October 26, 1992

Docket No. 50-315

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

Dear Mr. Fitzpatrick:

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNIT 1 - AMENDMENT NO. 167 TO FACILITY OPERATING LICENSE NO. DPR-58 (TAC NO. M71480 AND M75260)

The Commission has issued the enclosed Amendment No. 167 to Facility Operating License No. DPR-58 for the Donald C. Cook Nuclear Plant, Unit No. 1. The amendment consist of changes to the Technical Specifications in response to your application dated October 29, 1990 as supplemented June 18, 1991 and April 13, 1992.

This amendment changes Technical Specifications (TS) concerning, "Pressure-Temperature Limits," to limit the maximum heat up rate to 60°F/hr and to provide revised heat up and cooldown pressure-temperature (P-T) limit curves. In addition, the amendment changes the TS to reflect revisions to the pressure set point and enable temperature for the low-temperature overpressure protection system. The revisions are based on a reanalysis of reactor vessel sample material in accordance with Regulatory Guide 1.99, Revision 2. Through this licensing action the guidance of Generic Letter 88-11 has been followed for D. C. Cook, Unit No. 1.

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 F. Stang, Project Manager
Project Directorate III-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 167 to DPR-58
2. Safety Evaluation

cc w/enclosures:
See next page

*SEE PREVIOUS CONCURRENCE

OFFICE	LA:PD31*	PM:PD31 *	OGC*	D:PD31
NAME	MShuttleworth	JStang:sw	MZobler	LMarsh
DATE	09/09/92	9/30/92	10/08/92	10/23/92

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FILENAME: B:C071480.AMD

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Indiana Michigan Power Company

Donald C. Cook Nuclear Plant

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DATED: October 26, 1992

AMENDMENT NO. 167 TO FACILITY OPERATING LICENSE NO. DPR-58-D. C. COOK

Docket File
NRC & Local PDRs
PDIII-1 Reading
D.C. Cook Plant File
B. Boger
J. Zwolinski
L. Marsh
M. Shuttleworth
J. Stang
OGC-WF
D. Hagan, 3302 MNBB
G. Hill (4), P-137
Wanda Jones, MNBB-7103
C. Grimes, 11/F/23
S. Sheng, 7/D/4
H. Ableson, 7/D/4
ACRS (10)
GPA/PA
OC/LFMB
W. Shafer, R-III
A. DeAgazio, 14/C/7

cc: Plant Service list

201173

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

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 167
License No. DPR-58

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

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated October 29, 1990 as supplemented June 18, 1991 and April 13, 1992, 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-58 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 167 , 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 with full implementation within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

LBMarsh

Ledyard B. Marsh, Director
Project Directorate III-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: October 26, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 167

TO FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

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 1-11	3/4 1-11
3/4 1-11a	3/4 1-11a
3/4 4-3	3/4 4-3
3/4 4-25	3/4 4-25
3/4 4-26*	3/4 4-26*
3/4 4-27	3/4 4-27
3/4 4-28	3/4 4-28
3/4 4-31	3/4 4-31
3/4 5-7	3/4 5-7
3/4 5-8	3/4 5-8
B 3/4 1-2	B 3/4 1-2
B 3/4 4-1	B 3/4 4-1
B 3/4 4-6	B 3/4 4-6
B 3/4 4-7	B 3/4 4-7
B 3/4 5-2	B 3/4 5-2

*Overleaf page provided to maintain document completeness. No changes are contained in this page.

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3

a. One charging pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency bus.

b. One charging flowpath associated with support of Unit 2 shutdown functions shall be available.*

APPLICABILITY: Specification 3.1.2.3.a. - MODES 5 and 6
Specification 3.1.2.3.b. - At all times when Unit 2 is in MODES 1, 2, 3, or 4.

ACTION:

a. With no charging pump OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.**

b. With more than one charging pump OPERABLE or with a safety injection pump(s) OPERABLE when the temperature of any RCS cold leg is less than or equal to 152°F, unless the reactor vessel head is removed, remove the additional charging pump(s) and the safety injection pump(s) motor circuit breakers from the electrical power circuit within one hour.

c. The provisions of Specification 3.0.3 are not applicable.

d. In addition to the above, when Specification 3.1.2.3.b is applicable and the required flow path is not available, return the required flow path to available status within 7 days, or provide equivalent shutdown capability in Unit 2 and return the required flow path to available status within the next 60 days, or have Unit 2 in HOT STANDBY within the next 12 hours and HOT SHUTDOWN within the following 24 hours.

e. The requirements of Specification 3.0.4 are not applicable when Specification 3.1.2.3.b applies.

SURVEILLANCE REQUIREMENTS

4.1.2.3.1 The above required charging pump shall be demonstrated OPERABLE by verifying, that on recirculation flow, the pump develops a discharge pressure of greater than or equal to 2390 psig when tested pursuant to Specification 4.0.5.

*A maximum of one centrifugal charging pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 152°F.

**For purposes of this specification, addition of water from the RWST does not constitute a positive reactivity addition provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2.

REACTIVITY CONTROL SYSTEMS

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

4.1.2.3.2 All charging pumps and safety injection pumps, excluding the above required OPERABLE charging pump, shall be demonstrated inoperable by verifying that the motor circuit breakers have been removed from their electrical power supply circuits at least once per 12 hours, except when:

a. The reactor vessel head is removed, or

b. The temperature of all RCS cold legs is greater than 152°F.

4.1.2.3.3 Charging line cross-tie valves to Unit 2 will be cycled full travel at least once per 18 months. Following cycling, the valves will be verified to be in their closed positions.

REACTOR COOLANT SYSTEM

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 a. The coolant loops listed below shall be OPERABLE and in operation as required by items b and c:

1. Reactor Coolant Loop 1 and its associated steam generator and reactor coolant pump,*
2. Reactor Coolant Loop 2 and its associated steam generator and reactor coolant pump,*
3. Reactor Coolant Loop 3 and its associated steam generator and reactor coolant pump,*
4. Reactor Coolant Loop 4 and its associated steam generator and reactor coolant pump,*
5. Residual Heat Removal - East,**
6. Residual Heat Removal - West,**

b. At least two of the above coolant loops shall be OPERABLE and at least one loop in operation if the reactor trip breakers are in the open position, or the control rod drive system is not capable of rod withdrawal.***

c. At least three of the above reactor coolant loops shall be OPERABLE and in operation when the reactor trip system breakers are in the closed position and the control rod drive system is capable of rod withdrawal.

APPLICABILITY: MODES 4 and 5

* A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to 152°F unless 1) the pressurizer water volume is less than 62% of span or 2) the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures. Operability of a reactor coolant loop(s) does not require an OPERABLE auxiliary feedwater system.

** The normal or emergency power source may be inoperable in MODE 5.

*** All reactor coolant pumps and residual heat removal pumps may be de-energized for up to 1 hour provided 1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration****, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

**** For purposes of this specification, addition of water from the RWST does not constitute a dilution activity provided the boron concentration in the RWST is greater than or equal to the minimum required by specification 3.1.2.8.b.2 (MODE 4) or 3.1.2.7.b.2 (MODE 5).

REACTOR COOLANT SYSTEM

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, and

c. A maximum temperature change of $\leq$ 5°F in any one hour period, during hydrostatic testing operations above system design pressure.

APPLICABILITY: At all times.*

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an analysis 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 operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS Tavg and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

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* See Special Text Exception 3.10.3.

SURVEILLANCE REQUIREMENTS

4.4.9.1

a. 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.

b. The Reactor Coolant System temperature and pressure conditions shall be determined to be to the right of the criticality limit line within 15 minutes prior to achieving reactor criticality.

c. 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.

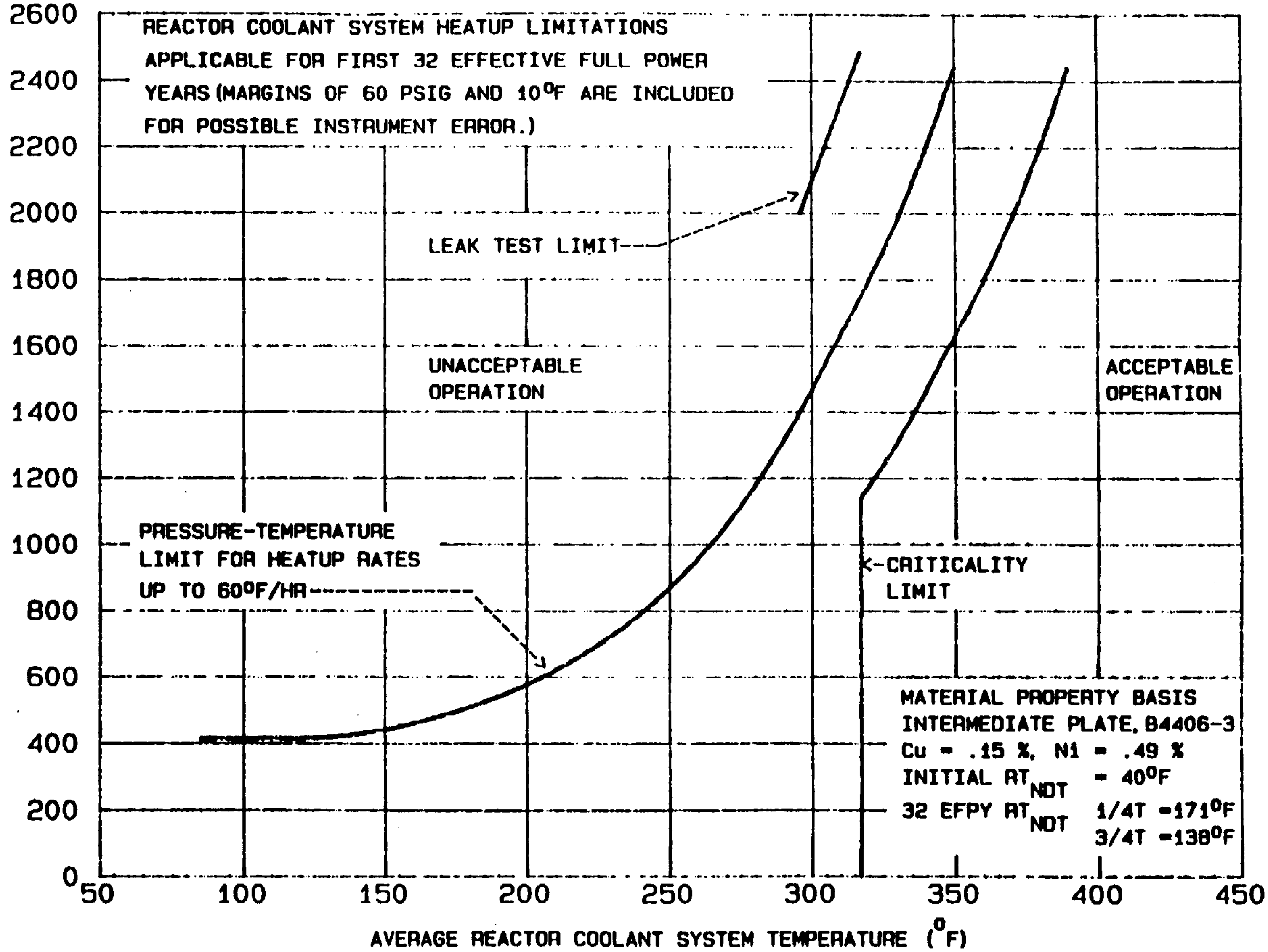

FIGURE 3.4-2

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

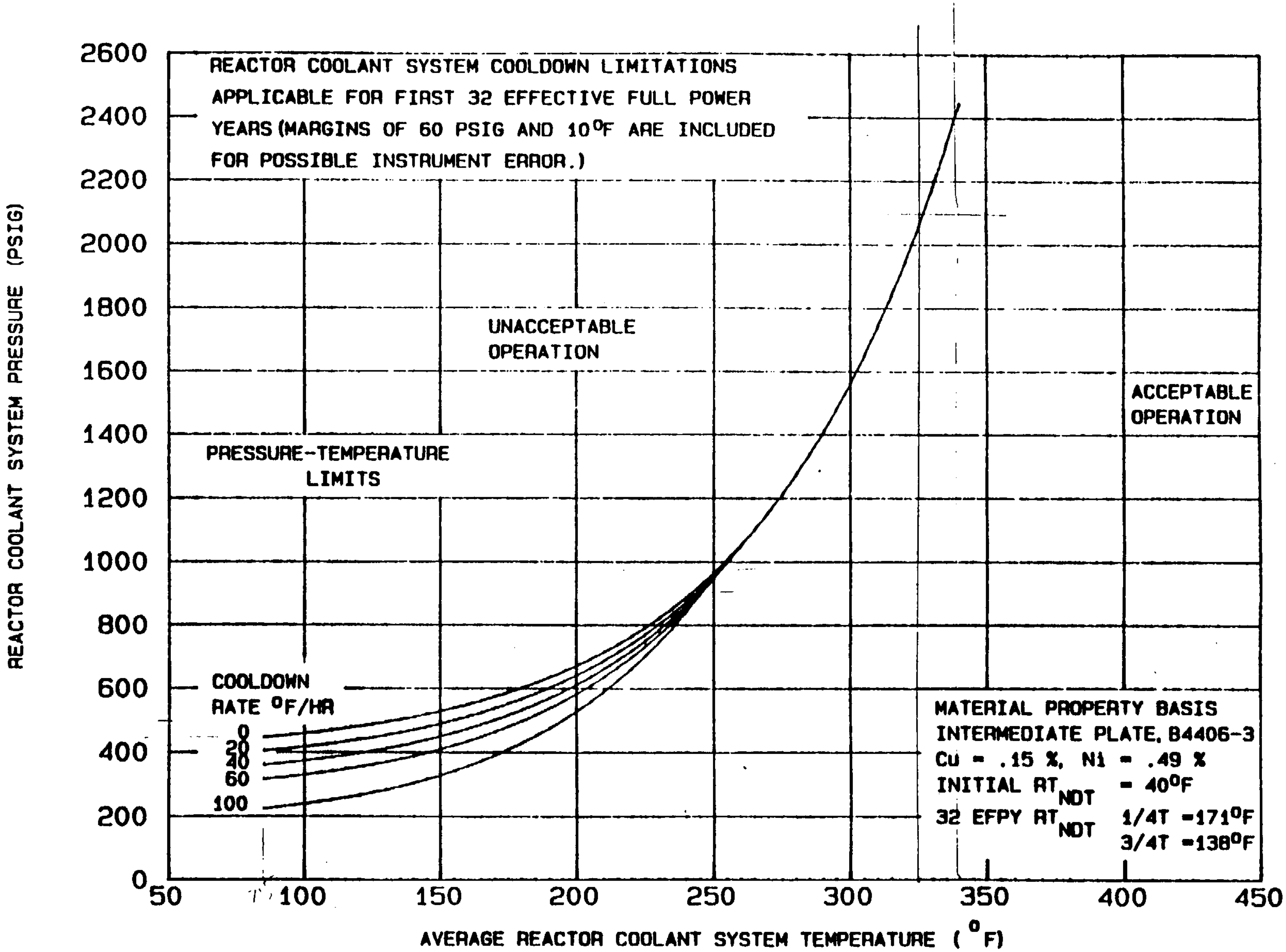

FIGURE 3.4-3

REACTOR COOLANT SYSTEM PRESSURE - TEMPERATURE LIMITS VERSUS COOLDOWN RATES

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 At least one of the following overpressure protection systems shall be OPERABLE:

a. Two power operated relief valves (PORVs) with a lift setting of less than or equal to 435 psig, or

b. One power operated relief valve (PORV) with a lift setting of less than or equal to 435 psig and the RHR safety valve with a lift setting of less than or equal to 450 psig, or

c. A reactor coolant system vent of greater than or equal to 2 square inches.

APPLICABILITY: When the temperature of one or more of the RCS cold legs is less than or equal to 152°F, except when the reactor vessel head is removed.

ACTION:

a. With two PORV's inoperable or with one PORV inoperable and the RHR safety valve inoperable, either restore the inoperable PORV(s) or RHR safety valve to OPERABLE status within 7 days or depressurize and vent the RCS through an at least 2 square inch vent(s) within the next 8 hours; maintain the RCS in a vented condition until the inoperable PORV or RHR safety valve has been restored to OPERABLE status.

b. With both PORVs inoperable, depressurize and vent the RCS through an at least 2 square inch vent(s) within 8 hours; maintain the RCS in a vented condition until both PORVs or one PORV and the RHR safety valve have been restored to OPERABLE status.

c. In the event either the PORVs, the RHR safety valve or the RCS vent(s) are used to mitigate a RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or vent(s) on the transient and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

EMERGENCY CORE COOLING SYSTEMS

ECCS SUBSYSTEMS - Tavg < 350°F

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE centrifugal charging pump,#

b. One OPERABLE residual heat removal heat exchanger,

c. One OPERABLE residual heat removal pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank upon being manually realigned and transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the centrifugal charging pump or the flow path from the refueling water storage tank, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of either the residual heat removal heat exchanger or residual heat removal pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System Tavg less than 350°F by use of alternate heat removal methods.

c. With more than one charging pump OPERABLE or with a safety injection pump(s) OPERABLE when the temperature of any RCS cold leg is less than or equal to 152°F, remove the additional charging pump(s) and the safety injection pump(s) motor circuit breakers from the electrical power circuit within 1 hour.

d. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

#A maximum of one centrifugal charging pump shall be OPERABLE and both safety injection pumps shall be inoperable whenever the temperature of one or more of the RCS cold legs is less than or equal to 152°F.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2.

4.5.3.2 All charging pumps and safety injection pumps, except the above required OPERABLE charging pump, shall be demonstrated inoperable, by verifying that the motor circuit breakers have been removed from their electrical power supply circuits, at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is less than or equal to 152°F as determined at least once per hour when any RCS cold leg temperature is between 152°F and 200°F.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC) (Continued)

concentration associated with fuel burnup. The confirmation that the measured and appropriately compensated MTC value is within the allowable tolerance of the predicted value provides additional assurances that the coefficient will be maintained within its limits during intervals between measurement.

3/4.1.1.5 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 541°F. This limitation is required to ensure 1) the moderator temperature coefficient is within its analyzed temperature range, 2) the protective instrumentation is within its normal operating range, and 3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and 4) the reactor pressure vessel is above its minimum RT_{NDT} temperature. Administrative procedures will be established to ensure the P-12 blocked functions are unblocked before taking the reactor critical.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid transfer pumps, 5) associated heat tracing systems, and 6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps and safety injection pumps, except the required OPERABLE charging pump, to be inoperable below 152°F, unless the reactor vessel head is removed, provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The boration capability of either system is sufficient to provide the required SHUTDOWN MARGIN from all operating conditions after xenon decay and cooldown to 200°F. The maximum expected boration capability, usable volume requirement, is 5641 gallons of 20,000 ppm borated water from the boric acid storage tanks or 99,598 gallons of 2400 ppm borated water from the refueling water storage tank. The minimum contained RWST volume is based on ECCS considerations. See Section B 3/4.5.5.

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with all reactor coolant loops in operation, and maintain DNBR above 1.69 during all normal operations and anticipated transients. A loss of flow in two loops will cause a reactor trip if operating above P-7 (11 percent of RATED THERMAL POWER) while a loss of flow in one loop will cause a reactor trip if operating above P-8 (31 percent of RATED THERMAL POWER).

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE. Three loops are required to be OPERABLE and to operate if the control rods are capable of withdrawal and the reactor trip breakers are closed. The requirement assures adequate DNBR margin in the event of an uncontrolled rod withdrawal in this mode.

In MODES 4 and 5, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two RHR loops to be OPERABLE.

The operation of one Reactor Coolant Pump or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump below P-7 with one or more RCS cold legs less than or equal to 152°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into or (2) by restricting starting of the RCP's to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

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 upon the most limiting value of the predicted adjusted reference temperature at the end of 32 EFPY.

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 the copper and nickel content of the material must be predicted. The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include the adjusted RT_{NDT} at the end of 32 EFPY, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The 32 EFPY heatup and cooldown curves were developed based on the following:

1. The intermediate shell plate, B4406-3, being the limiting material with a copper and nickel content of .15% and .49% respectively.

2. The fluence values contained in Table 6-14 of Westinghouse's WCAP-12483 report, "Analysis of Capsule U From the American Electric Power Company D. C. Cook Unit 1 Reactor Vessel Radiation Surveillance Program," dated January 1990.

3. Figure 1, NRC Regulatory Guide 1.99, Revision 2

The shift in RT_{NDT} of the reactor vessel material has been established by removing and evaluating the material surveillance capsules installed near the inside wall of the reactor vessel 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). Capsule V, W, and Z will remain in the reactor vessel, and will be removed to address industry reactor 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, 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.

EMERGENCY CORE COOLING SYSTEM

BASES

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps and safety injection pumps, except the required OPERABLE charging pump, to be inoperable below 152°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 167 TO FACILITY OPERATING LICENSE NO. DPR-58

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1

DOCKET NO. 50-315

1.0 INTRODUCTION

By letter dated October 29, 1990, the Indiana Michigan Power Company (IMPC) proposed changes to the Technical Specifications (TS) for Unit 1 which reflect revisions to the pressure set point and enable temperature for the low-temperature overpressure (LTOP) protection system. In addition, the October 29, 1990, letter provided the response for the D. C. Cook Nuclear Power Plant (DCCNP) for Generic Letter (GL) 88-11. Supplementary information was submitted by letters dated June 18, 1991 and April 13, 1992 in response to staff requests. Reevaluation and revision of the LTOP set point was necessitated by the generation of more restrictive 10 CFR Part 50, Appendix G heat up and cooldown curves based on a reactor vessel Capsule U analysis, conducted in accordance with Revision 2 to Regulatory Guide 1.99. Results of the Capsule U vessel exposures up to 32 effective full power-years (EFPY) were provided to the NRC in a submittal dated June 22, 1990. The revised pressure set point of 435 psig will replace the current value of 400 psig in TS 3.4.9.3. The revised set point is larger than the original value because a reduction in the allowance for instrument error more than offsets the change in the 10 CFR Part 50, Appendix G pressure limit. The revised enable temperature of 152°F will replace the current value of 170°F in the following TS: 3.1.2.3, 4.1.2.3.2, 3.4.1.3, 3.4.9.1, 3.4.9.3, 3.5.3, 4.5.3.2, and Bases 3/4.1.2, 3/4.4.1, 3/4.4.9.

The function of LTOPs is to prevent reactor coolant system (RCS) pressure from exceeding Appendix G limits during low temperature operations such as normal plant heat up, cooldown, and cold shutdown; especially, when the RCS is in a water-solid condition. This function is accomplished by venting through the two power-operated relief valves (PORVs) located near the top of the pressurizer whenever RCS pressure exceeds the LTOPs pressure set point and the temperature in one or more of the cold legs is below the enable temperature. The LTOPs pressure relieving capability supplements that of safety relief valve in the residual heat removal (RHR) system which, as per TS, is in operation and open to the RCS at temperature below 350°F. Additionally, DCCNP operating procedures require that a steam bubble be maintained in the pressurizer during low temperature operations. The presence of a steam bubble precludes water solid operation and, thus, mitigates the effect of a potential pressure transient.

Each of the PORVs is designed with adequate relief capacity to prevent RCS pressure from exceeding Appendix G limits when the following limiting transients occur in a water-solid RCS:

(1) A mass input transient caused by a charging/letdown flow mismatch. In this scenario, a single charging pump is in operation (in accordance with TS when RCS temperature is below the enable temperature) and the letdown path through the RHR system is inadvertently isolated due to spurious closure of the RHR isolation valves. As a result, the RHR safety relief valve becomes isolated, leaving only the LTOP activated pressurizer relief valves available for pressure relief.

(2) A heat input transient involving the startup of a reactor coolant pump (RCP) with temperature asymmetry existing between the primary side water in the steam generator and the coolant in the remainder of the RCS. In this scenario, all RCPs have been removed from service and the RCS is being cooled by the RHR system. Because of the reduced RCS circulation, particularly within the steam generator, the relatively stagnant water there remains at a higher temperature than the coolant in the remainder of the RCS loop. Upon startup of an RCP, the warmer water is "washed" out of the steam generator tubes and replaced by the cooler water which is then heated by the secondary side fluid. The resulting expansion of primary side water causes a pressure increase. The maximum possible temperature asymmetry is considered to be 50°F.

Selection of an LTOPs pressure set point entails the consideration of various system parameters. DCCNP employs a constant pressure set point, independent of temperature, in comparison to some plants which employ a variable set point based on temperature. An acceptable set point pressure range is established such that the maximum and minimum pressures occurring during the limiting transients stay within specified bounds. The upper bound is chosen as the lower of the RCS pressure corresponding to Appendix G limits or the RCS pressure corresponding to PORV piping structural limit (based on a water hammer analysis of PORV operation under water-solid conditions). In the case of DCCNP, the upper bound corresponds to the Appendix G limits. The lower bound corresponds to the minimum RCS pressure required for an RCP start (i.e., the RCP No. 1 seal limits) taken as 325 psig. In the event the set point range cannot accommodate both the upper and the lower bounds, the upper (Appendix G) bound takes precedence. For a given set point, the maximum pressure overshoot and undershoot occurring during a transient is dependent on the PORV opening and closure times, respectively, as well as related factors such as RCS pressure signal transmission delay and PORV volumetric capacity vs opening position.

Revision 2 of Branch Position RSB 5-2 in the Standard Review Plan defines the LTOPs enable temperature as RT(NDT) + 90(F). However, the licensee's proposed enable temperature (as well as the current and previous values) does not conform to this definition. The criterion used by the licensee in determining the proposed value is that the enable temperature be greater than the temperature below which the plant may operate in a water-solid condition. As noted earlier, this value is 150°F for DCCNP. Additionally, to ensure availability of the RHR safety valve while the RHR system is in service (and, thus, provide an alternative or supplementary means of pressure relief to LTOPs), the autoclosure interlock on the RHR isolation valves is administratively defeated.

2.0 EVALUATION

2.1 LTOP SET POINTS

The licensee utilized the LOFTRAN thermal-hydraulic code to model the limiting mass input and heat input transients described above. For the mass input transients, a series of cases was run at various LTOPs set points between 400 and 700 psig, over a range of mass injection rates and a range of PORV opening times varying from one to ten seconds. For conservatism, a low RCS temperature (85°F) corresponding to a high coolant bulk modulus was selected for the analyses. Valve closure time was fixed at four seconds. From these runs, a set of maximum and minimum pressures was determined for each set point value. Similarly, for the heat input transient, a series of cases was run at various set point values over the same range of PORV opening times and the same closure time. All heat input cases were run at RCS temperatures of both 85°F and 150°F, with an assumed asymmetry of 50°F with the steam generator. Again, a set of maximum and minimum pressure was computed for each set point value.

For both transients, only one of the two PORVs was assumed operable and no credit was taken for pressure relief through the RHR safety valve. A comparison of results from the mass input and heat input runs indicated that the most limiting temperature with respect to Appendix G criteria was 85°F and, at that temperature, the mass input transient represented the limiting case. Based on these analyses, a relationship between maximum allowable LTOPs set point and PORV opening time was developed for 32 EFPY.

The above analyses, as presented in the June 18, 1991 submittal, were originally prepared and previously submitted to support LTOPs set point TS changes for DCCNP Unit 2. However, because the LOFTRAN analyses were performed using the most conservative values of the relevant Unit 1 and Unit 2 plant parameters, the methodology and results are applicable to both units. The Appendix G limits for each unit, of course, are different and have been developed from the respective Capsule U analyses.

The proposed LTOPs set point pressure of 435 psig was selected to achieve consistency with the current Unit 2 set point value and is based on assumed PORV opening and closure times of 6.0 seconds and 4.0 seconds, respectively. Actual stroke times for the Unit 1 PORVs, as measured during past in-service testing, have been less than 5.0 seconds for opening and less than 2.0 seconds for closure. Since the maximum allowable LTOPs set point pressure decreases with increasing opening times, the assumed value of 6.0 seconds is, therefore, conservative. As a set point of 435 psig and a 6.0 second opening time, the maximum pressure overshoot was computed as approximately 506 psig. This value is below the Appendix G pressure list of around 515 psig at an RCS temperature of 85°F. The proposed pressure set point is, therefore, acceptable.

Using the criterion discussed earlier, the proposed enable temperature was selected as 152°F. This particular value was chosen to achieve consistency with the current Unit 2 value. As noted previously, the proposed enable temperature does not conform to the definition provided in Branch Position RSB 5-2 of the Standard Review Plan. However, the intent of the position is met because the availability of an alternative means of pressure relief, the RHR safety valve, is ensured because operating procedures preclude water-solid operation above the enable temperature. The proposed enable temperature is, therefore, acceptable.

2.2 GENERIC LETTER 88-11 RESPONSE

In response to Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Effect on Plant Operations," the IMPC requested permission to revise the pressure-temperature (P-T) limits for the Donald C. Cook Nuclear Power Plant Unit 1. The request was documented in a letter from the licensee dated October 29, 1990. This proposed P-T limits are valid for 32 effective full power years (EFPY). The proposed P-T limits were developed using Regulatory Guide (RG) 1.99, Rev. 2. Generic Letter 88-11 recommends RG 1.99, Rev. 2 be used in calculating P-T limits, unless the use of different methods can be justified. The P-T limits provide for the operation of the reactor coolants system during heat up, 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); RG 1.99, Rev.2; Standard Review Plan (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 plants in the U. S. Appendices G and H of 10 CFR Part 50 describe specific requirements for the 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 permittees 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 unirradiation 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.

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the DCCNP, Unit 1 reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The staff determined that the material with the highest ART at 32 EFPY for Cook 1 was the intermediate shell plate (B4406-3) with 0.15% copper (Cu), 0.49% nickel (Ni), and an initial RT_{ndt} of 40°F.

The licensee has removed four surveillance capsules from DCCNP, Unit 1. The results from capsules T, Y, X, and U in Unit 1 were published in Southwest Research Institutes Reports 02-4770, SwRI-7244-001/1, SwRI 02-6159, and Westinghouse Report WCAP-12483, respectively. All surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the DCCNP, Unit 1 limiting beltline material, the intermediate shell plate (B4406-3), the staff calculated the ART to be 161.3°F at 1/4T (T = reactor vessel beltline thickness) and 130.8°F for 3/4T at 32 EFPY. The staff used a neutron fluence of 1.41E19 n/cm^2 at the inside diameter of the vessel, which reduced to 8.46E18 n/cm^2 at 1/4T and 3.05E18 n/cm^2 at 3/4T. The ART was determined by Section 2 of RG 1.99, Rev. 2, because the most limiting beltline material was in the surveillance capsules.

The licensee used the method in RG 1.99, Rev. 2, to calculate an ART of 171°F at 1/4T and 138°F at 3/4T at 32 EFPY for the same limiting plate. The staff judges that a difference of 9.7°F between the licensee's ART at 1/4T of 171°F and the staff's ART of 161.3°F is acceptable because the licenseee has used a more conservative Chemistry Factor (CF-6) which was presented without explanation in the Capsule U report. Substituting these ARTs 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.A.2 of Appendix G states that when the pressure exceeds 20% of the pre-service 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 60°F from Section IV.A.2 of Appendix G, the staff has determined that the proposed P-T limits satisfy Section IV.A.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. Unirradiated Charpy USE data are available for the beltline plates and the surveillance weld. This surveillance weld, is representative of beltline welds and, therefore, can be used to predict the end of life (EOL) USE of the beltline welds. The staff evaluated the USE issue and determined that both the beltline plates and the beltline welds meet the 50 ft-lb EOL USE requirements.

3.0 SUMMARY

The staff concludes that the proposed P-T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 32 EFPY because the limits conform to the requirements of Appendices G and H of 10 CFR Part 50. The P-T limits also satisfy 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 in the DCCNP, Unit 1 Technical Specifications.

The license demonstrated, through use of limiting analyses, that the proposed LTOPs set point pressure was conservatively selected such that LTOPs will provide the necessary overpressure protection to prevent the revised Appendix G pressure-temperature limits from being exceeded. In addition, the proposed enable temperature has been demonstrated to meet the intent of Branch Position RSB 5-2 of the Standard Review Plan.

4.0 REFERENCES

1. Regulatory Guide 1.99, Radiation Embrittlement of Reactor Vessel Materials, Revision 2, May 1988.

2. NUREG-0800, Standard Review Plan, Section 5.3.2 Pressure-Temperature Limits.

3. November 7, 1977, Letter from J. Tillinghast (IMP) to E. G. Case (USNRC), Subject: Donald C. Cook Nuclear Plant Unit No. 1.

4. December 5, 1988, Letter from M. P. Alexich (IMP) to T. E. Murley (USNRC), Subject: Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations."

5. E. B. Norris, "Reactor Vessel Material Surveillance Program for Donald C. Cook Unit No. 1; Analysis of Capsule T." SwRI Report 02-4770, December 8, 1977.

6. E. B. Norris, "Reactor Vessel Material Surveillance Program for Donald C. Cook Unit No. 1; Analysis of Capsule Y," SwRI-7244-001/1, January 1984.

7. E. B. Norris, "Reactor Vessel Material Surveillance Program for Donald C. Cook Unit No. 1; Analysis of Capsule," SwRI Report 06-6159, June 22, 1981.

8 E. Terek, S. L. Anderson, L. Albertin, and N. K. Ray, "Analysis of Capsule U from American Electric Power Company D. C. Cook Unit 1 Reactor Vessel Radiation Surveillance Program," Westinghouse Report WCAP-12483, January 1990.

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes the requirements with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes in 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 (55 FR 49452). 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.

7.0 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.

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