May 24, 1990

Docket No. 50-316

DISTRIBUTION:

Docket Files	DHagan
NRC PDR	EJordan
Local PDR	PD31 R/F
GHill(8)	OGC
Wanda Jones	ARM/LFMB
JZwolinski	EButcher
JGiitter	GPA/PA
MRShuttleworth	ACRS(10)

Mr. Milton P. Alexich, Vice President
Indiana Michigan Power Company
c/o American Electric Power Service Corporation
1 Riverside Plaza
Columbus, Ohio 43216

Dear Mr. Alexich:

SUBJECT: AMENDMENT NO. 123 TO FACILITY OPERATING LICENSE NO. DPR-74: (TAC NOS. 71481 AND 75261)

The Commission has issued the enclosed Amendment No. 123 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Unit No. 2. The amendment consist of changes to the Technical Specifications in response to your application dated October 25, 1989.

This amendment changes Technical Specification (TS) 3/4.4.9, "Pressure/Temperature Limits," to limit the maximum heatup rate to 60°F/hr and to provide revised heatup and cooldown pressure-temperature (P-T) limit curves. The maximum heatup rate is currently limited to 100°F/hr. The revisions are based on a reanalysis of reactor vessel sample material in accordance with Regulatory Guide (RG) 1.99, Rev 2. Through this licensing action the requirements of Generic Letter 88-11 have been satisfied for D. C. Cook, Unit No. 2

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

Sincerely,

/s/

Joseph Giitter, Project Manager
Project Directorate III-1
Division of Reactor Projects - III, IV, V & Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No.123 to DPR-74
2. Safety Evaluation

cc w/enclosures:
See next page

LA/PD31:DRSP MMR	PM/PD31:DRSP	(A)D/PD31:DRSP	OGC
MRShuttleworth	JGiitter	JThoma	
04/17/90	04/20/90	04/23/90	04/30/90

OFFICIAL RECORD COPY

DF01

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

May 24, 1990

Docket No. 50-316

Mr. Milton P. Alexich, Vice President
Indiana Michigan Power Company
c/o American Electric Power Service Corporation
1 Riverside Plaza
Columbus, Ohio 43216

Dear Mr. Alexich:

SUBJECT: AMENDMENT NO. 123 TO FACILITY OPERATING LICENSE NO. DPR-74: (TAC NOS. 71481 AND 75261)

The Commission has issued the enclosed Amendment No. 123 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Unit No. 2. The amendment consist of changes to the Technical Specifications in response to your application dated October 25, 1989.

This amendment changes Technical Specification (TS) 3/4.4.9, "Pressure/Temperature Limits," to limit the maximum heatup rate to 60°F/hr and to provide revised heatup and cooldown pressure-temperature (P-T) limit curves. The maximum heatup rate is currently limited to 100°F/hr. The revisions are based on a reanalysis of reactor vessel sample material in accordance with Regulatory Guide (RG) 1.99, Rev 2. Through this licensing action the requirements of Generic Letter 88-11 have been satisfied for D. C. Cook, Unit No. 2

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,

for

Joseph Giitter, Project Manager
Project Directorate III-1
Division of Reactor Projects - III, IV, V & Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 123 to DPR-74
2. Safety Evaluation

cc w/enclosures:
See next page

Mr. Milton Alexich
Indiana Michigan Power Company

Donald C. Cook Nuclear Plant

cc:
Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Attorney General
Department of Attorney General
525 West Ottawa Street
Lansing, Michigan 48913

Township Supervisor
Lake Township Hall
Post Office Box 818
Bridgman, Michigan 49106

Al Blind, Plant Manager
Donald C. Cook Nuclear Plant
Post Office Box 458
Bridgman, Michigan 49106

U.S. Nuclear Regulatory Commission
Resident Inspectors Office
7700 Red Arrow Highway
Stevensville, Michigan 49127

Gerald Charnoff, Esquire
Shaw, Pittman, Potts and Trowbridge
2300 N Street, N.W.
Washington, DC 20037

Mayor, City of Bridgman
Post Office Box 366
Bridgman, Michigan 49106

Special Assistant to the Governor
Room 1 - State Capitol
Lansing, Michigan 48909

Nuclear Facilities and Environmental
Monitoring Section Office
Division of Radiological Health
Department of Public Health
3500 N. Logan Street
Post Office Box 30035
Lansing, Michigan 48909

Mr. S. Brewer
American Electric Power
Service Corporation
1 Riverside Plaza
Columbus, Ohio 43216

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

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 123
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 October 25, 1989, 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. 123 , 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 beginning of Cycle 8.

FOR THE NUCLEAR REGULATORY COMMISSION

Dominic C. DiIanni

Dominic C. DiIanni, Acting Director
Project Directorate III-1
Division of Reactor Projects - III, IV, V & Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: May 24, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 123

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 marginal lines indicating the area of change.

REMOVE	INSERT
-	3/4 4-23*
3/4 4-24	3/4 4-24
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 4-7	B 3/4 4-7
B 3/4 4-8	B 3/4 4-8
B 3/4 4-9	B 3/4 4-9
B 3/4 4-9a	B 3/4 4-9a
B 3/4 4-10	B 3/4 4-10

*Overleaf page provided to maintain document completeness. No change contained on this page.

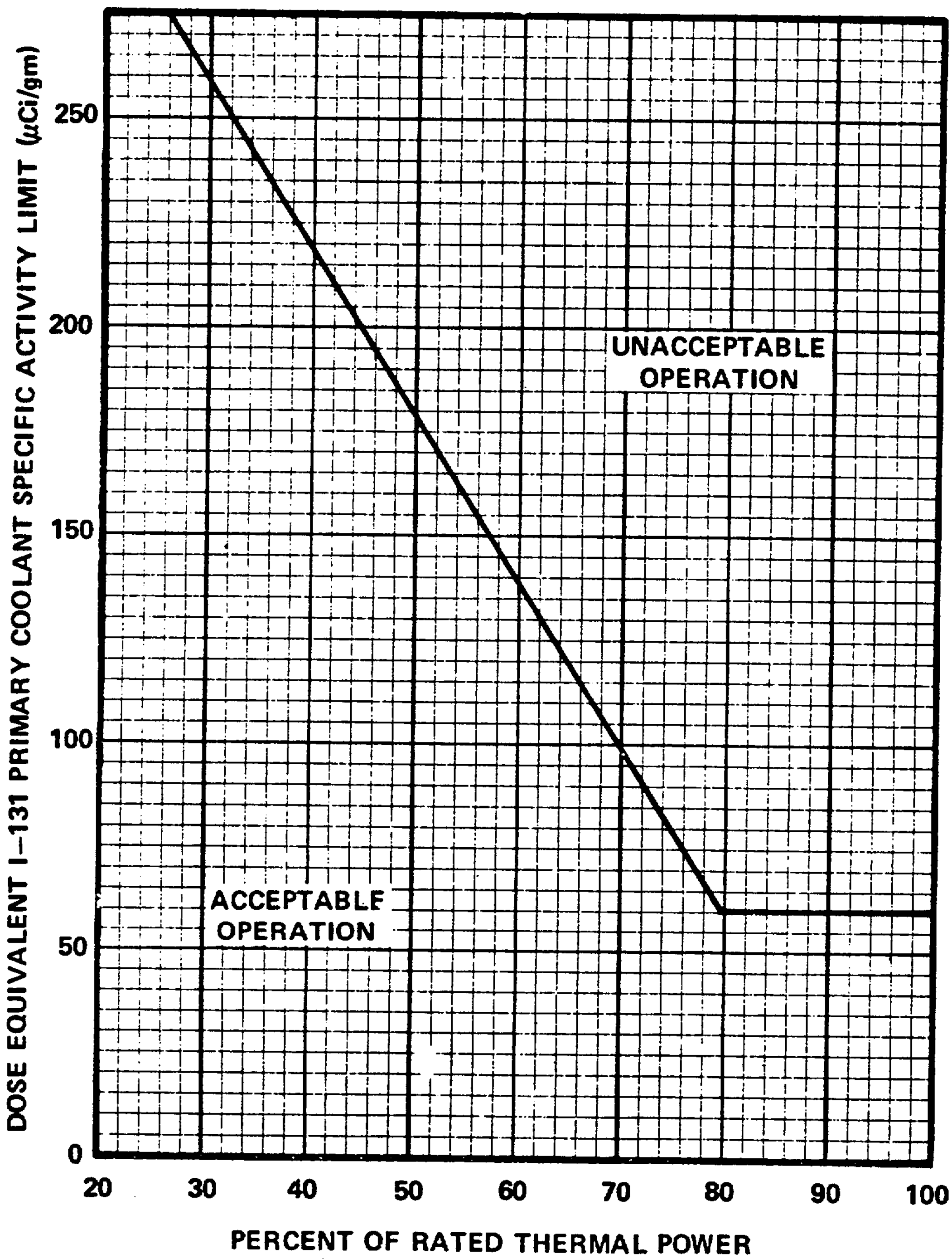

FIGURE 3.4-1

DOSE EQUIVALENT I–131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity > 1.0μCi/gram Dose Equivalent I-131

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 60°F in any one hour period.

b. A maximum cooldown of 100°F in any one hour period.

c. A maximum temperature of less than or equal to 5°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals shown in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3.

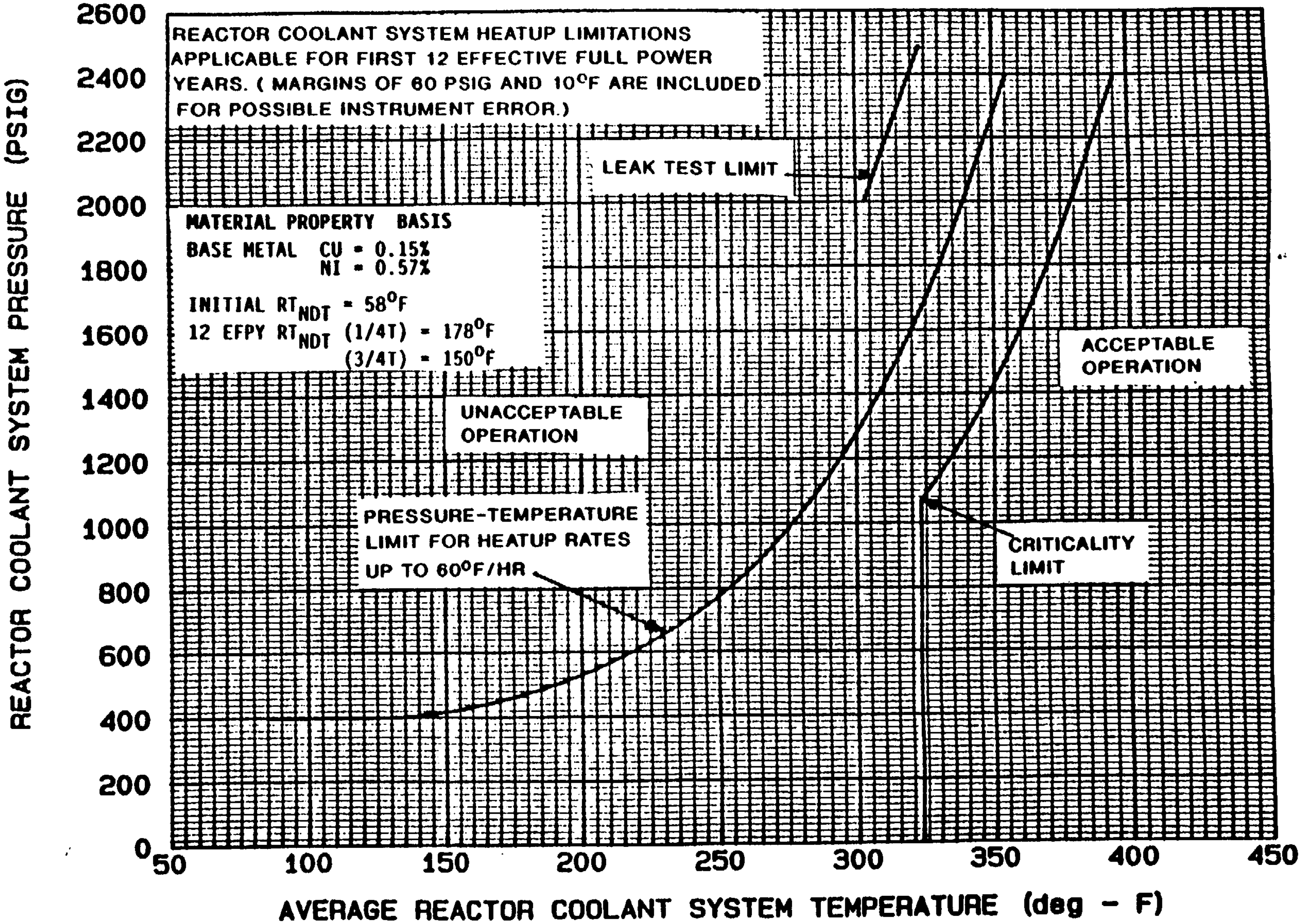

FIGURE 3.4-2

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

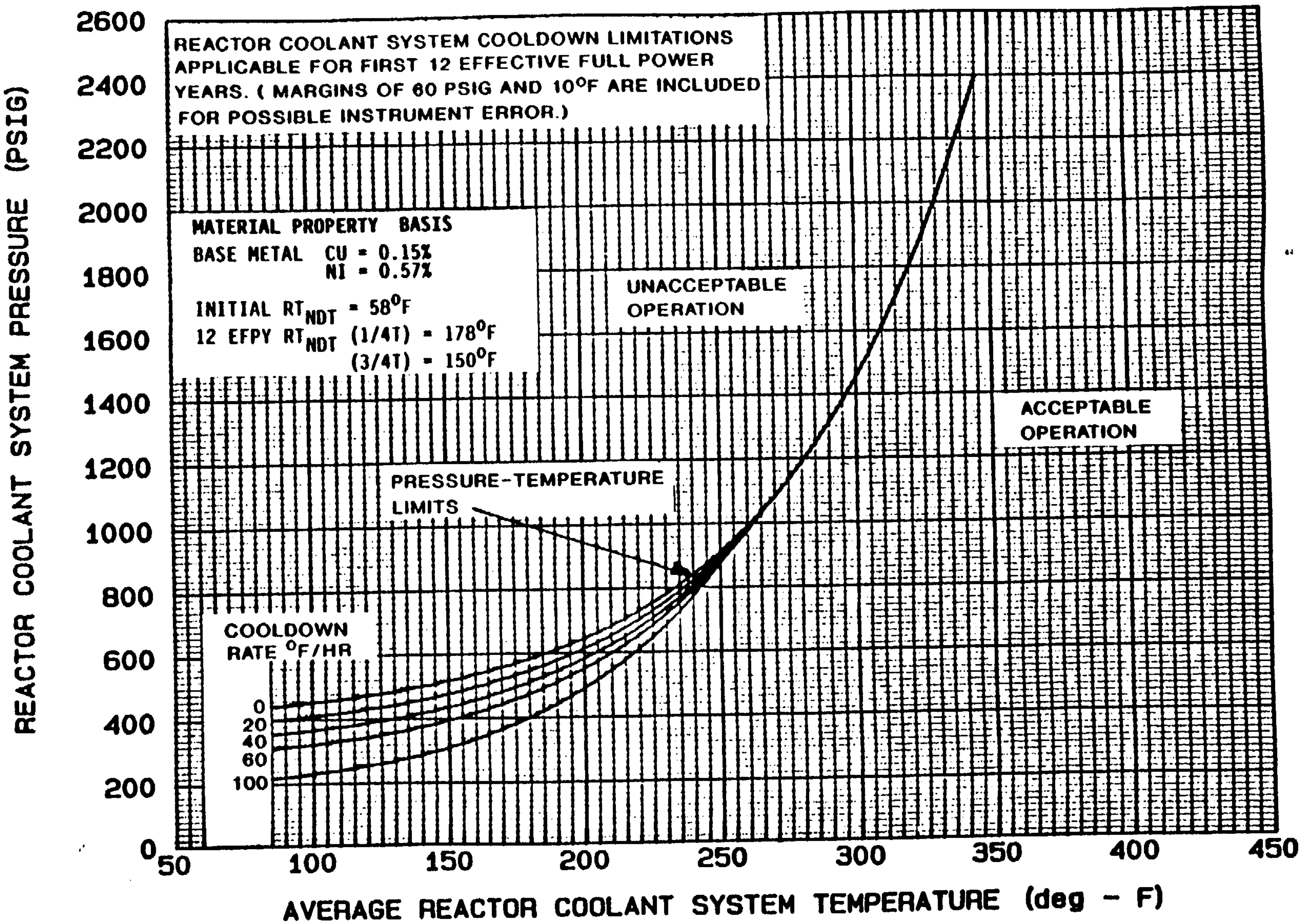

FIGURE 3.4-3

REACTOR COOLANT SYSTEM PRESSURE - TEMPERATURE LIMITS VERSUS 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 12 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.

INTENTIONALLY

LEFT

BLANK

INTENTIONALLY

LEFT

BLANK

TABLE B 3/4.4-1

REACTOR VESSEL TOUGHNESS

COMPONENT	CODE NO.	MATERIAL TYPE	CU (%)	NI (%)	T_{NDT} (°F)	50 FT-LB 35 MIL Temp(a) (°F)	RT_{NDT} (°F)	USE MWD (ft-lb)	USE NMWD(a) (ft-lb)
CL. HD. DOME	B0048-2	A533B CL.1	NA	0.64	-20	30	-20	148	96
CL. HD. SEG.	B9883-2	A533B CL.1	NA	0.66	-20	-3	-20	143.5	93
CL. HD. SEG.	A5189-2	A533B CL.1	NA	0.63	10	72	12	140.5	91
CL. HEAD FLG.	4437-V-1	A508 CL.2	NA	0.70	-20	5	-20	239	155
VESSEL FLANGE	4436-V-2	A508 CL.2	NA	0.70	30	15	30	161	105
INLET NOZZLE	269T-2	A508 CL.2	NA	0.85	-20	-15	-20	201.5	131
INLET NOZZLE	270T-1	A508 CL.2	NA	0.91	-20	-3	-20	239.5	156
INLET NOZZLE	269T-1	A508 CL.2	NA	NA	-10	NA	-10	NA	NA
INLET NOZZLE	270T-2	A508 CL.2	NA	NA	-10	NA	-10	NA	NA
OUTLET NOZZLE	271T-1	A508 CL.2	NA	0.80	0	12	0	>179	NA
OUTLET NOZZLE	271T-2	A508 CL.2	NA	0.80	0	-15	0	181	117.5
OUTLET NOZZLE	272T-1	A508 CL.2	NA	NA	-10	NA	-10	NA	NA
OUTLET NOZZLE	272T-2	A508 CL.2	NA	NA	0	NA	0	NA	NA
UPPER SHELL	C5518-2	A533B CL.1	.12	0.61	10	88	28	107.5	70
UPPER SHELL	C5521-1	A533B CL.1	.14	0.59	0	93	33	112	73
UPPER SHELL	C5518-1	A533B CL.1	.12	0.57	10	66	10	> 82.5	NA
INTER SHELL	C5556-2	A533B CL.1	.15	0.57	0	118(b)	58(b)	109.5	90(b)
INTER SHELL	C5521-2	A533B CL.1	.14	0.58	10	98(b)	38(b)	111.5	86(b)
LOWER SHELL	C5540-2	A533B CL.1	.11	0.64	-20	35(b)	-20(b)	113	110(b)
LOWER SHELL	C5592-1	A533B CL.1	.14	0.59	-20	25(b)	-20(b)	107	103(b)
BOT. HD. SEG.	C5823-2	A533B CL.1	NA	0.57	-10	45	-10	129	84
BOT. HD. SEG.	A4957-3	A533B CL.1	NA	0.51	-10	20	-10	149	97
BOT. HD. SEG.	B0019-18	A533B CL.1	NA	0.61	-50	0	-50	177	115
INTER. & LOWER SHELL LONG. and GIRTH WELD SEAM	(HT S3986 & SAW Linde 124 Flux Lot No. 0934)		.06	0.97	-40	25	-35(b)	NA	97(b)

a) Estimated per NRC Standard Review Plan
b) Actual values
NA - Not available or not applicable, as appropriate

MWD - Major Working Direction
NMWD - Normal to MWD

INTENTIONALLY

LEFT

BLANK

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 12 EFPY, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The 12 EFPY heatup and cooldown curves were developed based on the following:

1. The projected fluence values established by specimen analysis.

2. Intermediate shell plate C5556-2 being the limiting material as determined by Position 1 of Regulatory Guide 1.99, Revision 2, with a copper and nickel content of 0.15% and 0.57%, respectively.

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, one PORV and the RHR safety valve, or an RCS vent opening of greater than or equal to 2 square inches 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 steam generator less than or equal to 50°F above the RCS cold leg temperatures or (2) the start of a charging pump and its injection into a water solid RCS.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 123 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

In response to Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Effect on Plant Operations," the Indiana Michigan Power Company (the licensee) requested permission to revise the pressure/temperature (P/T) limits for the Donald C. Cook Nuclear Power Plant, Unit No. 2. The request was documented in letters from the licensee dated December 5, 1988 and October 25, 1989. This revision also changes the effectiveness of the P/T limits from 32 to 12 effective full power years (EFPY). The proposed P/T limits were developed based on the data from actual surveillance capsules. The proposed revision provides up-to-date P/T limits for the operation of the reactor coolant system during heatup, cooldown, criticality, and hydrotest.

To evaluate the P/T limits, the staff uses the following NRC regulations and guidance: Appendices G and H of 10 CFR Part 50; the ASTM Standards and the ASME Code, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); Regulatory Guide (RG) 1.99, Rev. 2; Standard Review Plant (SRP) Section 5.3.2; and Generic Letter 88-11.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide Technical Specifications for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions of operation be included in the Technical Specifications. The P/T limits are among the limiting conditions of operation in the Technical Specifications for all commercial nuclear power plants in the U.S. Appendices G and H of 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P/T limits. An acceptable method for constructing the P/T limits is described in SRP Section 5.3.2.

Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, that the beltline materials in the surveillance capsules be tested in accordance with Appendix H of 10 CFR Part 50. Appendix H, in turn, refers to ASTM Standards. These tests define the extent of vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature. Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE). Generic Letter 88-11 requested that licensees and permitters use the methods in RG 1.99, Rev. 2, to predict the effect of neutron irradiation on reactor vessel materials. This guide defines the ART as the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

Appendix H of 10 CFR Part 50 requires the licensee to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standards which, in turn, require that the capsules be installed in the vessel before startup and that they contain test specimens made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline.

2.0 EVALUATION

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the D. C. Cook Unit No. 2 (Cook 2) reactor vessel. The amount of irradiation embrittlement was calculated in accordance with Section 2 of RG 1.99, Rev. 2. The staff has determined that the material with the highest ART at 12 EFPY for Cook 2 was intermediate shell plate C5556-2 with 0.15% Cu, 0.57% Ni, and an initial RT_{ndt} of 58°F.

The licensee has removed three surveillance capsules from Cook 2. The results from capsules T, Y, and X in Unit 2 were published in Southwest Research Institute Reports 06-5928, SwRI-7244-002/1, and 06-8888, respectively. All surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the Cook 2 limiting beltline material, intermediate shell plate C5556-2, the staff calculated the ART to be 177.5°F at 1/4T (T = reactor vessel beltline thickness) and 149°F for 3/4T at 12 EFPY. The staff used a neutron fluence of 4.69E18 n/cm^2 at 1/4T and 1.66E18 n/cm^2 at 3/4T. The ART was determined by Section 1 of RG 1.99, Rev. 2, because the most limiting beltline was not in the surveillance capsule.

The licensee used the method in RG 1.99, Rev. 2, to calculate an ART of 178°F at 1/4T at 12 EFPY for the same limiting weld metal. Substituting the ART of 178°F into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Based on the flange reference temperature of 30°F, the staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G.

Section IV.B of Appendix G requires that the predicted Charpy USE at end of life be above 50 ft-lb. Using the method in RG 1.99, Rev. 2, the predicted Charpy USE of intermediate shell plate C5521-2 at the end of life will be 62.3 ft-lb. This is greater than 50 ft-lb and, therefore, is acceptable.

The staff concludes that the proposed P/T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 12 EFPY because the limits conform to the requirements of Appendices G and H of 10 CFR Part 50. The licensee's submittal also satisfies Generic Letter 88-11 because the licensee used the method in RG 1.99, Rev. 2 to calculate the ART. Hence, the proposed P/T limits may be incorporated into the Cook 2 Technical Specifications.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in 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 a change in a surveillance requirement. We have 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendments meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9).

4.0 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) issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: May 24, 1990

Principal Contributor: J. Tsao, DET/EMCB