November 25, '94

Mr. E. E. Fitzpatrick, Vice President
Indiana Michigan Power Company
c/o American Electric Power Service Corporation
1 Riverside Plaza
Columbus, OH 43215

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNIT NO. 2 - ISSUANCE OF AMENDMENT RE: UPDATED HEATUP AND COOLDOWN CURVES (TAC NO. M88889)

Dear Mr. Fitzpatrick:

The Commission has issued the enclosed Amendment No. 171 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Unit No. 2. The amendment consists of changes to the Technical Specifications (TS) in response to your application dated February 22, 1994.

The amendment revises the heatup and cooldown curves in TS section 3/4.4.9.1. This revision is based on a recent analysis of a reactor vessel material surveillance capsule performed by Westinghouse Electric Corporation.

A copy of our related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by

John B. Hickman, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-316

Enclosures: 1. Amendment No. 171 to DPR-74
2. Safety Evaluation

cc w/encls: See next page

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DATE	10/5/94	10/7/94	11/4/94	10/14/94

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Mr. E. E. Fitzpatrick
Indiana Michigan Power Company

Donald C. Cook Nuclear Plant

cc:

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DATED: November 25, 1994

AMENDMENT NO. 171 TO FACILITY OPERATING LICENSE NO. DRP-74-D. C. COOK, UNIT 2

Docket File
PUBLIC
PDIII-1 Reading
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cc: Plant Service list

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 171
License No. DPR-74

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated February 22, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-74 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 171 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Kim A. Connaughton for

John N. Hannon, Director
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: November 25, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 171

FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

REMOVE	INSERT
3/4 4-25	3/4 4-25
3/4 4-26	3/4 4-26
B 3/4 4-6	B 3/4 4-6
B 3/4 4-10	B 3/4 4-10

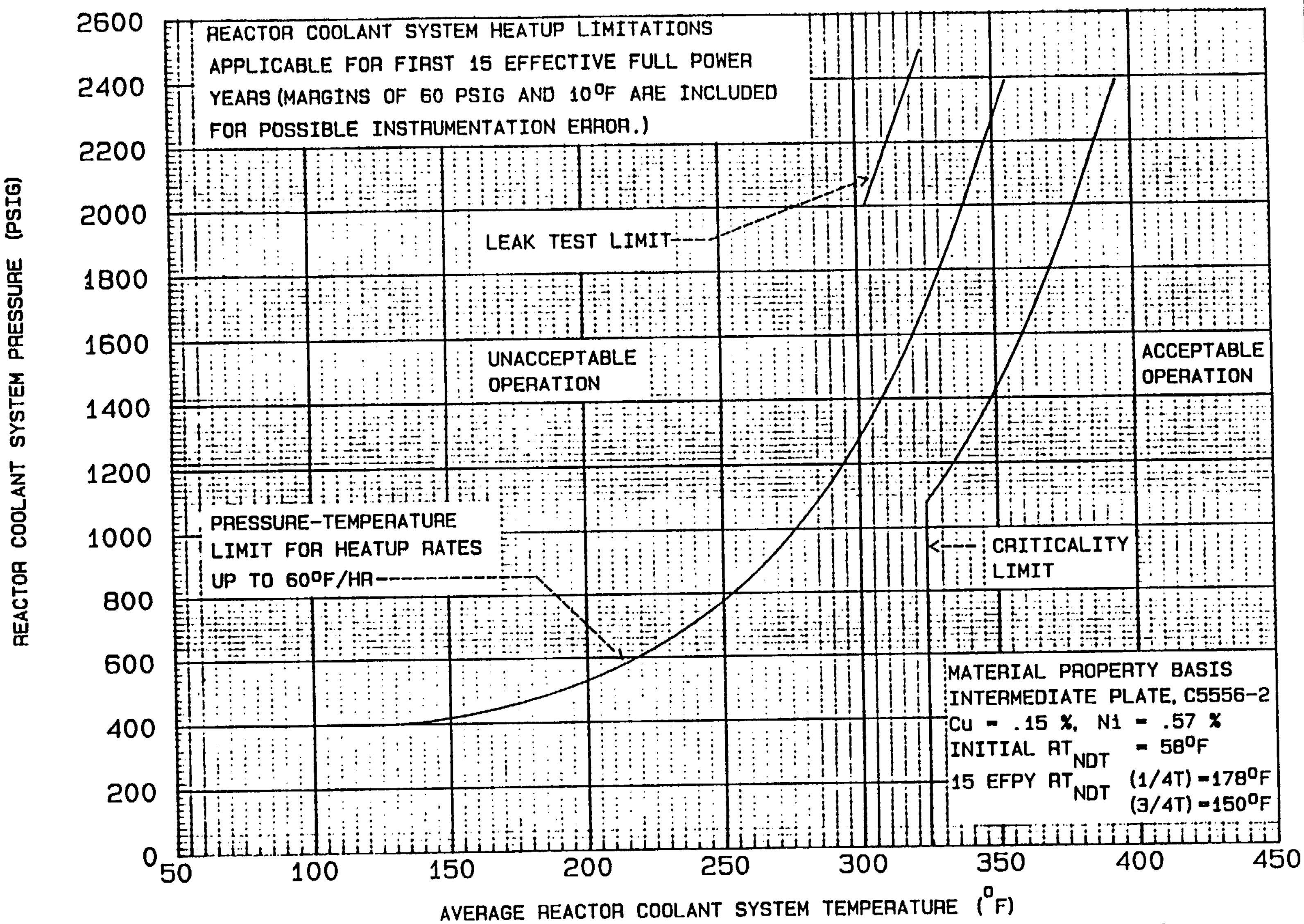

FIGURE 3.4-2

REACTOR COOLANT SYSTEM, PRESSURE - TEMPERATURE, LIMITS FOR 60 °F/HR RATE, CRITICALITY LIMIT AND HYDROSTATIC TEST LIMIT

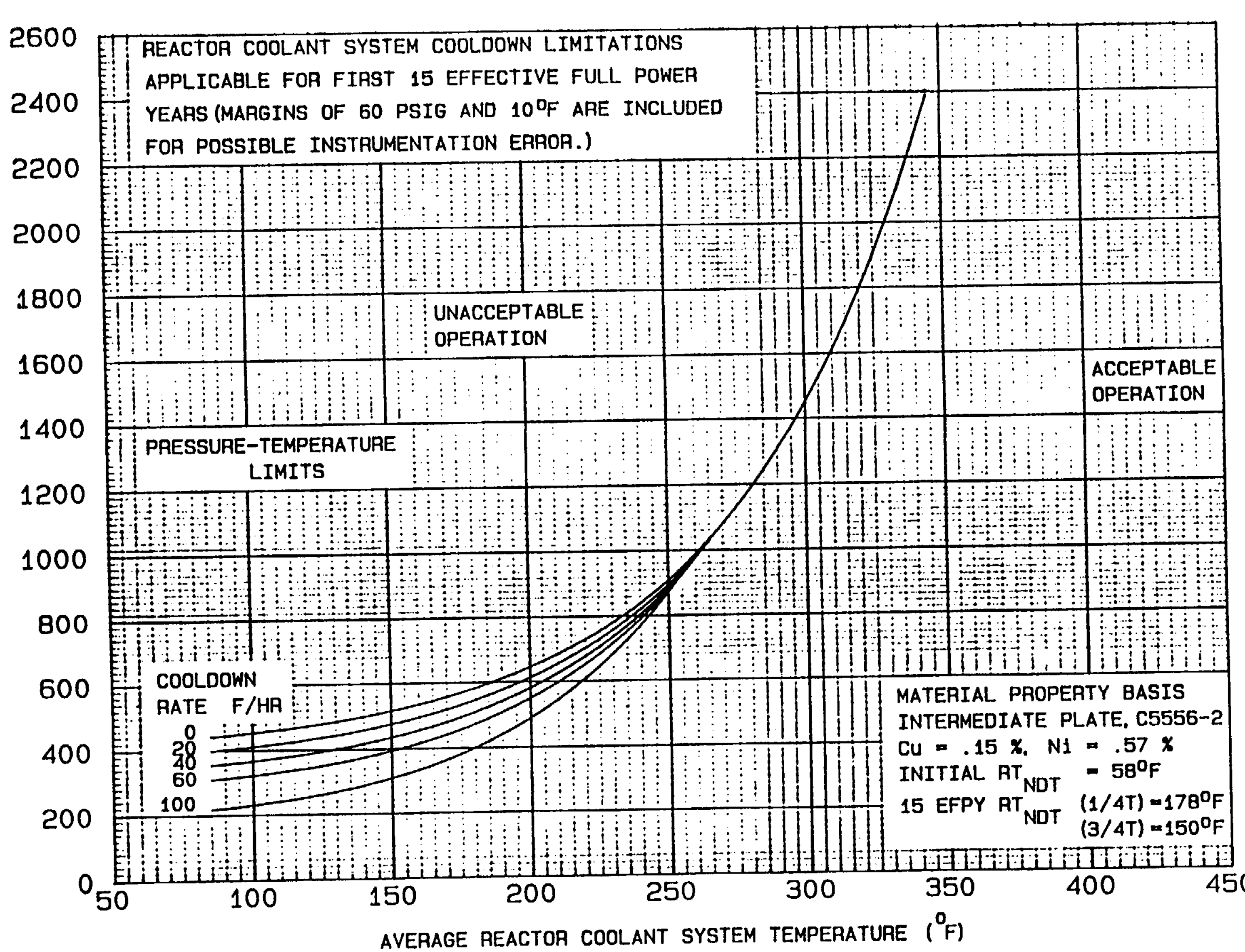

FIGURE 3.4-3

REACTOR COOLANT SYSTEM, PRESSURE – TEMPERATURE, LIMITS FOR VARIOUS COOLDOWN RATES

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 4.1.4 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

An ID or OD one-quarter thickness surface flaw is postulated at the location in the vessel which is found to be the limiting case. There are several factors which influence the postulated location. The thermal induced bending stress during heatup is compressive on the inner surface while tensile on the outer surface of the vessel wall. During cooldown, the bending stress profile is reversed. In addition, the material toughness is dependent upon irradiation and temperature and therefore, the fluence profile through the reactor vessel wall, the rate of heatup and also the rate of cooldown influence the postulated flaw location.

The heatup limit curve, Figure 3.4.2, is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 60°F per hour. The cooldown limit curves of Figure 3.4-3 are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The heatup and cooldown curves were prepared based on the most limiting value of the predicted adjusted reference temperature at the end of 15 EFPY.

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 must be predicted in accordance with Regulatory Guide 1.99, Revision 2. This prediction is based on the fluence and a chemistry factor determined from one of two Positions presented in the Regulatory Guide. Position (1) determines the chemistry factor from the copper and nickel content of the material. Position (2) utilizes surveillance data sets which relate the shift in reference temperature of surveillance specimens to the fluence. The selection of Position (1) or (2) is made based on the availability of credible surveillance data, and the results achieved in applying the two Positions.

The actual shift in the reference temperature of surveillance specimens and neutron fluence is established periodically by removing and evaluating reactor vessel material irradiation surveillance specimens and dosimetry installed near the inside wall of the reactor vessel in the core area.

The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of 15 EFPY, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The 15 EFPY heatup and cooldown curves were developed based on the following:

1. The intermediate shellplate, C5556-2, is the limiting material as determined by position 1 of Regulatory Guide 1.99, Revision 2, with a Cu and Ni content of 0.15% and 0.57%, respectively.

2. The fluence values contained in Table 6-14 of Westinghouse WCAP-13515 report, "Analysis of Capsule U From the Indiana Michigan Power Company D. C. Cook Unit 2 Reactor Vessel Radiation Surveillance Program", dated February 1993.

The RT_{NDT} shift of the reactor vessel material has been established by removing and evaluating the reactor material surveillance capsules in accordance with the removal schedule in Table 4.4-5. Per this schedule, Capsule U is the last capsule to be removed until Capsule S is to be removed after 32 EFPY (EOL). Capsules V, W, and Z will remain in the reactor vessel and will be removed to address industry reactor vessel embrittlement concerns, if required.

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.

The OPERABILITY of two PORVs, or of one PORV and the RHR safety valve ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 152°F. Either PORV or RHR safety valve has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the stem generator less than or equal to 50°F above the RCS cold leg temperatures of (2) the start of a charging pump and its injection into a water solid RCS. Therefore, any one of the three blocked open PORVs constitutes an acceptable RCS vent to preclude APPLICABILITY of Specification 3.4.9.3.

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.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 171 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 2

DOCKET NO. 50-316

1.0 INTRODUCTION

By letter dated February 22, 1994, the Indiana Michigan Power Company (the licensee) requested an amendment to the Technical Specifications (TS) appended to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Unit No. 2. The proposed amendment would revise pressure vessel heatup and cooldown curves and extend the applicability time from 12 effective full power years (EFPYs) of operation to 15 EFPYs. The proposed TS changes are based on the results of the D.C. Cook Unit 2 surveillance capsule U which was removed after exposure of 8.65 EFPYs and for which the results are documented in the Westinghouse report WCAP-13515 (Ref. 1).

Our review is based on the acceptability of the estimated fast neutron (E>1.0 MeV) fluence on the inside surface of the pressure vessel.

2.0 EVALUATION

WCAP-13515 reports the evaluation of the fast neutron dosimetry measurements and the estimated values at the location of the surveillance capsule as well as for the extrapolated values for the inside surface of the pressure vessel. The calculations for the processing of the dosimetry data as well as for the estimation of the fluence at the inside surface complied with staff recommendations for such calculations. WCAP-13515 used the two-dimensional discrete ordinates DOT code (Ref. 2) with a S_8 order of angular quadrature and a P_3 cross section approximation. The SAILOR cross section library (Ref. 3) was used with a 47 group cross section set based on ENDF/B-IV. The neutron sources were based on plant specific source estimates for the first 8 cycles and extrapolation to 16 and 32 EFPYs. The method and the approximations described above comply with staff recommendations in Regulatory Guide 1.99, Revision 2 and are acceptable.

The results provide the azimuthal fast neutron (E>1.0 MeV) fluence on the inside surface of the pressure vessel. A comparison of the calculated versus the measured exposure at the surveillance capsule falls within 8 percent of the total value and provides confidence for the results of the measurements and calculations.

Based on the use of acceptable methodologies and the acceptable agreement between the calculated and measured results we find the proposed values of the fluence for 16 EFPYs acceptable for the estimation of the heatup and cooldown curves. Therefore, based on the acceptable fluence predictions for 16 EFPYs, the extended applicability of the heatup and cooldown curves from 12 to 15 EFPYs is acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Michigan State official was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to the surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (59 FR 14891). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The staff has concluded, based on the considerations discussed above that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 REFERENCES

1. Fitzpatrick, E. E., Indiana Michigan Power Company, letter to T. Murley, U. S. Nuclear Regulatory Commission, March 12, 1993 transmitting Westinghouse Report No. WCAP-13515, "Analysis of Capsule U from the Indiana Michigan Power Company D.C. Cook Unit 2 Reactor Vessel Radiation Surveillance Program," February 1993.

2. Soltesz, R. G., WANL-PR(LL)-034, Volume 5, "Nuclear Rocket Shielding Methods, Modification, Update and Input Data Preparation, Vol. 5 Two-Dimensional Discrete Ordinates Transport Technique," August 1972.

3. DLC-76, "SAILOR, Coupled Self-Shielded, 47 Neutron 20 Gamma-Ray, P_3 Cross Section Library for Light Water Reactors," ORNL RSIC Library Collection.

Principal Contributor: L. Lois, SRXB/DSSA

Date: November 25, 1994