specification Figure 3.4-1)
2. Prior to the accident, the secondary coolant iodine concentration is assumed to equal the Technical Specification limit (LCO 3.7.1.4) for full power operation - 0.1 uCi/gram of dose equivalent I-131
3. Eleven percent of the core is predicted to fail as a result of DNB. This results in the release of 11% of the core gap activity to the primary coolant. The fraction of core activity contained in the gap (gap fraction) is assumed to be 10% for all nuclides. Thus, a total of 1.1% of the core activity is released.
4. The total primary-to-secondary leak rate is assumed to be at the Technical Specification limit (LCO 3.4.6.2 item c) of 1 gpm for the duration of the event.
5. An iodine partition coefficient of 100 between the steam generator liquid and steam phases is assumed. This value is suggested in Standard Review Plan (NUREG-0800) Section 15.6.3, Revision 2. It is a conservative estimate of the partition coefficient due to secondary coolant boil-off.
6. Offsite power is lost.
7. Steam release to the environment: 0 to 2 hr - 600,000 lbm
2 to 8 hr - 1,200,000 lbm
8. Eight hours after the accident the Residual Heat Removal System is assumed to start. No additional steam or radioactivity is released to the environment after 8 hrs.

Reactor core and coolant fission product inventories, dose conversion factors, gamma energies, atmospheric dispersion factors, and breathing rates are consistent with the values used in WCAP-12135.

Results

	Dose in rem	
	Site boundary	Low population zone
Thyroid	3.0	6.0
Whole-body gamma	0.3	0.2

Conclusion

The dose acceptance criteria is based on the recommendations of Standard Review Plan Section 15.5.3, i.e., 30 rem thyroid and 2.5 rem whole-body. The calculated doses for the locked rotor event are within the acceptance criteria.