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

August 9, 1985

Docket No. 50-315

| DISTRIBUTION | |
|---|---|
| Docket File | NRC PDR |
| L PDR | ORB#1 Rdg |
| Gray File | HThompson |
| DWigginton | CParrish |
| OELD | SECY |
| LHarmon | EJordan |
| BGrimes | JPartlow |
| TBarnhart 4 | WJones |
| MVirgilio | ACRS 10 |
| CMiles | RDiggs |
| RBallard | ELantz |
| PRandall | |

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

Dear Mr. Dolan:

The Commission has issued the enclosed Amendment No. 88 to Facility Operating License No. DPR-58 for the Donald C. Cook Nuclear Plant, Unit No. 1. The amendment consists of changes to the Technical Specifications in response to your applications transmitted by letters dated July 18, 1985 and July 19, 1985 and supplemented by letter dated July 3, 1985.

The amendment revises the Technical Specifications to account for heatup and cooldown, and low temperature (cold) overpressure protection through twelve effective full power years of reactor operation.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular bi-weekly Federal Register notice.

Sincerely,

/s/D. Wigginton

David L. Wigginton, Project Manager
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 88 to DPR-58
2. Safety Evaluation

cc: w/enclosures
See next page

| ORB#1:DL | ORB#1:DL | BC-ORB#1:DL | OELD | AD/OR:DL |
|---|---|---|---|---|
| CParrish | DWigginton;ps | SVarga | | GLainas |
| 8/2/85 | 8/7/85 | 8/7/85 | 8/7/85 | 8/9/85 |

Mr. John Dolan
Indiana and Michigan Electric Company

Donald C. Cook Nuclear Plant

cc:
Mr. M. P. Alexich
Vice President
Nuclear Operations
American Electric Power Service Corporation
1 Riverside Plaza
Columbus, Ohio 43215

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

W. G. Smith, Jr., 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
1800 M Street, N.W.
Washington, DC 20036

Mayor, City of Bridgeman
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

The Honorable John E. Grotberg
United States House of Representatives
Washington, DC 20515

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

J. Feinstein
American Electric Power
Service Corporation
1 Riverside Plaza
Columbus, Ohio 43216

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

# INDIANA AND MICHIGAN ELECTRIC COMPANY

## DOCKET NO. 50-315

## DONALD C. COOK NUCLEAR PLANT UNIT NO. 1

## AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 88
License No. DPR-58

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

   A. The applications for amendments by Indiana and Michigan Electric Company (the licensee) dated July 18, 1985 and July 19, 1985, as supplemented by letter dated July 3, 1985, comply 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 applications, 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:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 88 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The change to the Technical Specifications is to be effective within 30 days of issuance.

4. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 9, 1985

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 88 FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Revise Appendix A as follows:

| Remove Pages | Insert Pages |
|---|---|
| 3/4 1-11 | 3/4 1-11 |
| 3/4 4-3 | 3/4 4-3 |
| 3/4 4-3d | 3/4 4-3d |
| 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 1-3 | B 3/4 1-3 |
| B 3/4 4-1 | B 3/4 4-1 |
| B 3/4 4-5* | B 3/4 4-5* |
| 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 4-10 | B 3/4 4-10 |
| B 3/4 4-11 | B 3/4 4-11 |
| B 3/4 5-2 | B 3/4 5-2 |

*Included for convenience. No changes.

## CHARGING PUMP - SHUTDOWN

### LIMITING CONDITION FOR OPERATION

3.1.2.3 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.

APPLICABILITY: MODES 5 and 6.

ACTION:

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

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 170°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 ar not applicable.

### SURVEILLANCE REQUIREMENTS

4.1.2.3.1 The above required charging pump shall be demonstrated OPERABLE at least once per 31 days by:

a. Starting (unless already operating) the pump from the control room,

b. Verifying, that on recirculation flow, the pump develops a discharge pressure of $\geq$ 2390 psig,

c. Verifying pump operation for at least 15 minutes, and

d. Verifying that the pump is aligned to receive electrical power from an OPERABLE emergency bus.

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 170°F.

LIMITING CONDITION FOR OPERATION

3.4.1.3 a. At least two of the coolant loops listed below shall be OPERABLE:

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 one of the above coolant loops shall be in operation.***

APPLICABILITY: MODES 4 and 5

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; be in COLD SHUTDOWN within 20 hours.

b. With no coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

*A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to 170 °F unless 1) the pressurizer water volume is less than 62.00% 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.

ACTION (Continued)

Below P-7:#

a. Startup and Power operation below P-7 may proceed provided at least two reactor coolant loops and associated pumps are in operation.

b. Hot standby, hot shutdown, and cold shutdown operation may proceed provided at least one reactor coolant loop in operation with an associated reactor coolant or residual heat removal pump; however, operation for up to 15 minutes with no pump in operation is permissible to accommodate transition between residual heat removal pump and reactor coolant pump operation.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1 With one reactor coolant loop and associated pump not in operation, at least once per 7 days determine that:

a. The applicable reactor trip system and/or ESF actuation system instrumentation channels specified in the ACTION statements above have been placed in their tripped conditions, and

b. If P-8 interlock setpoint has been reset for 3 loop operation, its setpoint is $\leq$ 76% of RATED THERMAL POWER.

4.4.1.4.2 Within 30 minutes prior to the start of a reactor coolant pump when any RCS cold leg temperature is $\leq$ 170 °F, verify that:

a. The temperature of the secondary water of each steam generator is $\leq$ 50°F above the temperature of each of the RCS cold legs, or

b. The pressurizer water volume is less than 1116 cubic feet, equivalent to less than 62% indicated on the wide range level indicator.

#A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to 170 °F unless 1) the pressurizer water volume is less than 1116 cubic feet (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.

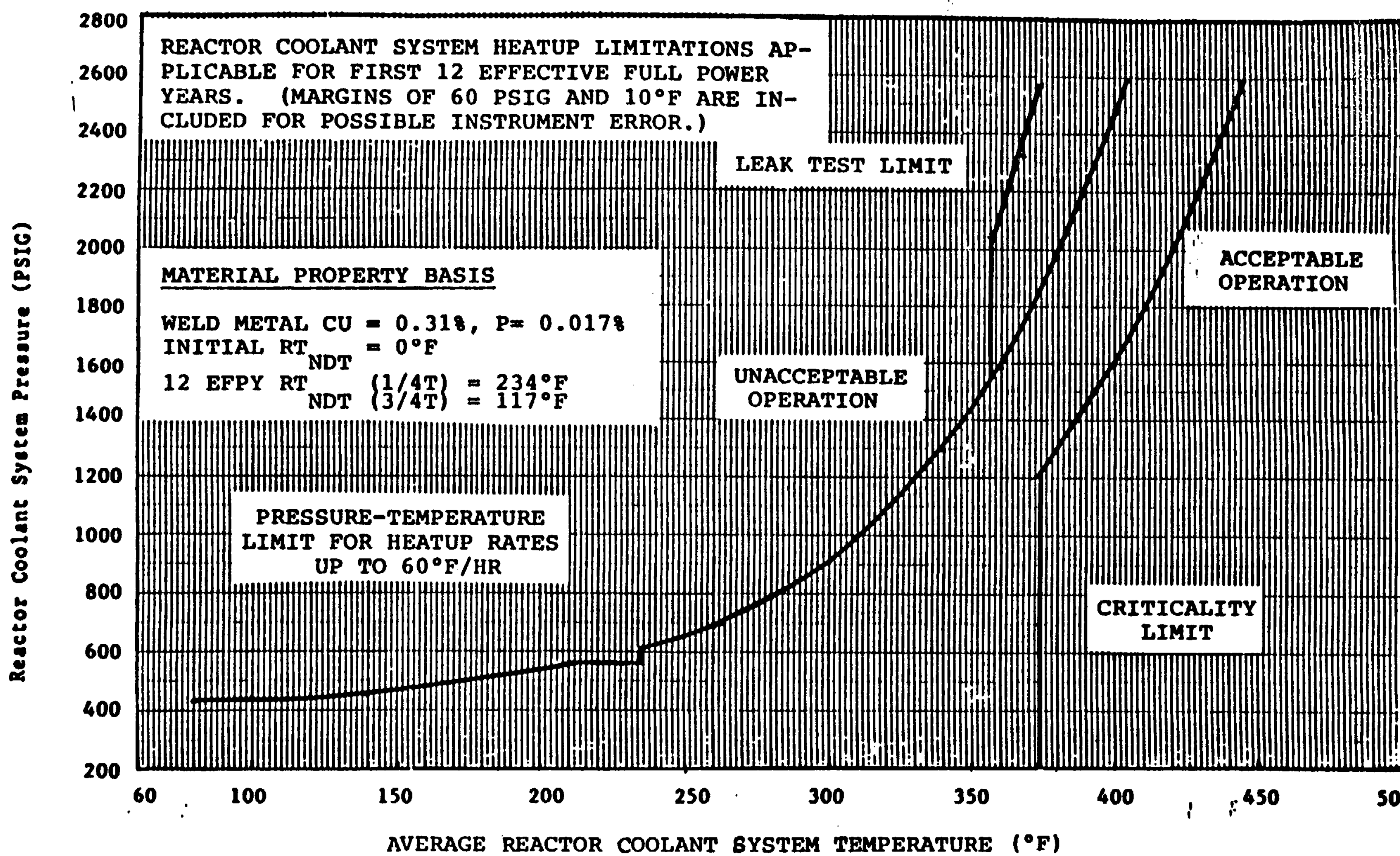


Figure 3.4-2

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

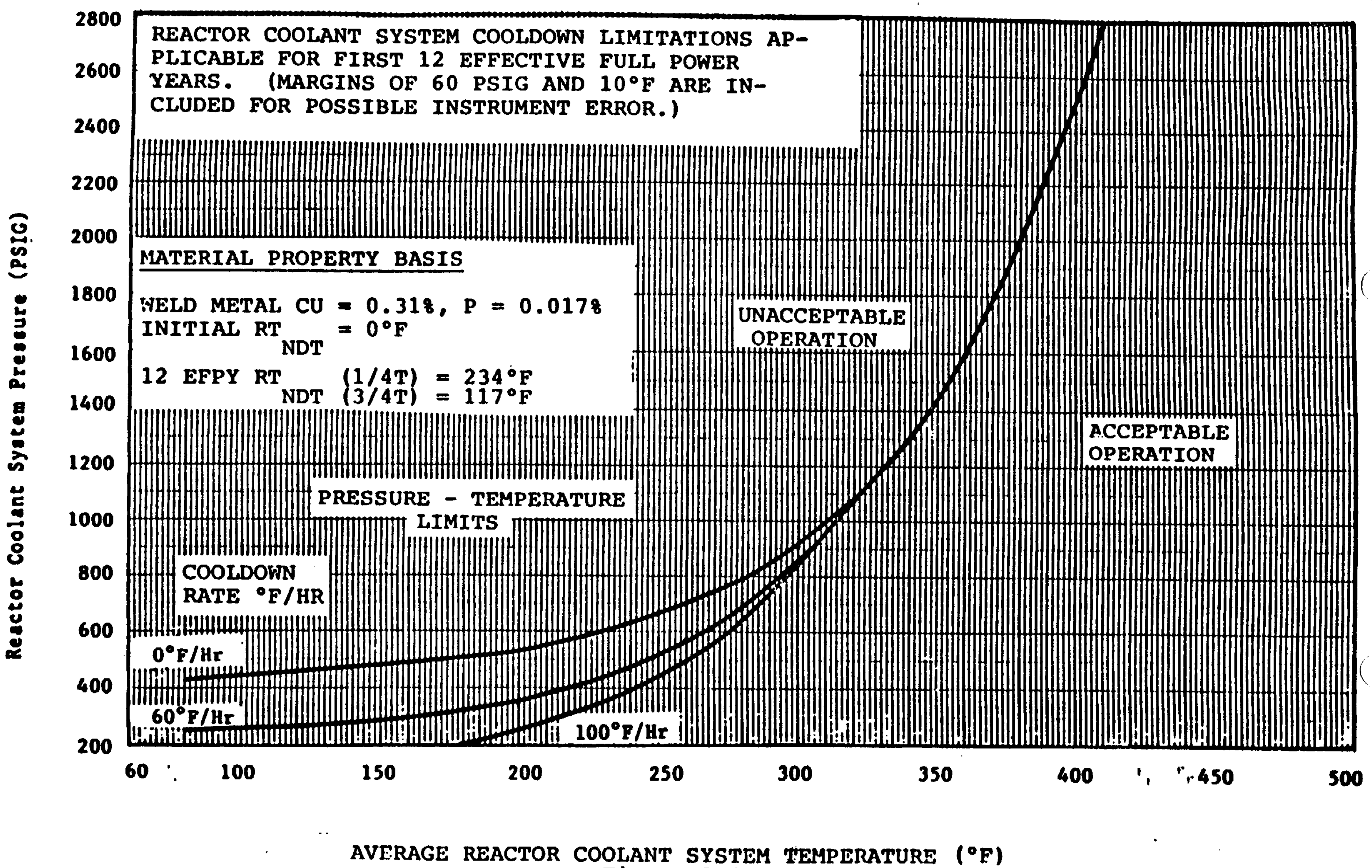


Figure 3.4-3

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 400 psig, or

b. One power operated relief valve (PORV) with a lift setting of less than or equal to 400 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 170°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.

ECCS SUBSYSTEMS - $T_{avg}$ < 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 $T_{avg}$ 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 170°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 170°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 170°F as determined at least once per hour when any RCS cold leg temperature is between 170°F and 200°F.

BORATION SYSTEMS (Continued)

With the RCS average temperature above 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

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 170°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 required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% $\Delta k/k$ after xenon decay and cooldown from 200°F to 140°F. This condition requires either 835 gallons of 20,000 ppm borated water from the boric acid storage tanks of 9690 gallons of 1950 ppm borated water from the refueling water storage tank.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) limit the potential effects of rod ejection accident. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original criteria are met. Misalignment of a rod requires measurement of peaking factors or a restriction in THERMAL POWER; either of these restrictions provide assurance of fuel rod integrity during continued operation. The reactivity worth of a misaligned rod is limited for the remainder of the fuel cycle to prevent exceeding the assumptions used in the accident analysis for a rod ejection accident.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the accident analyses. Measurement with $T_{avg} \geq 541°F$ and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

## 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 1.30 during all normal operations and anticipated transients. With one reactor coolant loop not in operation, THERMAL POWER is restricted to < 51 percent of RATED THERMAL POWER until the Overtemperature $\Delta T$ trip is reset. Either action ensures that the DNBR will be maintained above 1.30. 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 (51 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.

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 170° 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.

### 3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at the valve set point. The relief capacity of a single safety valve is adequate to relieve any over-pressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop , connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

## 3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the D. C. Cook site, such as site boundary location and meteorological conditions, were not considered in this evaluation. The NRC is finalizing site specific criteria which will be used as the basis for the reevaluation of the specific activity limits of this site. This reevaluation may result in higher limits.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity > 1.0 μCi/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1.0 μCi/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.4-1 must be restricted to no more than 10 percent of the unit's yearly operating time since the activity levels allowed by Figure 3.4-1 increase the 2 hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam generator tube rupture.

Reducing $T_{avg}$ to <500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

### 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 12 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 copper and phosphorus content of the material must be predicted. 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 core beltline weld material being the limiting material with a copper and phosphorus content of .31% and .017%.

2. The projected fluence values contained in Table XII of the Southwest Research Institute report, "Reactor Vessel Material Surveillance Program for Donald C. Cook Unit No. 1, Analysis of Capsule Y," dated January 1984.

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

The shift in $RT_{NDT}$ of the vessel material will be established periodically during operation by removing and evaluating reactor vessel material irradiation surveillance specimen dosimetry installed near the inside wall of the reactor vessel. The projected fluence values obtained will be used to calculate the change in $RT_{NDT}$ in accordance with Regulatory Guide 1.99, Revision 1.

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 170°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.

INTENTIONALLY

LEFT

BLANK

INTENTIONALLY

LEFT

BLANK

INTENTIONALLY

LEFT

BLANK

INTENTIONALLY

LEFT

BLANK

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 170°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.

## 3/4.5.4 BORON INJECTION SYSTEM

The OPERABILITY of the boron injection system as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS system cooldown. RCS cooldown can be caused by inadvertent depressurization, a loss-of-coolant accident or a steam line rupture.

The limits on injection tank minimum contained volume and boron concentration ensure that the assumptions used in the steam line break analysis are met.

The OPERABILITY of the redundant heat tracing channels associated with the boron injection system ensure that the solubility of the boron solution will be maintained above the solubility limit of 135°F at 21000 ppm boron.

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

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

# RELATED TO AMENDMENT NO. 88 TO FACILITY OPERATING LICENSE NO. DPR-58

# INDIANA AND MICHIGAN ELECTRIC COMPANY

# DONALD C. COOK NUCLEAR PLANT UNIT NO. 1

# DOCKET NO. 50-315

## Introduction

By letter dated February 14, 1985, the Indiana and Michigan Electric Company submitted an application to amend the Technical Specifications for the Donald C. Cook Nuclear Plant Unit Nos. 1 and 2. The proposed changes would revise the heatup and cooldown curves based on the recent capsule data test results. On June 27, 1985 in Amendment No. 69 to Facility Operating License No. DPR-74, the Unit 2 Technical Specification change was approved. In response to questions by the staff at that time, the licensee again researched the data files to assure the review was based on the best available information on material composition. On July 3, 1985, the licensee submitted the revised material information and on July 18, 1985, the licensee submitted an application for amendment which supersedes the Unit 1 application in the February 14, 1985 letter.

In addition to the proposed changes to the heatup and cooldown curves, the licensee submitted an application on July 19, 1985 to amend the Technical Specifications to revise the low temperature (cold) overpressurization limits. These limits are related to the proposed heatup and cooldown curves.

## Discussion

### 1) Heatup and Cooldown Curves

The materials information submitted by letter of July 3 enclosed a letter from Westinghouse Electric Corporation which gave eight measured values of Cu, Ni and P content from welds made using the same weld wire heat number as the circumferential beltline weld in the Unit-1 vessel. It was recommended that the average values (0.31% Cu, 0.74% Ni, and 0.16% P) be used, and the submittal to the NRC did so. We agree that this is reasonable, because the uncertainty in composition is covered by the margin added to the calculated value of $RT_{NDT}$.

The surveillance weld for Unit-1 was made with one of the weld wire heat numbers used in making the longitudinal welds. Its copper and nickel content were 0.27% and 0.74% respectively. The measured shift reported from Capsule Y was less than that predicted by Regulatory Guide 1.99, Revision 1 but was very consistent with the results from recent surveillance data. Nevertheless, the surveillance result was not used directly, because the girth weld material, which has a higher copper content, is expected to be controlling.

The surveillance report for Capsule Y was used as the source of the fluence value. The reported calculated maximum fluence at the vessel I.D. surface was $3.6 \times 10^{18}$ n/cm$^2$ (E > 1 MeV), corresponding to 4.94 EFPY. There being no announced plans for flux reduction, these numbers were ratioed and multiplied by 12 to obtain an I.D. surface fluence of $8.8 \times 10^{18}$ n/cm$^2$ for the desired period of serviceability of the new P-T limits (12 EFPY). We accept this value as the basis for the calculation of P-T limits.

To calculate $RT_{NDT}$, the proposed revised Technical Specifications reference the surveillance report for Capsule Y, prepared by Southwest Research Institute (SRI), and the methods found acceptable by Regulatory Guide 1.99, Revision 1. This is a satisfactory basis. In our check of the P-T limits, we used both Revision 1 and some more recent information collected to provide a basis for updating Revision 1. We found the two were in agreement within ± 10°F. To clarify the basis for calculating the $RT_{NDT}$, the licensee at our request submitted revised bases pages B 3/4 4-6 thru 4-11. The submittal dated August 1, 1985 contains no new information on the proposed amendment. The revised pages are acceptable.

In conclusion, the staff has used the method of calculating pressure-temperature limits in USNRC Standard Review Plan 5.3.2, NUREG-0800, Rev. 1, July 1981 to evaluate the proposed pressure-temperature limits. The amount of neutron irradiation damage to the limiting beltline material was calculated using the method recommended in Regulatory Guide 1.99, Revision 1. Our conclusion is that the proposed pressure temperature limits meet the safety margins of Appendix G, 10 CFR 50 for twelve (12) EFPY and may be incorporated into the Unit-1 technical specifications.

2) Low Temperature (Cold) Overpressurization Limits

In the July 19, 1985 letter, the licensee stated that the proposed cold overpressurization limits are to conform to the updated heatup and cooldown curves with respect to the temperature vs pressure limits for the reactor coolant system. The licensee has proposed to reduce the Low Temperature Overpressure Protection System temperature to 170°F, and the low pressure settings on the pressurizer power operated relief valves to 400 psig. The former settings were 188°F and 435 psig. Since the allowable pressure on the new isothermal curves at 170°F is over 500 psig, we find these new settings and the proposed Technical Specifications acceptable.

Final Determination - No Significant Hazards Determination

In our review of the heatup and cooldown curves and the related low temperature overpressurization limits, we have determined that the licensee has used acceptable methods and materials information and that the proposed pressure temperature limits meet the safety margins of Appendix G, 10 CFR 50 for twelve (12) effective full power years of reactor operation. The proposed revision reflects conservative values of the Reference Nil-Ductility Transition Temperature ($RT_{NDT}$) as calculated by the method recommended by Regulatory Guide 1.99, Revision 1. Therefore, the proposed amendment does not change the consequences or probabilities of accidents previously evaluated and operation with the new limits assures that the margin of safety is maintained. The changes do not create a new or different kind of accident from any previously evaluated. On this basis, the staff has made a final determination that the proposed heatup and cooldown curves and the low temperature overpressurization limits involve no significant hazards consideration.

## Environmental Consideration

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this 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 this amendment.

## Conclusion

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: August 9, 1985

Principal Contributors:

E. Lantz

P. N. Randall

D. Wigginton