June 9, 1989

Docket No. 50-315

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PD31 Gray File	

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. 126 TO FACILITY OPERATING LICENSE NO. DPR-58 (TAC NO. 71062)

The Commission has issued the enclosed Amendment No. 126 to Facility Operating License No. DPR-58 for the D. C. Cook Nuclear Plant, Unit No. 1. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated October 14, 1988 and supplements dated December 30, 1988, and June 5, 1989.

This amendment revises the TSs to allow operation of future reload cycles of D. C. Cook Unit 1 at reduced pimary coolant system temperature and pressure conditions. The reduced temperature and pressure (RTP) conditions will decrease the steam generator U-tube stress corrosion cracking of the type observed at D. C. Cook Unit 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,

original signed by

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

Enclosures:
1. Amendment No. 126 to DPR-58
2. Safety Evaluation

DF01
1/1

cc w/enclosures:
See next page
DOCUMENT NAME: cook tac 71062
*See previous concurrence

*LA/PD31:DRSP	*PM/PD31:DRSP	*PM/PD31:DRSP	*(A)D/PD31:DRSP	*SRXB
PShuttleworth:sam	JStang	AGody	LYandell	WHodges
5/24/89	5/25/89	5/25/89	5/25/89	5/25/89
*EMTB	*EMEB	*SPLB	*OGC	
CYCheng	TMarsh	JWermiel		
5/25/89	5/25/89	5/25/89	6/6 /89	

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EMTB	EMEB	SPLB	OGC	
CYCheng	TMarsh	JWermiel		
5/25/89	5/25/89	5/25/89	5/ /89	

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

June 9, 1989

Docket No. 50-315

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

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Sincerely,

Lawrence A Yandell for

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

Enclosures:
1. Amendment No. 126 to DPR-58
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

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

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

Township Supervisor
Lake Township Hall
Post Office Box 818
Bridgeman, 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
2300 N Street, N.W.
Washington, DC 20037

Mayor, City of Bridgeman
Post Office Box 366
Bridgeman, 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

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. 126
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 14, 1988 as supplemented December 30, 1988, and June 5, 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-58 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 126, 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 its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Lawrence A. Yandell

Lawrence A. Yandell, 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: June 9, 1989

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 126 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
1-7	1-7
1-10	1-10
2-2	2-2
2-5	2-5
2-7	2-7
2-8	2-8
2-9	2-9
B 2-1(a)	B 2-1(a)
B 2-2	B 2-2
B 2-4	B 2-4
B 2-5	B 2-5
3/4 1-6	3/4 1-6
3/4 2-5	3/4 2-5
3/4 2-7	3/4 2-7
3/4 2-8	3/4 2-8
3/4 2-9	3/4 2-9
3/4 2-14	3/4 2-14
3/4 2-15	3/4 2-15
3/4 3-10	3/4 3-10
3/4 3-24	3/4 3-24
3/4 3-26	3/4 3-26
3/4 4-6	3/4 4-6
3/4 5-1	3/4 5-1
3/4 5-5	3/4 5-5
3/4 5-6	3/4 5-6
3/4 7-5	3/4 7-5
B 3/4 2-1	B 3/4 2-1
B 3/4 6-2	B 3/4 6-2

MEMBER(S) OF THE PUBLIC

1.35 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or its vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

SITE BOUNDARY

1.36 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased or otherwise controlled by the licensee.

UNRESTRICTED AREA

1.37 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY to which access is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials or any area within the site boundary used for residential quarters or industrial, commercial, institutional and/or recreational purposes.

ALLOWABLE POWER LEVEL (APL)

1.38 APL means "allowable power level" which is that power level, less than or equal to 100% RATED THERMAL POWER, at which the plant may be operated to ensure that power distribution limits are satisfied.

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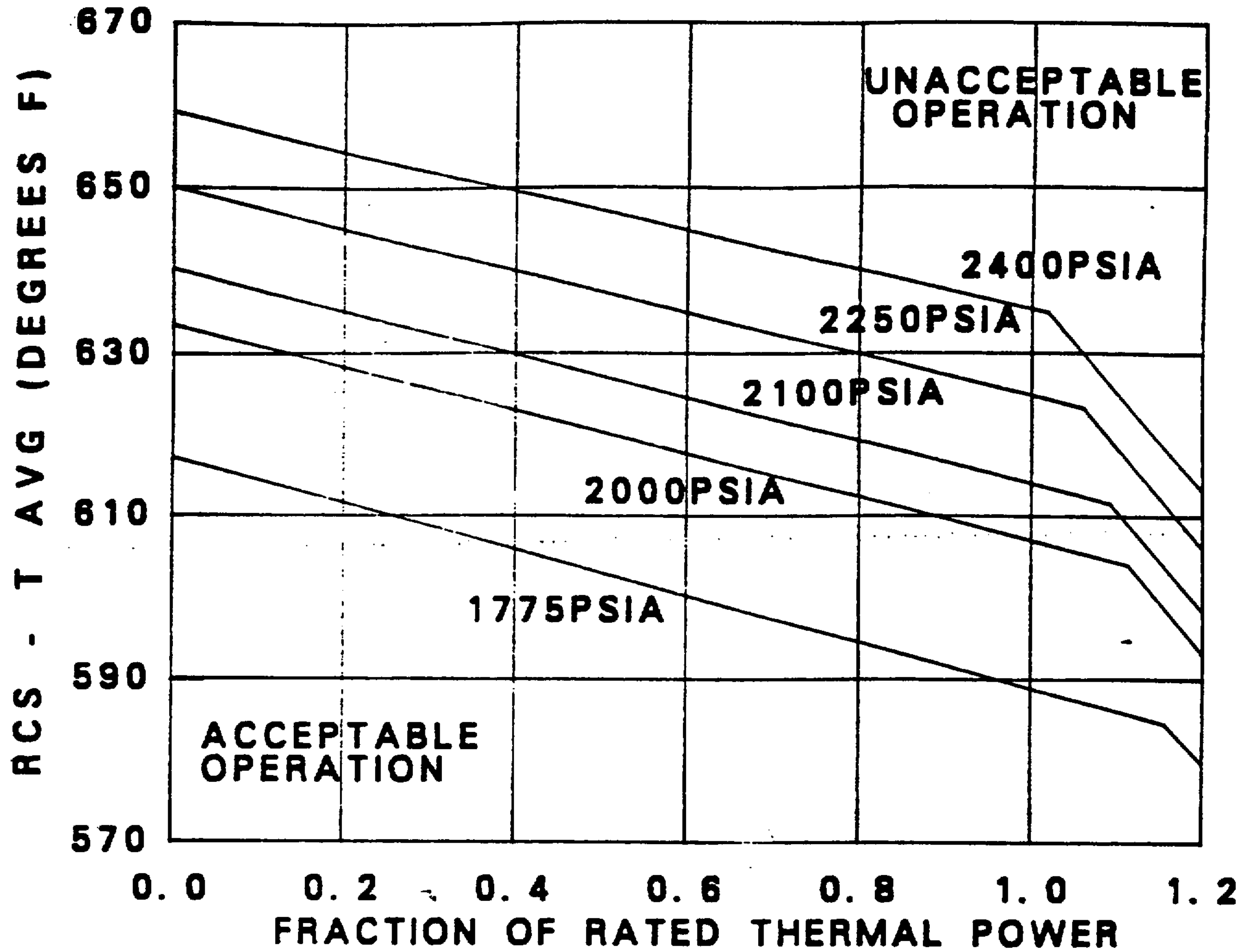

PRESSURE (PSIA)	BREAKPOINTS (FRACTION RATED THERMAL POWER, T AVG IN DEGREES F)
1775	(0. 0, 617. 1), (1. 16, 584. 5), (1. 20, 579. 7)
2000	(0. 0, 633. 5), (1. 11, 603. 9), (1. 20, 593. 1)
2100	(0. 0, 640. 3), (1. 09, 611. 5), (1. 20, 598. 3)
2250	(0. 0, 650. 0), (1. 06, 623. 2), (1. 20, 606. 0)
2400	(0. 0, 659. 0), (1. 02, 634. 8), (1. 20, 613. 0)

Figure 2. 1-1 Reactor Core Safety Limits
Four Loops In Operation

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Range, Neutron Flux	Low Setpoint - $\leq$ 25% of RATED THERMAL POWER High Setpoint $\leq$ 109% of RATED THERMAL POWER	Low Setpoint - $\leq$ 26% of RATED THERMAL POWER High Setpoint - $\leq$ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds
5. Intermediate Range, Neutron Flux	$\leq$ 25% of RATED THERMAL POWER	$\leq$ 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	$\leq 10^5$ counts per second	$\leq 1.3 \times 10^5$ counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 4
9. Pressurizer Pressure--Low	$\geq$ 1875 psig	$\geq$ 1865 psig
10. Pressurizer Pressure--High	$\leq$ 2385 psig	$\leq$ 2395 psig
11. Pressurizer Water Level--High	$\leq$ 92% of Instrument span	$\leq$ 93% of Instrument span
12. Loss of Flow	$\geq$ 90% of design flow per loop*	$\geq$ 89.1% of design flow per loop*

*Design flow is 91,600 gpm per loop.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

Note 1: Overtemperature $\Delta T \leq \Delta T_o \left[K_1 - K_2 \left[\frac{1+\tau_1 S}{1+\tau_2 S}\right](T-T') + K_3(P-P') - f_1(\Delta I)\right]$

where: ΔT_o = Indicated Δ T at RATED THERMAL POWER

T = Average temperature, $^{\circ}F$

T' = Indicated T_{avg} at RATED THERMAL POWER ($\leq 567.8^{\circ}F$)

P = Pressurizer pressure, psig

P' = Indicated RCS nominal operating pressure (2235 psig or 2085 psig)

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation

τ_1, τ_2 = Time constants utilized in the lead-lag controller for T_{avg} τ_1 = 22 secs. τ_2 = 4 secs.

S = Laplace transform operator

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATIONS (Continued)

Operation with 4 Loops

K_1 = 1.32

K_2 = 0.0230

K_3 = 0.00110

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) For $q_t - q_b$ between -37 percent and +2 percent, $f_1(\Delta I)=0$ (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) For each percent that the magnitude of $(q_t - q_b)$ exceeds -37 percent, the ΔT trip setpoint shall be automatically reduced by 0.33 percent of its value at RATED THERMAL POWER.

(iii) For each percent that the magnitude of $(q_t - q_b)$ exceeds +2 percent, the ΔT trip setpoint shall be automatically reduced by 2.17 percent of its value at RATED THERMAL POWER.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_o \left[K_4 - K_5 \left[\frac{\tau_3 S}{1+\tau_3 S}\right] T - K_6 (T - T'') - f_2(\Delta I)\right]$

where:

ΔT_o	=	Indicated ΔT at RATED THERMAL POWER
T	=	Average temperature, ^oF
T''	=	Indicated T_{avg} at RATED THERMAL POWER ($\leq 567.8^oF$)
K_4	=	1.083
K_5	=	$0.0177/^oF$ for increasing average temperature and 0 for decreasing average temperature
K_6	=	0.0015 for $T > T''$; $K_6 = 0$ for $T \leq T''$
$\frac{\tau_3 S}{1+\tau_3 S}$	=	The function generated by the rate lag controller for T_{avg} dynamic compensation
τ_3	=	Time constant utilized in the rate lag controller for T_{avg} $\tau_3 = 10$ secs.
S	=	Laplace transform operator
$f_2(\Delta I)$	=	0

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 3.2 percent ΔT span.

Note 4: The channel's maximum trip point shall not exceed its computed trip point by more than 2.1 percent ΔT span.

4 Loop Operation

Westinghouse Fuel
(15x15 OFA)

(WRB-1 Correlation)

	Typical Cell*	Thimble Cell**
Correlation Limit	1.17	1.17
Design Limit DNBR	1.33	1.32
Safety Analysis Limit DNBR	1.45	1.45

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than the applicable design DNBR limit, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

----

* represents typical fuel rod
** represents fuel rods near guide tube

The curves are based on an enthalpy hot channel factor $F^N_{\Delta H}$, of 1.49 for Westinghouse fuel and a reference cosine axial power shape with a peak of 1.55. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power, based on the expression:

$$F^N_{H} = 1.49 [1 + 0.3 (1-P)]$$

where P is the fraction of RATED THERMAL POWER

Note, do not include a 4% uncertainty value, since this measurement uncertainty has been included in the design DNBR limit values, which are listed in the bases for Section 2.1.1.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion, assuming the axial power imbalance is within the limits of the f_1 (delta I) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature delta T trips will reduce the setpoints to provide protection consistent with the core safety limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

SAFETY LIMITS

BASES

The Power Range Negative Rate Trip provides protection for control rod drop accidents. At high power, a rod drop accident could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate Trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate Trip for those control rod drop accidents for which the DNBR's will be greater than the applicable design limit DNBR value for each fuel type.

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor startup. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The source Range Channels will initiate a reactor trip at about 10^{+5} counts per second, unless manually blocked when P-6 becomes active. The Intermediate Range Channels will initiate a reactor trip at a current level proportional to approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

Overtemperature delta T

The Overtemperature delta T trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the High and Low Pressure reactor trips. This setpoint includes corrections for changes in density and heat capacity of water with temperature and dynamic compensation for piping delays from the core to the loop temperature detectors. The reference average temperature (T') and the reference operating pressure (P') are set equal to the full power indicated Tavg and the nominal RCS operating pressure, respectively, to ensure protection of the core limits and to preserve the actuation time of the Overtemperature delta T trip for the range of full power average temperatures assumed in the safety analyses. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table 2.2-1.

Overpower delta T

The Overpower delta T reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature delta T protection, and provides a backup to the High Neutron Flux trip. The setpoint includes corrections for changes in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The reference average temperature (T") is set equal to the full power indicated Tavg to ensure fuel integrity during overpower conditions for the range of full power average temperatures assumed in the safety analysis. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The High Pressure trip provides protection for a Loss of External Load event. The Low Pressure trip provides protection by tripping the reactor in the event of a loss of reactor coolant pressure.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief through the pressurizer safety valves. The pressurizer high water level trip precludes water relief for the Uncontrolled RCCA Withdrawal at Power event.

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.5 The Reactor Coolant System lowest operating loop temperature (Tavg) shall be $\geq$ 541°F when the reactor is critical.

APPLICABILITY: Modes 1 and 2*#.

ACTION:

With a Reactor Coolant System operating loop temperature (Tavg) < 541°F, restore (Tavg) to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.5 The Reactor Coolant System temperature (Tavg) shall be determined to be $\geq$ 541°F:

a. Within 15 minutes prior to achieving reactor criticality, and

b. A least once per 30 minutes when the reactor is critical and the Reactor Coolant System Tavg is less than 545°F or when the low Tavg alarm is inoperable.

*See Special Test Exception 3.10.3

#With $K_{eff} \geq 1.0$.

HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq \frac{[2.15]}{P}[K(Z)] \qquad P > 0.5$$

$$F_Q(Z) \leq [4.30]\ [K(Z)] \qquad P \leq 0.5$$

$$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

$F_Q(Z)$ is the measured hot channel factor including a 3% manufacturing tolerance uncertainty and a 5% measurement uncertainty.

K(Z) is the function obtained from Figure 3.2-3.

APPLICABILITY: MODE 1

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower delta T Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit.

b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a, above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

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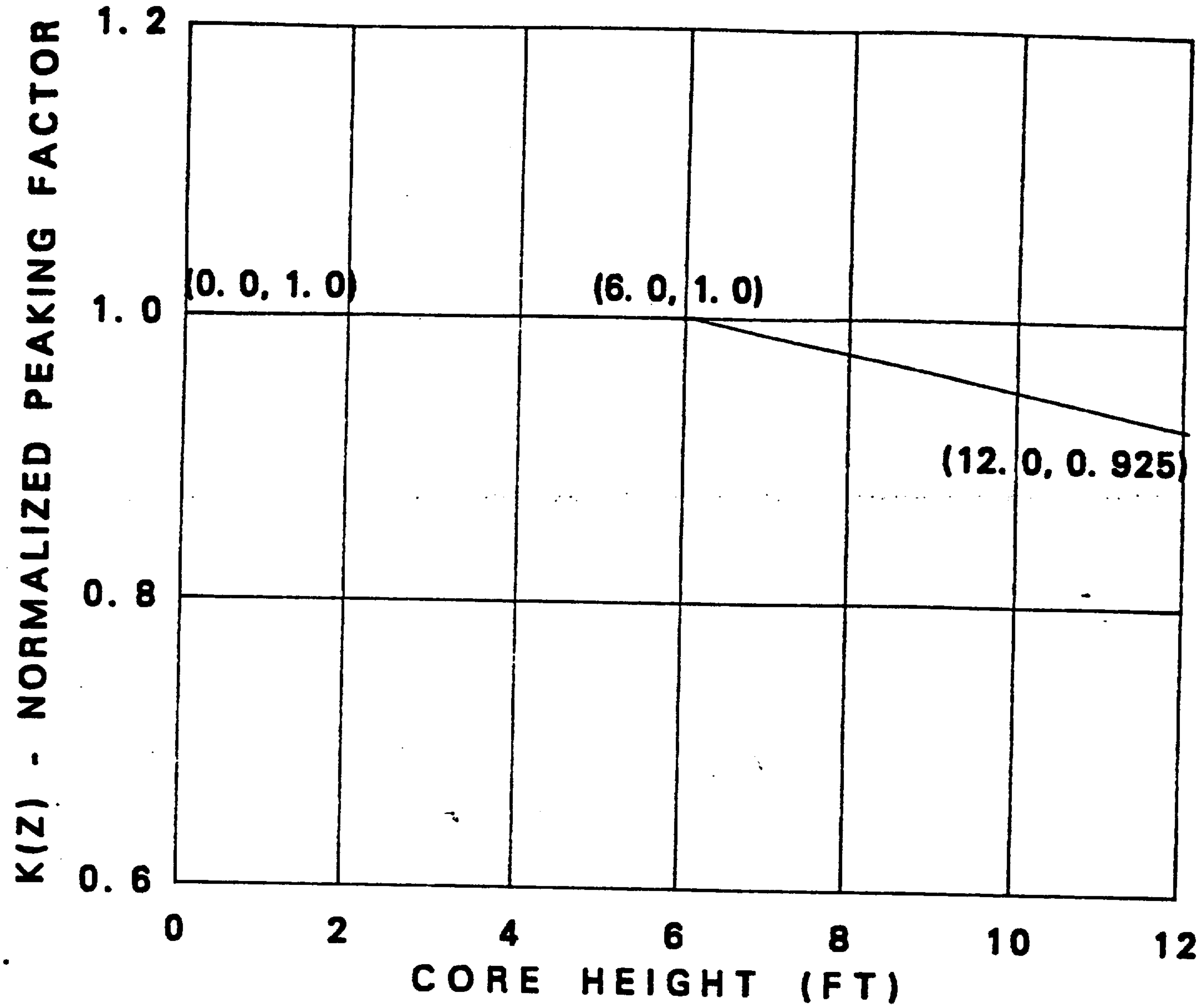

FIGURE 3. 2-3 K(Z) - Normalized F sub Q(z) as a function of Core Height

POWER DISTRIBUTION LIMITS

NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F^N_{\Delta H}$

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship:

$$F^N_{\Delta H} \leq 1.49\ [1 + 0.3\ (1-P)]$$

Where P is the fraction of RATED THERMAL POWER

APPLICABILITY: MODE 1

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours,

b. Demonstrate through in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION may proceed, provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL POWER and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS 4 Loops in Operation at RATED THERMAL POWER
Reactor Coolant System Tavg	$\leq 570.9^{\circ}F^{*}$
Pressurizer Pressure	$\geq$ 2050 psig**
Reactor Coolant System Total Flow Rate	$\geq$ 366,400 gpm***

*Indicated average of at least three OPERABLE instrument loops.

**Limit not applicable during either a THERMAL POWER ramp increase in excess of 5 percent RATED THERMAL POWER per minute or a THERMAL POWER step increase in excess of 10 percent RATED THERMAL POWER.

***Indicated value.

ALLOWABLE POWER LEVEL - APL

LIMITING CONDITION FOR OPERATION

3.2.6 THERMAL POWER shall be less than or equal to ALLOWABLE POWER LEVEL (APL), given by the following relationship:

APL = min over Z of $\frac{2.15\ K(Z)}{F_Q(Z) \times V(Z) \times F_p}$ x 100%, or 100%, whichever is less.

*$F_Q(Z)$ is the measured hot channel factor including a 3% manufacturing tolerance uncertainty and a 5% measurement uncertainty.

*V(Z) is the function defined in the Peaking Factor Limit Report.

*F_p = 1.00 except when successive steady-state power distribution maps indicate an increase in max over Z of $\frac{F_Q(Z)}{K(Z)}$ with exposure. Then either of the penalties, F_p, shall be taken:

F_p = 1.02, or

F_p = 1.00 provided that Surveillance Requirement 4.2.6.2 is satisfied once per 7 Effective Full Power Days until two successive maps indicate that the max over Z of $\frac{F_Q(Z)}{K(Z)}$ is not increasing.

*The above limit is not applicable in the following core regions.

1) Lower core region 0% to 10% inclusive.
2) Upper core region 90% to 100% inclusive.

APPLICABILITY: MODE 1

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	NOT APPLICABLE
2. Power Range, Neutron Flux	
a. High Setpoint	≤ 0.5 seconds*
b. Low Setpoint	≤ 0.5 seconds*
3. Power Range, Neutron Flux, High Positive Rate	NOT APPLICABLE
4. Power Range, Neutron Flux, High Negative Rate	≤ 0.5 seconds*
5. Intermediate Range, Neutron Flux	NOT APPLICABLE
6. Source Range, Neutron Flux	NOT APPLICABLE
7. Overtemperature Delta T	≤ 6.0 seconds*
8. Overpower Delta T	NOT APPLICABLE
9. Pressurizer Pressure--Low	≤ 1.0 seconds
10. Pressurizer Pressure--High	≤ 1.0 seconds
11. Pressurizer Water Level--High	≤ 2.0 seconds

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

TABLE 3.4-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION, TURBINE TRIP, FEEDWATER ISOLATION, AND MOTOR DRIVEN FEEDWATER PUMPS		
a. Manual Initiation	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High	$\leq$ 1.1 psig	$\leq$ 1.2 psig
d. Pressurizer Pressure--Low	$\geq$ 1815 psig	$\geq$ 1805 psig
e. Differential Pressure Between Steam Lines--High	$\leq$ 100 psi	$\leq$ 112 psi
f. Steam Flow in Two Steam Lines--High Coincident with T_{avg}--Low-Low or Steam Line Pressure--Low	$\leq$ 1.42 x 10^6 lbs/hr from 0% load to 20% load. Linear from 1.42 x 10^6 lbs/hr at 20% load to 3.88 x 10^6 lbs/hr at 100% load $T_{avg} \geq$ 541°F $\geq$ 500 psig steam line pressure	$\leq$ 1.56 x 10^6 lbs/hr from 0% load to 20% load. Linear from 1.56 x 10^6 lbs/hr at 20% load to 3.93 x 10^6 lbs/hr at 100% load. $T_{avg} \geq$ 539°F $\geq$ 480 psig steam line pressure

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
2. Containment Radioactivity -- High Train A (VRS-1101, ERS-1301, ERS-1305)	See Table 3.3-6	Not Applicable
3. Containment Radioactivity -- High Train B (VRS-1201, ERS-1401, ERS-1405)	See Table 3.3-6	Not Applicable
4. STEAM LINE ISOLATION		
a. Manual	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure -- High-High	$\leq$ 2.9 psig	$\leq$ 3 psig
d. Steam Flow In Two Steam Lines -- High Coincident with T_{avg} Low-Low or Steam Line Pressure Low	$\leq 1.42 \times 10^6$ lbs/hr from 0% load to 20% load. Linear from 1.42×10^6 lbs/hr at 20% load to 3.88×10^6 lbs/hr at 100% load. $T_{avg} \geq 541^oF$ $\geq$ 500 psig steam line pressure	$\leq 1.56 \times 10^6$ lbs/hr from 0% load to 20% load. Linear from 1.56×10^6 lbs/hr at 20% load to 3.93×10^6 lbs/hr at 100% load. $T_{avg} \geq 539^oF$ $\geq$ 480 psig steam line pressure
5. TURBINE TRIP AND FEEDWATER ISOLATION		
a. Steam Generator Water Level -- High-High	$\leq$ 67% of narrow-range instrument span each steam generator	$\leq$ 68% of narrow-range instrument span each steam generator

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with a water volume less than or equal to 92% of span and at least 150 kW of pressurizer heaters.

APPLICABILITY: MODES 1,2, and 3.

ACTION:

With the pressurizer inoperable due to an inoperable emergency power supply to the pressurizer heaters either restore the inoperable emergency power supply within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours. With the pressurizer otherwise inoperable, be in at least HOT SHUTDOWN with the reactor trip breakers open within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.4.1 The pressurizer water volume shall be determined to be within its limits at least once per 12 hours.

4.4.4.2 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by transferring power from the normal to the emergency power supply and energizing the required capacity of heaters.

ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between 921 and 971 cubic feet,

c. A boron concentration of between 2400 ppm and 2600 ppm, and

d. A nitrogen cover-pressure of between 585 and 658 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within one hour or be in HOT SHUTDOWN within the next 8 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in HOT STANDBY within one hour and be in HOT SHUTDOWN within the next 8 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying the water level and nitrogen cover-pressure in the tanks, and

2. Verifying that each accumulator isolation valve is open.

* Pressurizer Pressure above 1000 psig.

d. At least once per 18 months by:*

1. Verifying automatic isolation and interlock action of the RHR system from the Reactor Coolant System when the Reactor Coolant System pressure is above 600 psig.

2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or abnormal corrosion.

e. At least once per 18 months, during shutdown, by:*

1. Verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection test signal.

2. Verifying that each of the following pumps start automatically upon receipt of a safety injection test signal:

   a) Centrifugal charging pump

   b) Safety injection pump

   c) Residual heat removal pump

f. By verifying that each of the following pumps develops the indicated discharge pressure on recirculation flow when tested pursuant to Specification 4.0.5 at least once per 31 days on a STAGGERED TEST BASIS.

1.	Centrifugal charging pump	$\geq$ 2405 psig
2.	Safety Injection pump	$\geq$ 1345 psig
3.	Residual heat removal pump	$\geq$ 165 psig

g. By verifying the correct position of each mechanical stop for the following Emergency Core Cooling System throttle valves:

1. Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE.

*The provisions of Specification 4.0.6 are applicable.

SURVEILLANCE REQUIREMENTS (Continued)

2. At least once per 18 months.

Boron Injection Throttle Valves		Safety Injection Throttle Valves	
	Valve Number		Valve Number
1.	1-SI-141 L1	1.	1-SI-121 N
2.	1-SI-141 L2	2.	1-SI-121 S
3.	1-SI-141 L3		
4.	1-SI-141 L4		

h. By performing a flow balance test during shutdown following completion of modifications to the ECCS subsystem that alter the subsystem flow characteristics and verifying the following flow rates:

Boron Injection System Single Pump*	Safety Injection System Single Pump**
Loop 1 Boron Injection Flow 117.5 gpm	Loop 1 and 4 Cold Leg Flow $\geq$ 300 gpm
Loop 2 Boron Injection Flow 117.5 gpm	Loop 2 and 3 Cold Leg Flow $\geq$ 300 gpm
Loop 3 Boron Injection Flow 117.5 gpm	**Combined Loop 1,2,3 and 4 Cold Leg Flow (single pump) less than or equal to 640 gpm. Total SIS (single pump) flow, including miniflow, shall not exceed 700 gpm.
Loop 4 Boron Injection Flow 117.5 gpm	

*The flow rate in each Boron Injection (BI) line should be adjusted to provide 117.5 gpm (nominal) flow in each loop. Under these conditions there is zero miniflow and 80 gpm plus or minus 5 gpm simulated RCP seal injection line flow.

The actual flow in each BI line may deviate from the nominal so long as:

a) the difference between the highest and lowest flow is 25 gpm or less.

b) the total flow to the four branch lines does not exceed 470 gpm.

c) the minimum flow (total flow) through the three most conservative (lowest flow) branch lines must not be less than 300 gpm.

d) The charging pump discharge resistance ($2.31xPd/Qd^2$) must not be less than 4.73E-3 ft/gpm^2 and must not be greater than 9.27E-3 ft/gpm^2. (Pd is the pump discharge pressure at runout; Qd is the total pump flow rate.)

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two feedwater pumps, each capable of being powered from separate emergency busses, and

b. One feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two auxiliary feedwater pumps inoperable, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying that each motor driven pump develops an equivalent discharge pressure of $\geq$ 1375 psig at 60°F in recirculation flow.

2. Verifying that the steam turbine driven pump develops an equivalent discharge pressure of $\geq$ 1285 psig at 60°F and at a flow of $\geq$ 700 gpm when the secondary steam supply pressure is greater than 310 psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core greater than or equal to the safety limit DNBR during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of hot channel factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

Target flux difference is determined at equilibrium xenon conditions. The full length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady state operation at high power levels. The value of the target flux difference obtained under these condtions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that 1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 8 psig and 2) the containment peak pressure does not exceed the design pressure of 12 psig during LOCA conditions.

The maximum peak pressure resulting from a LOCA event is calculated to be 11.89 psig, which includes 0.3 psig for initial positive containment pressure.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that 1) the containment air mass is limited to an initial mass sufficiently low to prevent exceeding the design pressure during LOCA conditions and 2) the ambient air temperature does not exceed that temperature allowable for the continuous duty rating specified for equipment and instrumentation located within containment.

The containment pressure transient is sensitive to the initially contained air mass during a LOCA. The contained air mass increases with decreasing temperature. The lower temperature limit of 60°F will limit the peak pressure to 11.89 psig which is less than the containment design pressure of 12 psig. The upper temperature limit influences the peak accident temperature slightly during a LOCA; however, this limit is based primarily upon equipment protection and anticipated operating conditions. Both the upper and lower temperature limits are consistent with the parameters used in the accident analyses.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment steel vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that (1) the steel liner remains leak tight and (2) the concrete surrounding the steel liner remains capable of providing external missile protection for the steel liner and radiation shielding in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.126 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 14, 1988, as supplemented December 30, 1988, and June 5, 1989,the Indiana Michigan Power Company (the licensee) requested an amendment to the Technical Specifications (TSs) appended to Facility Operating License No. DPR-58 for the Donald C. Cook Nuclear Plant, Unit No. 1. The proposed amendment would permit the operation of future reload cycles of Unit 1 at reduced primary system temperature and pressure conditions. The reduced temperature and pressure (RTP) conditions will decrease the steam generator U-tube stress corrosion cracking of the type observed at the D. C. Cook Nuclear Plant, Unit 2. The licensee's contractor (Westinghouse) has determined that this RTP program should more than double the time to reach a given level of steam generator U-tube corrosion in comparison to the original temperatures and pressure.

D. C. Cook, Unit 1 is presently licensed to operate at 3250 MWt, which is rated thermal power defined by Definition 1.3 of the Technical Specifications. Some transient and accident analyses are performed at a higher power level to position Unit 1 for a potential power uprating. However, not all of the analyses have been performed at this higher power level. The small break loss-of-coolant accident (LOCA) analysis was, for example performed at a power of level of 3250 MWt with the high head safety injection cross-tie valve shut and at 3588 MWt for all other analyzed plant conditions. The staff's review of the RTP program for Unit 1 did not consider any issues related to a future power uprating.

The licensee performed analyses and evaluations to support the RTP program for D. C. Cook, Unit 1. The licensee's efforts addressed full rated thermal power operation (3250 MWt) with a range of vessel average temperature between 547°F and 576.3°F. Two discrete values of the pressure, 2100 psia and 2250 psia, were used in the analyses and evaluations. The analyses and evaluations support a maximum average tube plugging level of 10%, with a peak steam generator tube plugging level of 15%. The licensee will select the desired operating temperature and the pressure on a cycle-by-cycle basis.

The licensee performed the safety analyses and evaluations at conservatively high power levels and high primary system temperatures in order to position both of the D. C. Cook units for future power uprating and in order to support potential future operation of Unit 2 at reduced temperatures and pressure. The potential uprated power for Unit 1 that is partially supported by this analysis and evaluation is 3425 MWt, which corresponds to a reactor power level of 3413 MWt. The design power capability parameters are given in Table 2.1-1 of Reference 2.

2.0 EVALUATION

2.1 NUCLEAR STEAM SUPPLY SYSTEM (NSSS)

2.1.1 Large and Small Break LOCA Analyses

The licensee performed a large break LOCA analysis using the 1981 version of the Westinghouse ECCS Evaluation Model, which uses the BASH computer code.

The analysis assumptions include a total peaking factor, F_Q, of 2.15, a hot channel enthalpy rise factor, F-delta H, of 1.55, 10% safety injection flow degradation, a reactor power level of 3413 MWt, and 15% uniform steam generator tube plugging level. A range of hot-leg temperatures of 580.7°F to 611.2°F and a range of cold-leg temperatures of 513.3°F to 546.2°F, consistent with the temperature range of the RTP program, were considered in the analysis. In the analysis, the reactor coolant system pressure was varied to justify plant operation at either 2100 psia or 2250 psia. A large-break LOCA analysis was also performed with the RHR cross-tie valve closed. For this case, a reduced core power of 3250 MWt was used to compensate for the reduction in safety injection flow caused by the closed RHR cross-tie valve. For those limiting pressure and temperature conditions which produced the largest peak clad temperature, a full break spectrum of discharge coefficients was performed.

The limiting break size was determined to be a cold-leg guillotine break with a discharge coefficient, C_d, of 0.6, a hot-leg temperature of 611.2°F and a primary system pressure of 2250 psia, assuming maximum safety injection flow. The peak clad temperature was calculated to be 2180.5°F. Based on these results, the requirements of 10 CFR 50.46 have been met for the Unit 1 large-break LOCA analysis.

The licensee performed a small-break LOCA analysis using the Westinghouse small-break ECCS Evaluation Model, which uses the NOTRUMP code. The analysis assumptions included a total peaking factor of 2.32, a hot channel enthalpy rise factor of 1.55, safety injection flow rates based on pump performance curves degraded 10% below design head and including the effect of closure of the high head safety injection cross-tie valve, and a uniform 15% steam generator tube plugging level. The analysis was performed at a core power level of 3250 MWt, a range of operating core average temperatures of 547°F to 581.3°F, and reactor pressure of either 2100 psia or 2250 psia. All other plant conditions were analyzed at a power of 3588 MWt. The licensee analyzed a spectrum of cold-leg breaks at the limiting reactor coolant system temperature and pressure conditions. The limiting break size from this analysis was then analyzed at other temperature and pressure points of the operating range. The limiting case was determined to be a three-inch diameter cold-leg break at a pressure of 2100 psia and at a core average temperature of 547°F. This limiting break resulted in a peak clad temperature of 2122°F. Based on these results, the requirements of 10 CFR 50.46 have been met for the Unit 1 small-break LOCA analysis.

The licensee reviewed the effect of the RTP program on the post-LOCA hot-leg recirculation time to prevent boron precipitation. This time is affected by power level and various systems' water volumes and boron concentrations. Because these systems' water volumes and boron concentrations are not affected by the RTP program, there is no effect on the post-LOCA hot-leg switchover time.

The licensee reviewed the effect of the RTP program on the post-LOCA hydrogen generation rates. The assumption of 120°F maximum normal operations containment temperature bounds, for the analysis of record, the effect of the primary system temperature changes of the RTP program on the post-LOCA hydrogen generation rates.

2.1.2 Non-LOCA Transients and Accidents

The licensee has evaluated the impact of the RTP program on the non-LOCA events presented in Chapter 14 of the D. C. Cook, Unit 1 FSAR. The approved reload core design methodology and design codes were used. The evaluations were performed to support the operation of Unit 1 at a core power of 3250 MWt over a vessel average temperature range between 547°F and 576.3°F at a primary system pressure of either 2100 psia or 2250 psia. The evaluation assumes a steam generator tube plugging level of 10%, with a peak steam generator tube plugging level of 15%. The non-LOCA safety evaluation supports the parameters of the RTP program with the exceptions of the steamline break mass and energy releases outside containment, which were evaluated at a full power vessel average temperature no greater than the current D. C. Cook Unit 1 full power average temperature, T_{avg}, of 567.8°F.

The evaluation performed by the licensee also considered the parameters for a potential uprating of Unit 1 to reactor core power level of 3413 MWt, with a vessel average temperature range between 547°F and 578.7°F at a primary system pressure of either 2100 psia or 2250 psia. The steam generator tube plugging level is assumed to be the same as for the RTP program. Even though the non-LOCA evaluation may have been performed for the uprated core power and its associated parameters, the staff's review of this license amendment does not address a D. C. Cook Unit 1 power uprating.

The licensee revised certain reactor trip and engineered safeguards features (ESF) setpoints to provide adequate operating margins for the RTP operating conditions. Revised reactor trip setpoints were incorporated in the overtemperature-delta T (OTDT) and overpower-delta T (OPDT) trip functions. The revised ESF setpoints affects the low steamline pressure value of the high-high steamline flow coincident with a low steamline pressure actuation logic. The new OPDT and OTDT reactor trip setpoints were developed by the licensee for a new set of core thermal safety limits for the RTP program at a reactor core power level of 3413 MWt. The approved setpoint methodology of Reference 3 was used. For those events analyzed with the approved Improved Thermal Design Procedure (ITDP), Reference 4, a safety-limit value of 1.45 was used for the Departure from Nucleate Boiling Ratio (DNBR). This is conservative compared to the design DNBR value of 1.32 for a thimble cell and 1.33 for a typical cell required to meet the DNB design basis.

In the safety analysis for D. C. Cook, Unit 1, the licensee assumed the high pressurizer water level trip setpoint of 100% (nominal reactor setpoint). Furthermore, the reference average temperature used in the OPDT and OTDT trip setpoint equations are rescaled to the full power average temperature each time the cycle average temperature is changed. Similarly, the appropriate value of primary system pressure of either 2100 or 2250 psia was used in the two trip setpoint equations. For the revised ESF setpoint of the high-high steamline flow coincident with low steamline pressure, the low steamline pressure setpoint was lowered from 600 psig to 500 psig to accommodate the range of conditions of the RTP program and a potential power uprating.

2.1.3 Steamline Break Mass/Energy Releases

The current mass and energy releases for the inside containment analysis is based on analyses performed for Cook Unit 2, which are also applicable to Cook Unit 1. Data are represented in Chapter 14 of the FSAR for Unit 2 at power levels of 0, 30, 70, and 100% power. For the "at power" analyses, the initial primary system temperature and secondary steam pressures of the RTP program are lower than those in the Unit 2 FSAR analyses. The mass blowdown rate is dependent on steam pressure and since the steam pressure will be less than the current analyses, the initial mass blowdown rate will be lower. The lower steamline pressure setpoint (500 psig) of the ESF actuation signal does not significantly impact the analysis because the lead-lag compensation results in a steamline pressure signal which anticipates the rapid decrease in pressure caused by a steamline break. Based on these considerations, the licensee concludes that the RTP program will result in a lower integrated energy release into containment and that the data used in the Unit 2 FSAR remains bounding.

A study was performed for Unit 1 of the mass and energy release outside containment to address equipment qualification issues (Ref. 5). Cases at 70% and 100% power were analyzed. The analysis presented in Reference 5 assumed the full power vessel average temperature to be 567.8°F. Any reduction in full power T_{avg} from the analyzed T_{avg} and the associated reduction in initial steam pressure will result in less limiting releases. The low steamline pressure value assumed in the analysis supports the reduced value of the setpoint to 500 psig. The increased level of steam generator tube plugging is acceptable because the analysis assumed better heat transfer characteristics. The licensee concludes that the current mass and energy release analysis is acceptable for the RTP program as long as the full power T_{avg} is equal to or less than 567.8°F.

2.1.4 Startup of an Inactive Loop

The licensee evaluated the startup of an inactive loop event. This event cannot occur above the P-7 permissive setpoint of 10% power as restricted by the Technical Specifications. The parameters assumed in the FSAR analysis for three-pump operation at 10% power remain bounding for the parameters for 10% power condition. The licensee concludes, therefore, that the conclusions presented in the FSAR remain valid.

2.1.5 Uncontrolled Rod Bank Withdrawal from a Subcritical Condition

The uncontrolled rod bank withdrawal from a subcritical condition transient causes a power excursion. This power excursion is terminated, after a fast power rise, by the negative Doppler reactivity coefficient of the fuel, and a reactor trip on source, intermediate, or power range flux instrumentation. The power excursion results in a heatup of the moderator/coolant and the fuel. The analysis used a reactivity insertion rate of 75 pcm (note that one pcm is equal to a reactivity of 10^{-5} delta K/K). This reactivity insertion rate is greater than for the simultaneous withdrawal of the two sequential control banks having the greatest combined worth at the maximum speed of 45 inches/minute. The neutron flux overshoots the nominal full power value; however, the peak heat flux is much less than the full power nominal value because of the inherent thermal lag of the fuel. The analysis, with the reduced system pressure of 2100 psia, yields the minimum value of DNBR. The analysis is performed using the Standard Thermal Design Procedure (STDP). The W-3 DNB correlation was issued to evaluate DNBR in the span between the lower non-mixing vane grid and the first mixing vane grid. The WRB-1 DNB correlation is applied to the remainder of the fuel assembly. From the analysis performed, the licensee concludes that the DNB design bases are met for all regions of the core, and therefore, the conclusions in the FSAR remain applicable for a reduction in nominal system pressure to 2100 psia.

2.1.6 Uncontrolled Control Rod Assembly Bank Withdrawal at Power

The uncontrolled rod bank withdrawal from a power condition transient leads to a power increase. The transient results in an increase in the core heat flux and an increase in the reactor moderator/coolant temperature. The reduction in pressure for the RTP program is non-conservative with respect to DNB. In addition, a revised Overtemperature Delta-T setpoint equation is being assumed in the Cook Unit 1 analyses. The Power Range High Neutron Flux and Overtemperature Delta-T reactor trips provide the primary protection against DNB. Both minimum and maximum reactivity cases were analyzed over a range of reactivity insertion rates. The licensee provided quantitative results for the maximum reactivity feedback case for power levels of 10%, 60%, and 100% power for a range of reactivity insertion rates. The results indicate that the DNBR limit is met for all the cases.

The licensee examined a number of cases associated with the pressurizer water volume transient caused by an uncontrolled control rod assembly bank withdrawal-at-power event. It was determined that credit for high pressurizer water level reactor trip was required to prevent the pressurizer from filling. The licensee assumed a value of 100% narrow range span (NRS) for the high pressurizer water level reactor trip setpoint. A time delay of 2 seconds was assumed for trip actuation until rod motion becomes adequate to terminate the transient.

Thus the high neutron flux and overtemperature-delta T reactor trips provide adequate protection over the range of possible reactivity insertion rates in that the minimum value of DNBR remains above the safety-limit DNBR value. In addition, the high pressurizer water level reactor trip prevents the pressurizer from filling.

2.1.7 Rod Cluster Assembly Misalignment

The rod cluster control assembly misalignment events consist of three separate events: (1) a dropped control rod, (2) a dropped control bank, and (3) a statically misaligned control rod. These events were reanalyzed because the reduction in pressure for the RTP program is nonconservative with respect to the DNB transient. A dropped control rod or control bank may be detected in the following manner: (1) by a sudden drop in the core power as seen by the nuclear instrumentation system; (2) by an asymmetric power distribution as seen by the excore neutron detectors or the core exit thermocouples; (3) by rod bottom signal; (4) by the rod position deviation monitor; and (5) by rod position indicators. A misaligned control rod may be detected in the following manner; (1) by an asymmetric power distribution as seen by the excore neutron detectors or the core exit thermocouples; (2) by the rod position deviation monitor; and (3) by rod position indicators. The resolution of the rod position indicator channel is ±5 percent or ±12 steps (±7.5 inches). Deviation of any control rod from its group by twice this distance (±24 steps or ±15 inches) will not cause power distribution worse than the design limits. The rod position deviation monitor provides an alarm before a rod deviation can exceed ± 24 steps or ± 15 inches.

The dropped rod event was analyzed using an approved methodology (Ref. 6). A dropped rod or rods from the same group will result in a negative reactivity insertion which may be detected by the negative neutron flux rate trip circuitry. If detected, a reactor trip occurs in about 2.5 seconds. For those dropped rod events for which a reactor trip occurs, the core is not adversely impacted because the rapid decrease in reactor power will reach an equilibrium value dependent on the reactivity feedback or control bank withdrawal (if in automatic control). The limiting case for this class of events is the case with the reactor in automatic control. For this case a power overshoot occurs before an equilibrium power condition is reached. The licensee states that, using the methodology of Reference 6, all analyzed cases result in DNBR values which are within the safety-limit DNBR value.

The licensee states that a dropped rod bank results in a reactivity insertion of at least 500 pcm. This will be detected by the negative neutron flux rate trip circuitry and cause a reactor trip within about 2.5 seconds of the initial motion of the rod bank. Power decreases rapidly and there is, therefore, no adverse impact on the reactor core.

The most severe misalignment cases, with respect to DNBR, are those in which one control rod is fully inserted or where control bank "D" is fully inserted but with one control rod fully withdrawn. Multiple alarms alert the operator before adverse conditions are reached. The control bank can be inserted to its insertion limit with any control rod fully withdrawn without DNBR falling below the safety-limit DNBR value, as shown by analysis. An evaluation performed by the licensee indicates that control rod banks other than the control bank would give less severe results. For the case with one rod fully inserted, DNBR remains above the safety-limit DNBR value. For all cases following identification of a control rod misalignment, the operator is required to perform actions in accordance with plant Technical Specifications and procedures.

2.1.8 <u>Chemical and Volume Control System Malfunction</u>

The boron dilution event was analyzed by the licensee for startup and power operation. The analysis is performed to show that sufficient time is available to the operator to determine the cause of the dilution event and take corrective action before the shutdown margin is lost. The licensee reports that 45 minutes is available for Mode 1 (power operation) and 68 minutes for Modes 2 or 3 (startup or hot standby conditions) (Ref. 7).

2.1.9 <u>Loss of Reactor Coolant Flow</u>

The loss-of-flow transient causes the reactor power to increase until the reactor trips on either a low-flow trip signal or reactor coolant pump power supply undervoltage signal. The reactor power increase causes a reactor moderator/coolant temperature increase. This initial coolant temperature increase causes a positive reactivity insertion because of the positive moderator temperature coefficient. The licensee analyzed both a partial loss-of-flow (loss of one pump with four coolant loops in operation) transient and a complete loss-of-flow transient (loss of four pumps with four coolant loops in operation). For the partial loss-of-flow transient, the reactor is assumed to be tripped on a low-flow signal. For a complete loss-of-flow transient, the reactor is assumed to be tripped on a pump undervoltage signal. For either event, the average and hot channel heat fluxes do not increase significantly above their initial values and the DNBR remains above the safety-limit DNBR value.

2.1.10 Locked Rotor Accident

The locked rotor accident causes a rapid reduction in the fluid flow through the affected loop. The reactor trips on a low-flow signal which rapidly reduces the neutron flux upon control rod insertion. Control rod motion starts 1 second after the flow in the affected loop reaches 87% of its nominal value. The licensee evaluated this accident assuming that offsite power is available. No credit is taken for the pressure-reducing effect of the pressurizer relief valves, pressurizer spray, steam dump, or controlled feedwater flow after reactor trip. The licensee performed an analysis to determine the DNB transient and to demonstrate that the peak system pressure and the peak clad temperature remain below limit values. The peak reactor coolant system pressure of 2588 psia reached during the transient is less than that which would cause stresses to exceed the faulted conditions stress limits. The peak clad temperature reached is 1959°F. Less than 4.5% of the fuel rods in the most limiting fuel assembly reach values of DNBR less than the safety-limit DNBR value. These results indicate that the RTP program assumptions give acceptable consequences for the locked rotor accident.

2.1.11 Loss of External Electrical Load

The loss-of-external-electrical-load event was analyzed by the licensee to show the adequacy of pressure-relieving devices and to demonstrate core protection. This reanalysis was necessary because of changes in reactor pressure and temperature conditions for the RTP program and because of changes to the Overtemperature-Delta T reactor trip setpoint equation. Maximum and minimum reactivity feedback cases were examined, with the case analyzed with and without credit for pressurizer sprays and power-operated relief valves. For the minimum reactivity feedback case with pressurizer pressure control, the reactor trips on a high pressurizer pressure signal. For the maximum reactivity feedback case with pressurizer pressure control, the reactor trips on a low-low steam generator water level signal. For the minimum reactivity feedback case without pressurizer pressure control, the reactor trips on a high pressurizer pressure signal. For all four cases, the minimum value of DNBR remains well above the safety-limit DNBR value and the Overtemperature-Delta T setpoint was not reached. The analysis confirms that the conclusions of the FSAR remain valid for this event for the RTP program.

2.1.12 Loss of Normal Feedwater Flow

The loss-of-normal-feedwater-flow event was analyzed by the licensee to show that the auxiliary feedwater system is capable of removing the stored and decay heat, thus preventing overpressurization of the reactor coolant system or uncovering the core, and returning the plant to a safe condition. The reanalysis was based on a positive moderator temperature coefficient. A conservative decay heat model based on the ANSI/ANS-5.1-1979 decay heat standard (Ref. 8) was used. Pressurizer power operated relief valves and the maximum pressurizer spray flow rate were assumed to be available since a lower pressure results in a greater system expansion. The initial pressurizer water level was assumed to be at the maximum nominal setpoint of 62% narrow range span. Reactor trip occurred when the low-low steam generator water level trip setpoint was reached. The results of the analysis show that a loss of normal feedwater does not adversely affect the reactor core, the reactor coolant system, or the steam system, and that the auxiliary feedwater system is sufficient to prevent water relief through the pressurizer relief or safety valves. The pressurizer does not fill and, therefore, the conclusions of the FSAR remain valid for this event, including RTP conditions.

2.1.13 Excessive Heat Removal Due to Feedwater System Malfunctions

The excessive-heat-removal event due to feedwater system malfunction was analyzed by the licensee to demonstrate core protection. This analysis was necessary because of changes in reactor core temperatures and pressure for the RTP program and because of changes to the OTDT and OPDT trip setpoints. This event is an excessive-feedwater-addition event caused by a control system malfunction or an operator error which allows a feedwater control valve to open fully. The licensee analyzed both full power and hot zero power cases. Both cases assumed a conservatively large negative moderator temperature coefficient. The full power case assumed the reactor was in automatic or manual control. The Improved Thermal Design Procedure (ITDP) of Reference 4 was used in the analysis. For the accidental full opening of one feedwater control valve with the reactor at hot-zero power conditions, the licensee determined that the maximum reactivity insertion rate is less than the maximum reactivity insertion rate analyzed in the Uncontrolled-Rod-Cluster-Assembly-Bank-Withdrawal-at-Subcritical-Condition event. Thus, this hot-zero power case is bounded by the results obtained previously for the other event. In addition, if the event were to occur at a hot-zero power and an exactly critical condition, the power range high neutron flux trip (low setting) of about 25% of nominal full power will trip the reactor. The hot-full power case with the reactor in automatic control is more severe than the case with the reactor in manual control. For all excessive feedwater cases, continuous addition of cold feedwater is prevented by automatic closure of all feedwater isolation valves on steam generator high-high level signal. A turbine trip is then initiated and a reactor trip on a turbine trip is then assumed. The results presented by the licensee demonstrate the safe response of Cook Unit 1 to the event, at hot-full power and in automatic control, with the DNBR remaining well above the safety-limit DNBR value.

2.1.14 Excessive Increase in Secondary Steam Flow

The excessive-increase-in-secondary-steam-flow event was analyzed by the licensee to demonstrate core protection. This event is an overpower transient for which the fuel temperature will rise. It was analyzed because of reactor core temperature and pressure changes for the RTP program and because of changes to the OTDT and OPDT setpoints. The Cook Unit 1 reactor control system is designed to accommodate a 10% step load increase and a 5%-per-minute ramp load increase over the range of 15 to 100 percent of full power. Load increase in excess of these rates would probably result in a reactor trip. Four cases were analyzed by the licensee. These included minimum and maximum reactivity feedback cases with each case analyzed for both manual and automatic reactor control. For the minimum reactivity feedback cases, a zero moderator temperature coefficient was assumed to bound the positive moderator temperature coefficient. For all the cases, no credit was taken for the pressurizer heaters. The analyses used the ITDP of References 4. The studies show that the reactor reaches a new equilibrium condition for all the cases studied, with DNBR remaining well above the safety-limit DNBR value. The operators would follow normal plant procedures to reduce power to an acceptable value to conclude the event.

2.1.15 Loss of all AC Power to the Plant Auxiliaries

The loss-of-all-AC-power-to-the-plant-auxiliaries event was analyzed to demonstrate the adequacy of the heat removal capability of the auxiliary feedwater system. This transient is the limiting transient with respect to the possibility of pressurizer overfill. This event is more severe than the loss-of-load event because the loss of AC power results in a flow coastdown due to the loss of all four reactor coolant pumps. This results in a reduced capacity of the primary coolant to remove heat from the core. A positive moderator temperature coefficient was assumed in the analysis. A conservative decay heat model based on the ANSI/ANS-5.1-1979 decay heat standard (Ref. 8) was used. No credit is taken for the immediate release of the control rods caused by the loss of offsite power. Instead a reactor trip is assumed to occur on a steam generator low-low level signal. Pressurizer power operated relief valves and the maximum pressurizer spray flow rate was assumed to be available since a lower pressure results in a greater system expansion. The initial pressurizer water level is assumed to be at the maximum nominal setpoint of 62% narrow range span plus uncertainties of 5% narrow range span. The results demonstrate that natural circulation flow is sufficient to provide adequate decay heat removal following reactor trip and reactor coolant pump coastdown. The pressurizer does not fill. Thus, the loss of AC power does not adversely affect the core, the reactor coolant system, or the steam system, and the auxiliary feedwater system is sufficient to prevent water relief through the pressurizer relief or safety valves.

2.1.16 Steamline Break

The steamline break accident was analyzed by the licensee to assess the impact of the reduced reactor coolant system pesssure of the RTP program and the low steam pressure setpoint (lowered from 600 psig to 500 psig) of the coincidence logic with high-high steam flow for steamline isolation and safety injection actuation. An end-of-life shutdown margin of 1.6% delta K/K for no load, equilibrium xenon conditions, with the most reactive control rod stuck in its fully withdrawn position, was assumed. A negative moderator temperature coefficient corresponding to the end-of-line rodded core was assumed. The licensee evaluated four combinations of break sizes and initial plant conditions to determine the core power transient which can result from large area pipe breaks. The first case was the complete severance of a pipe downstream of the steam flow restrictor with the plant at no-load conditions and all reactor coolant pumps running. The second case was the complete severance of a pipe inside the containment at the outlet of the steam generator with the plant at no-load conditions and all reactor coolant pumps running. The third case is the same as the first case with the loss of offsite power simultaneous with the generation of a Safety Injection Signal (loss of offsite power results in reactor coolant pump coastdown). The fourth case is the same as the second case with loss of offsite power simultaneous with the generation of a Safety Injection Signal. A fifth case was performed to show that the DNBR remains above the safety-limit DNBR value in the event of the spurious opening of a steam dump or relief valve. The licensee determined that the first case was the limiting case, that is, the double-ended rupture of a main steam pipe located upstream of the flow restrictor with offsite power available and at no-load conditions. The results indicate that the core becomes critical with the control rods inserted (however, with the most reactive control rod stuck out) before boron solution at 2400 ppm enters the reactor coolant system. The core power peaks at less than the nominal full core power. The DNB analysis showed that the minimum DNBR remained above the safety limit DNBR value, even though this event is classified as an accident with fuel rods undergoing DNB not precluded. The analysis performed by the licensee demonstrates that a steamline break accident will not result in unacceptable consequences.

2.1.17 Rupture of Control Rod Drive Mechanism Housing (Rod Ejection Accident)

The rod ejection accident is analyzed at full power and hot, zero-power conditions for both beginning-of-cycle (BOC) and end-of-cycle (EOC). The analysis used ejected rod worth and transients peaking factors that are conservative. Reactor protection for a rod ejection is provided by neutron flux trip, high and low setting, and by the high rate of neutron flux increase trip. The analysis modeled the high neutron flux trip only. The maximum fuel temperature and enthalpy occurred for hot, full-power BOC case. The peak fuel enthalpy was, however, below 200 cal/gm for all the cases analyzed. For the hot, full-power cases, the amount of fuel melting in the hot pellet was less than 10%. Because fuel and clad temperatures and the fuel enthalpy do not exceed the FSAR limits, the conclusions of the FSAR remain valid.

Based on a review of the licensee's evaluation and analysis of the non-LOCA transients and accidents (2.1.3 through 2.1.17) for the reduced temperature and pressure operation (the RTP program), the staff concludes that they are acceptable because (1) approved methodologies and computer codes have been used, and (2) all applicable safety criteria have been met. This review is based on (1) a full power vessel average temperature of less than or equal to 567.8°F, (2) a steam generator tube plugging level of 10% with a peak tube plugging level of 15%, and (3) the minimum measured flow requirement of 91,600 gpm per loop is met.

2.1.18 Steam Generator Tube Rupture (SGTR) Accident

The licensee analyzed the steam generator tube rupture (SGTR) event for Cook Unit 1 using methodology and assumptions consistent with those used for the Cook FSAR SGTR analysis. The range of parameters associated with a future rerating program and the RTP program were used in sensitivity analyses to assess the impact of these programs on the primary-to-secondary break flow and the steam released to the atmosphere by the affected steam generator. These two factors affect the radiological consequences of an SGTR accident. In addition, the licensee's evaluation of the radiological doses considers the effect of the noble gas concentrations. The licensee states that the results of the analyses show that the doses remain within a small fraction (10%) of the 10 CFR Part 100 guidelines for both the thyroid and whole body doses. Since the worst case doses are within the 10 CFR Part 100 guidelines, the staff concludes that the analysis of the SGTR is acceptable

2.1.19 Fuel Structural Evaluation

The fuel assembly lift and buoyancy forces are increased for the RTP program at Cook Unit 1 because a reduction in reactor coolant system temperature of about 20°F will increase the coolant density by about 3%. The licensee evaluated this force increase against the fuel assembly allowable holddown load. The results of the evaluation show that the increased force is well within the minimum spring holddown force design margin. In addition, the licensee determined that the cold-leg break remains the most limiting pipe rupture transient with respect to lateral and vertical hydraulic forces. Based on the licensee's review, the staff concludes that the 15x15 fuel assembly design remains acceptable.

The fuel rod design was evaluated to assess the impact of future rerating. The licensee determined that the rod internal pressure criterion will continue to be the more important factor in fuel burnup capabilities. The fuel will also undergo more severe fuel duty because of the uprated power. The licensee plans to perform cycle-specific verification for each reload to assure that all fuel rod design criteria are met.

2.1.20 Justification for Pressurizer Level

The purpose of the Pressurizer High Level Limit is to ensure that a steam bubble is present in the pressurizer prior to power operation to minimize the consequences of overpressure transients and the possibility of passing water through the relief and safety valves. The safety analysis assumes a maximum water volume which corresponds to about 65% indicated level. This nominal indicated level is maintained during normal operation by the pressurizer level control system.

The licensee (and the fuel supplier - Westinghouse) recommends the use of 92% for the Pressurizer High Level trip limit. They state that this new trip limit will still ensure the presence of a steam bubble in the pressurizer. The pressurizer level will, however, be controlled to the nominal value. For normal operations (Condition I event), the reactor parameters, including the pressurizer level, do no significantly deviate from their nominal values. The licensee concludes that, for the pressurizer level to exceed the nominal level, a transient or accident must occur for which protective action is provided by the Reactor Protection System. Any other possible conditions for which the nominal level would be exceeded before and during a transient would require a transient or transients beyond those usually considered for an FSAR type of analysis. The staff concludes on the basis of the licensee's evaluation that a Pressurizer High Level Trip of 92% is acceptable.

2.2 BALANCE OF PLANT SYSTEMS

The licensee states that balance of plant (BOP) systems and components were analyzed for the effects of operation at reduced temperature and pressure conditions. The secondary side conditions for these analyses were determined using the Performance Evaluation and Power System Efficiencies (PEPSE) heat balance data (14.20 E6 lb/hr main steam flow and main feed flow). The systems reviewed were the non safety-related secondary side power generating and nonpower generating systems. Included in the licensee's analysis were portions of the main feedwater, main steam, steam generator blowdown (SGBS), component cooling water (CCWS), auxiliary feedwater (AFS), heating, ventilation, and air conditioning (HVAC), service water, waste disposal, fire protection, radiation monitoring, and spent fuel pool (SFP) cooling and cleanup systems.

The performance of the above BOP systems was evaluated at the reduced temperature and pressure by using the new primary side NSSS data (14.20E6 lb/hr main steam and main feed flow, and 434°F main feed temperature) furnished by Westinghouse. The licensee states that the impact on containment pressures and temperatures following a postulated design basis main steam line break was evaluated and its effect on equipment qualification was verified. The flooding analysis in safety-related areas of the plant as a result of a postulated pipe break was reevaluated due to the slight increase in flow rates in the main feed, condensate, and main steam systems. The turbine-generator system was also evaluted to confirm its integrity and performance at the increased steam volumetric flow rate and to verify that the original turbine missile analysis remains valid.

The licensee's analysis of BOP system performance provided the following findings concerning the RTP conditions at the present licensed power level of 3250 MWt NSSS power:

(a) The capability of the safety-related portion of the main feedwater system will not be affected and will continue to perform its safety function because the proposed RTP conditions are bounded by the existing main feedwater system design. The licensee's analysis of the pressure/temperature rating conditions for the system confirms that pressure boundary integrity will not be affected. In addition, the main feedwater system isolation valve closure time is not affected by the RTP-imposed conditions.

(b) The capability of the steam generator blowdown system to remove impurities from the secondary side remains essentially the same for the RTP-imposed conditions during normal operation based on the exsisting design.

(c) The reactor makeup water system's (MSW) capability to provide demineralized water for makeup and flushing operations throughout the NSSS auxilliaries, the radwaste systems, and fuel pool cooling and cleanup system is not challenged because the existing system design is based on the worst case demand which bounds the RTP conditions.

(d) The licensee confirmed that safety-related equipment will not be affected by changes in the flooding analysis due to the RTP conditions. Flooding in the auxiliary building due to failure of nonseismic Class I piping has been reviewed. The licensee analyzed systems having access to large water volumes and/or potentially large flowrates were considered as discussed in the FSAR. The only such system is the main feedwater system. Since the changes in flow in the main feedwater system are still within the design limits, the results concerning flooding discussed in the FSAR are still applicable.

Flooding in the containment is slightly increased due to the larger initial water mass in the reactor coolant system because of the higher density at the reduced temperature. This change was found to be within the volume margins used to determine the maximum flood-up elevation. The containment flooding evaluation in the FSAR remains valid at the RTP-induced conditions.

(e) The adequacy of the AFW system for accident mitigation was demonstrated in the Westinghouse accident analysis performed in support of the RTP program under the following scenarios:

1. Loss of main feedwater
2. Loss of offsite power
3. Main steam line rupture

Each accident analysis demonstrated acceptance criteria such as system overpressure limits or DNB limits. The AFW system's ability for design basis accident decay heat removal calculated in the RTP analysis is unaffected.

(f) As evaluated in the RTP analysis, the heat loads in both the primary and secondary systems due to reactor decay heat remain unchanged. Therefore, the Component Cooling Water System (CCWS) analysis and service water system (SWS) analysis in the FSAR remain valid.

(g) For main steam line breaks inside the containment structure, the pressure and temperature will remain within the bounds of the peak pressure and temperature used in the evaluation of containment performance. The initial primary temperatures and secondary steam pressures under the RTP conditions will be lower than those used in the FSAR analysis. The licensee has confirmed that containment environmental qualification of equipment inside containment is not affected.

(h) The superheated mass and energy release analysis outside containment was evaluated to address equipment qualification issues. The primary temperatures and secondary steam pressures resulting from the RTP conditions will be lower than those used in the FSAR analysis. The mass and energy release will be lower and operation with RTP will result in lower temperatures in the break areas. As such, the current superheat mass and energy release analysis outside containment remains bounding provided the full power vessel average temperature is restricted to the currently-licensed 567.8°F and below.

(i) The secondary pressure conditions assumed in the high energy steam line break analysis will be lower than those presented in the FSAR. These bound the proposed RTP conditions and therefore the current analysis is sufficient.

(j) The primary function of the spent fuel pool cooling system (SFPCS) is to remove decay heat that is generated by the elements stored in the pool. Decay heat generation is proportional to the amount of radioactive decay in the elements stored in the pool which is proportional to the reactor power history. Since the plant's rated power level of 3250 MWt remains unchanged, the demand on the SFPCS is not increased. The purification function is controlled by SFPCS demineralization and filtration rates that are not affected by the RTP conditions.

(k) The fire protection systems and fire hazards are independent of the plant operating characteristics with the exception of the slightly increased current requirements for the electric motor driven pumps in the primary system. The increased load is due to the more dense water being pumped under the RTP conditions. The increased current required is small and therefore is not considered to be a fire hazard.

(l) The licensee confirmed that BOP systems have the capability to maintain plant operation under the RTP-induced conditions without modification to the existing design.

The staff has reviewed the FSAR and licensee submittals in order to verify that safety-related BOP system performance capability, as analyzed, bounds the changes in design basis accident assumptions created by the RTP operation. The staff has confirmed that safety-related BOP system design capability, flooding protection, and equipment qualifications are bounded for the proposed rerating and therefore are considered acceptable as is.

Based on the above, the staff concludes that the proposed license amendment for the D.C. Cook Nuclear Plant Unit 1 concerning the Reduced Temperature and Pressure is within the existing safety-related BOP system design capability for design basis accident mitigation and, therefore, the staff's previous approval against the applicable licensing criteria for the main steam system, main feed system, CCWS, SWS, AFS, MSW, SGBS, SFPCS, flooding protection, containment performance, and equipment qualifications remain valid. The staff, therefore, finds the BOP systems concerned acceptable for continued operation at the proposed reduced temperature and pressure.

2.3 REACTOR VESSEL AND VESSEL INTERNALS

The reactor vessel is designed to the ASME Boiler and Pressure Vessel Code, Section III (1965 Edition with addenda through the winter 1966). The licensee has determined that the operation of the reactor vessel under the most limiting conditions of the RTP rerating is acceptable for its original 40-year design objective. All of the stress intensity and usage factor limits of the applicable code for the Unit 1 reactor vessel are still satisfied when the RTP is incorporated, with the exception of the 3Sm limit for the Control Rod Drive Motor (CRDM) housings and outlet nozzle safe end. However, the code permits exceeding the 3Sm limit provided plastic or elastic/plastic analysis criteria are met.

The licensee's review of the reactor vessels internals for the RTP program included three seperate areas: a thermal/hydraulic assessment, a RCCA drop time evaluation, and a structural assessment. Force increases were calculated for the upper core plate, across the core barrel, and in the upper internals near the outlet nozzles. In these areas the existing margin was determined to be sufficient to accommodate the increased stresses. The results of this review indicate that the original reactor internals components remain in compliance with the current design require-ments when operating at the new range of primary temperatures and pressures.

The PTS rule requires that at the end-of-life of the reactor vessel, the projected reference temperature (calculated by the method given in 10 CFR 50.61(b)(2), RT/pts) value for the materials in the reactor vessel beltline be less than the screening criterion in 10 CFR 50.61(b)(2). The RT/pts value is dependent upon the initial reference temperature, margins for uncertainty in the initial reference temperature and calculational procedures, the amounts of nickel and copper in the material, and the neutron fluence at the end-of-life of the reactor vessel. Of these properties, only neutron fluence is affected by rerating with RTP. Since the colder coolant in the downcomer region is more dense and thus provides for a more efficient neutron shield for the reactor vessel, fluence estimates are lower than those at current operating conditions. All other properties are independent of the RTP-induced conditions.

The effects of NRC Generic letter 88-11, dated July 12, 1988, regarding Regulatory Guide 1.99 Rev. 2 were evaluated by Westinghouse and determined to not be significant for RTP. The effect of RTP will be incorporated by the licensee in future PTS submittals.

An evaluation was performed to determine the impact of RTP rerating on the applicability of the PTS screening criteria in terms of vessel failure. A probabilistic fracture mechanics sensitivity study of limiting PTS transient characteristics, starting from a lower operating temperature, showed that the conditional probability of reactor vessel failure will not be adversely affected. Therefore, the overall risk of vessel failure will not be adversely impacted, meaning that the screening criteria in the PTS Rule are still applicable for the D.C. Cook Nuclear Plant Unit 1 reactor vessel relative to rerated conditions.

Analysis of the CRDM housings and the outlet nozzle safe end shows the maximum range of primary plus secondary stress intensity exceed the 3Sm limit. The licensee, however, performed a simplified elastic/plastic analysis in accordance with paragraph NB-3228.3 of the ASME Boiler and Pressure Vessel Code, Section III (1971 or later edition) and the higher range of stress intensity is justified.

Therefore, based on the licensee's reviews and analysis of the above portions of the reactor vessel and internals, the staff concludes that the conditons imposed on the reactor vessel and internals by the RTP rerating are acceptable.

2.4 TURBINE MISSILES

The FSAR turbine missile analysis is based on a low pressure turbine failure. The licensee's analysis of the slightly changed steam conditions entering the low pressure turbine shows that the probabilty of a low pressure turbine missile is virtually unaffected.

The factors that directly or indirectly cause stress corrosion cracking in the low pressure turbine wheels are steam pressure and temperature, mass flow rate, steam moisture content, water chemistry, oxygen level, and turbine speed. The licensee reported that changes in these factors are negligible due to the RTP-induced conditions. The only noticeable change that the staff can determine is a 1.0% increase in the steam flow rate.

The staff's conclusion, based on the licensee's review, is that the turbine missile hazard is neglibily affected by the RTP conditons and is, therefore, acceptable.

2.5 PLANT STRUCTURAL AND THERMAL DESIGN

The NSSS review consisted of comparing the existing NSSS design with the performance requirements at the rerated RTP conditions.

The current components of the Cook Unit 1/model 51 steam generators continue to satisfy the requirements of the ASME B&PV Code, Section III,(the code applicable for the design of the Cook Nuclear Plant Unit 1), for this program. In addition, thermal hydraulic evaluations of the steam generators show acceptable stability and circulation ratios at the RTP rerated conditions. Circulation ratio is primarily a function of power, which is unchanged, therefore is itself virtually unchanged. The dampening factor characterizes the thermal and hydraulic stability of the steam generator. Westinghouse has determined that all dampening factors are negative at nearly the same value as the current operating conditions. A negative dampening factor indicates a stable device. Since the code requirements continue to be satisfied, and since stability and circulation ratios have been determined by Westinghouse to be within the design criteria, the staff concludes that RTP operation is acceptable for the Model 51 steam generators.

The pressurizer structural analysis was performed by modifying the original D.C. Cook Nuclear Plant Pressurizer analysis ("Model 51 Series Pressurizer Report"). The analysis was performed to the requirements of the ASME Code 1968 Edition, which is the design basis for the D.C. Cook Nuclear Units. The only ASME Code requirement affected by the transient modifications was fatigue. The limiting components for fatigue usage factors are the upper shell and the spray nozzle, which are calculated to be 0.97 and 0.99 respectively. These remain, however, within the ASME acceptance criteria of 1.0 and are, therefore, acceptable to the staff.

Reactor coolant pump hydraulics and motor adequacy were reviewed for the proposed RTP conditions by Westinghouse. The increased hot horsepower and stator temperature conditions are within the NEMA Class B limits. A review of generic Reactor Coolant Pump stress reports for model 93A pumps by Westinghouse finds that all the design requirements provide adequate bounding of the RTP-induced conditions and, therefore, the staff finds this acceptable.

Due to lower temperatures from the RTP program, the RCS will not expand as much as currently designed. This will result in support gaps being present in locations that were previously zero. The small gaps in the support structure may result in increased dynamic loading (both seismic and LOCA) in localized areas. The overall LOCA loadings on the RCS, however, remain approximately the same for the following reasons:

1. The lower RCS temperatures yield lower thermal loadings.

2. The D. C. Cook Nuclear Plant has a leak before break design methodology which allows the faulted condition evaluation to proceed without having to consider loadings from postulated breaks in the primary loop piping.

The seismic margin available for this plant is also significant which means that there are no components in the system which are close to their allowable stresses. Based on the above, the temperatures associated with the RTP rerating are, therefore, acceptable to the staff for the loop piping, the loop supports, and the primary equipment nozzles.

The effects of the D.C. Cook Nuclear Plant RTP rerating on the operability and design basis analysis of the CRDM's of Unit 1 were reviewed. The RTP rerating does not affect the operability or service duration of the CRDM latch assembly, drive rod, or coil stack. The CRDM latch assembly and drive rod were originally designed for 650°F, and the design basis stress and fatigue calculations remain representative for these components since the components are exposed to the hot leg temperature, which has not increased. The coil stack is located on the outside of the pressure housing which is subject to ambient containment temperatures, which have not changed. An evaluation was performed on the impact of the RTP rerated operating conditions on the structural analysis of the CRDM pressure housing. The component of the pressure housing which experiences the greatest stress range and has the highest fatigue usage factor is the upper canopy. This is the pressure housing seal weld between the rod travel housing and the cap. Westinghouse provided a review on the impact of the differences between the original normal and upset condition transients and those of the RTP on the code allowable stress levels and fatigue usage factors. The results of the evaluation are:

1. The maximum stress intensity range is equal to 109,960 psi, which is less than the maximum allowable range of thermal stress of 127,105 psi which was previously found to be acceptable.

2. The total fatigue usage factor is equal to 0.672, which is less than the allowable limit of 1.0 (ASME Section III, 1971 Edition).

The staff concludes, based on licensee evaluations, that the impact of the RTP program on the CRDM's is within design criteria and, therefore, is found to be acceptable.

2.6 CONTAINMENT EVALUATION

Short-Term Containment Response

As part of the analysis to support RTP operation, the reactor cavity and loop subcompartments short-term pressurization in the event of a break of large coolant piping or a steam line was reanalyzed by Westinghouse. In some of those areas, the analyzed pressure exceeded the structural limits as expressed in the FSAR. These structures were reevaluated using the peak pressures obtained from the RTP analysis, WCAP 11902 (ref.2), to confirm that the acceptance criteria of Section 5.2.2.3 of the updated FSAR, titled "Containment Design Stress Criteria," were met.

The original design of the containment included a number of considerations of which the subcompartment pressures were but one. For example, radiation shielding requirements may have dictated a thicker concrete slab than was necessary from a structural perspective. The actual capacity is generally greater than the design pressures stated in the FSAR, and is further increased due to the fact that the materials used are stronger than the required minimum design strengths. In the RTP structural review, advantage was taken of these greater capacities by performing manual or finite element evaluations of the affected structural elements. The greater material strengths were used in the analysis where appropriate.

Loop Subcompartments

The containment building subcompartments are the fully or partially enclosed spaces within the containment which contain high energy piping. The subcompartments are designed to limit the adverse effects of a postulated high energy pipe rupture.

The results of the short term containment analyses and evaluations for the D.C. Cook Nuclear Plant Unit 1 demonstrate that, for the pressurizer enclosure, the fan accumulator room, and the steam generator enclosure, the resulting peak pressures remain below the allowable design peak pressures. For the loop compartments, the peak calculated pressures at the RTP rerated conditions are higher than the FSAR design allowables. For these areas, structural evaluations were performed as discussed above for the revised peak pressures, and the structural adequacy of the containment subcompartments have been confirmed (Ref. 10) as follows:

Differential Pressure, Node 1 or 6 to Node 25

This is the differential pressure from the reactor coolant loop compartments adjacent to the refueling canal nodes 1 or 6 across the operating deck to the upper containment.

Original Design pressure	16.6 psi
Original Calculated pressure	14.1 psi
New Calculated pressure	18.7 psi

The licensee demonstrated the increased differential pressure to be acceptable by review of existing computer analysis of the reactor coolant pump hatch covers and reevaluation of the operating deck load carrying capacity.

Differential Pressure, Node 2 or 5 to Node 25

This is the differential pressure across the operating deck from the reactor coolant loop compartments located 90 degrees from the refueling canal to the upper containment.

Original Design pressure	12.0 psi
Original Calculated pressure	10.6 psi
New Calculated pressure	13.0 psi

The licensee demonstrates the increased differential pressure to be acceptable by comparison to Node 1 and Node 6 areas. The slabs in both areas are the same.

Peak Shell Pressure

This is the differential pressure across the containment shell to the outside, for nodes located in the ice condenser inlet areas closest to the refueling canal.

Original Design pressure	12.0 psi
Original Calculated pressure	10.8 psi
New Calculated pressure	14.0 psi

The licensee demonstrates the increased pressure to be acceptable by evaluation on a localized basis. The containment shell can handle pressures well in excess of the overall 12 psi design pressure. The average pressure over the structurally significant portion of the containment shell surrounding and including these nodes is smaller than the 12 psi containment shell design pressure.

Reactor Cavity

The reactor cavity is the structure surrounding the reactor with penetrations for the main coolant piping. This structure is designed to limit the adverse effects of the initial pressure response to a loss of coolant accident. The results of the reactor cavity analysis and evaluations for the D. C. Cook Nuclear Plant Unit 1 demonstrate that, for the reactor vessel annulus and pipe annulus, the resulting peak pressures at the RTP rerated conditions are within the FSAR design allowables. For the upper and lower reactor cavities the peak calculated pressures under RTP conditions exceeded the structural design pressures (Ref. 2, Sections 3.7.2 and 3.7.3) as stated in the FSAR. For these areas, structural evaluations were performed for the revised peak pressures, and the structural adequacy of the containment subcompartment has been confirmed (Ref. 10) as follows:

Missile Shield, Refueling Canal Bulkhead Blocks, and Upper Reactor Cavity Wall Differential Pressures

The upper reactor cavity walls surround the reactor head. The missile shields and the refueling canal bulkheads are blocks separating the upper reactor cavity from upper containment. The missile shield is bolted down during operation, and is removable for refueling. The refueling canal bulkheads fit snugly in grooves in the upper reactor cavity walls.

	Cavity Wall	Missile Shield and Bulkheads
Original Design pressure	48.0 psi	48.0 psi
Original Calculated pressure	44.1 psi	44.1 psi
New Calculated pressure	48.4 psi	54.3 psi

The licensee demonstrates the increased pressure for the cavity wall to be acceptable by finite element analysis of the entire upper reactor cavity wall.

The licensee has demonstrated the increased pressure for the missile shields and the bulkheads to be acceptable by manual calculation. The test cylinder break strength of the concrete, which is higher than the design strength, was also taken into consideration.

Peak Lower Cavity Pressure

This is the cavity located under the reactor vessel. The peak pressure is used in the structural analysis rather than the differential pressure since most of the cavity walls are in the foundation mat.

Original Design pressure	15.0 psi
Original Calculated pressure	13.8 psi
New Calculated pressure	18.5 psi

The licensee demonstrated that the increased pressures are acceptable by manual calulation.

The staff concludes, based on the licensee's demonstration, that the D. C. Cook Nuclear Plant's design basis pertaining to containment short term response, as stated in Chapter 5.2.7.3 of the FSAR, is adequate for RTP operation, and therefore, is acceptable. The licensee must update the FSAR to reflect the higher structural design values.

Long Term Containment Pressure

The long term peak containment pressure analysis supports operation with the RHR crosstie valves closed at a power level of 3425 MWt for both Units 1 and 2 containment structure. This analysis contained additional justification for operation under the RTP conditions (Ref. 11) and was approved by the staff Safety Evaluation dated January 30, 1989 (Ref. 12).

2.7 NUCLEAR, PROCESS AND POST-ACCIDENT SAMPLING SYSTEMS

The Nuclear Sampling System (NSS) is designed to provide representative samples for laboratory analyses used to guide the operation of various primary and secondary systems throughout the plant during normal operation. Since reduction of sample pressure and temperature, when necessary, is already being done by heat exchangers and needle valves, the parameters associated with the RTP program do not affect the performance of the NSS. With no power uprating, the source term remains unchanged. Therefore, the staff concludes that operation under RTP conditions is acceptable for the NSS.

The staff finds that, since no power uprating is being proposed at this time, there is an insignificant effect on the post-accident containment thermal conditions and therefore the existing post-accident sampling system remains adequate and is acceptable.

Operation under RTP conditions results in slight reductions in secondary side temperatures and pressures with no change in the source term. The staff concludes that the change can be accommodated by the process sampling system without causing degradation of their performance, and is, therefore, acceptable.

2.8 ELECTRIC SYSTEMS DESIGN

Operation under RTP conditions results in minor changes to the heat balance. The only impact noted on the electrical systems is the slight increase in motor current for the motors used as prime movers of primary coolant. The required power is increased by the higher densities encountered due to the RTP program. The licensee has reviewed cable penetrations, busses, and motor ratings to conclude that there is sufficient design margin to handle the increased load. The staff finds, based on the licensee's evaluation, that the proposed RTP program minimally affects the electric power system and associated loads and is therefore, acceptable.

3.0 TECHNICAL SPECIFICATIONS

1. Definition 1.38 on design thermal power is being deleted on page 1-7 of the Technical Specifications (TS's) because there is no longer a single design thermal power at which all the transient and accident analyses have been performed. The licensed power level for Cook 1 remains 3,250 MWt. This change is acceptable.

2. Table 1-3 on page 1-10 is being deleted because it previously gave information on the analyses performed at the design thermal power. This change is acceptable because the definition of design thermal power is being deleted also.

3. Figure 2.1-1 on page 2-2 is being revised to reflect the revised DNBR safety limit of 1.45. This change is acceptable because it is supported by the safety analysis.

4. The pressurizer pressure low setpoint (Item 9 of Table 2.2-1 on page 2-5) is increased by 10 psig. This is acceptable because it was assumed in the large- and small-break LOCA analyses.

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4. The pressurizer pressure low setpoint (Item 9 of Table 2.2-1 on page 2-5) is increased by 10 psig. This is acceptable because it was assumed in the large- and small-break LOCA analyses.

5. The Overtemperature-Delta T trip setpoint equation (pages 2-7 and 2-8) is being revised in terms of rated thermal power rather than design thermal power. In addition, this revised OTDT trip setpoint protects the core safety limits of Figure 2.1-1. This change is acceptable because it is supported by the non-LOCA safety analyses.

6. The Overpower-Delta T trip setpoint equation (page 2-9) is being revised to reflect the revised core safety limits of Figure 2.1-1. This equation is also being defined in terms of the indicated T_{avg} at rated thermal power. These changes are acceptable because they are supported by the safety analysis for the RTP program.

7. Technical Specification 3.2.2 on page 3/4 2-5 is being revised from a maximum F_Q of 2.10 to 2.15. This change is acceptable because it is supported by the large-break LOCA analysis. The F_Q values for Exxon fuel are being deleted because this fuel will no longer be used at Cook Unit 1.

8. The K(Z) curve applicable to Exxon fuel (page 3/4 2-7) is being deleted. This is acceptable because Exxon fuel will no longer be used at Cook Unit 1.

9. The K(Z) curve for Westinghouse fuel (page 3/4 2-8) is being revised. This is acceptable because it is supported by the new LOCA analysis for Cook Unit 1.

10. The F-Delta H limit applicable to Exxon fuel (page 3/4 2-9) is being deleted. This is acceptable because Exxon fuel will no longer be used at Cook Unit 1.

11. Table 3.2-1 on page 3/4 2-14 on DNB parameters is being revised. T_{avg} must be less than or equal to 570.9°F, the pressurizer

pressure must be less than or equal to 2050 psig, and the reactor coolant system total flow rate must be greater than or equal to 366,400 gpm. These changes are acceptable because they reflect the safety analysis for the RTP program.

12. Technical Specification 3.2.6 on page 3/4 2-15 is being revised to change F_Q in the APL limit to 2.15. This change is acceptable because it reflects the new F_Q limit of Specification 3.2.2. The limits on APL applicable to Exxon fuel are being deleted because Exxon fuel will no longer be used at Cook Unit 1.

13. Functional Units 2 and 11 of Table 3.2-2 on page 3/4 3-10 are being changed. Functional Unit 2 incorporates an editorial change to indicate that the response time is applicable to both the high and low setpoints of the Power Range Neutron Flux trip. This change is acceptable because it is editorial in nature. Functional Unit 11 is being changed from a response time of "not applicable" to "equal to or less than 2 seconds." This is acceptable because this trip on pressurizer water level-high was modeled in the analysis of the control rod withdrawal-at-power event.

14. Functional Units 1.f and 4.d of Table 3.3-4 on pages 3/4 3-24 and 3/4 3-26 are being changed to decrease the steamline pressure low setpoint by 100 psig. These changes are acceptable because they are supported by the steamline break analysis and the steamline break mass and energy evaluations.

15. Technical Specification 3.4.4 on page 3/4 4-6 is being revised to 92% of span. This change is acceptable because it is supported by the safety analysis.

16. Technical Specification 3.5.1.b on page 3/4 5-1 is being revised from an accumulator borated minimum water volume of 929 to 921 cubic feet. This change is acceptable because it is consistent with the LOCA analysis for Cook Unit 1.

17. Surveillance Requirement 4.5.2.f is being revised to reduce the discharge pressure of the safety injection pump and the residual heat removal pump. These changes are acceptable because they are consistent with the LOCA analyses.

18. Surveillance Requirement 4.5.2.h is being revised by adding a requirement to verify that the charging pump discharge coefficient is within a specified range following ECCS modifications. The footnote is broken into four parts for clarity. This change is acceptable because it ensures that the flow delivered to the core by the charging pumps in the event of a LOCA is within the analyzed values.

19. Surveillance Requirement 4.7.1.2 on page 3/4 7-5 is being revised to change the discharge pressure requirements of the motor and turbine driven auxiliary feedwater pumps to 1375 psig and 1285 psig, respectively. This corresponds to a 5% degradation of the pumps

from the manufacturer's pump head curve. These changes are acceptable because they are consistent with the changes for the RTP program.

20. Basis page B 2-1(a) is being changed to incorporate the design limit and safety analysis limit DNBR values. The DNB limits for Exxon fuel are being deleted since Exxon fuel is no longer used at Cook Unit 1. The design limit and safety analysis limit DNBR values are acceptable because they are consistent with the RTP program.

21. Basis page B 2-2 is being revised to delete reference to F-Delta H for Exxon fuel and to design thermal power. These changes are acceptable because references to both items have been deleted in the Specifications.

22. Bases page B 2-4 is being revised to reflect the changes to the Overtemperature-Delta T trip function. The changes are acceptable because they reflect changes made to the Specifications.

23. Bases page B 2-5 is being revised to reflect the changes to the Overpower-Delta T trip function and the pressurizer water level-high trip. These changes are acceptable because they reflect changes to the Specifications.

24. Bases page B 3/4 2-1 is being revised to replace the minimum DNBR value of 1.69 by the words "the safety limit DNBR". This change is acceptable because it will avoid changes to the Bases if the safety limit DNBR value is changed.

25. Surveillance Requirement 4.1.1.5.b is being changed to require T_{avg} determination of T_{avg} every 30 minutes when the reactor is critical and T_{avg} is less than 545°F. This change is supported by Reference 9 and allows a full power T_{avg} of 550°F for Cook Unit 1 Cycle 11 without requiring a monitoring every 30 minutes while at full power, which the previous value of 551°F would have required. This change is acceptable because the intent of maintaining the minimum coolant temperature for criticality of Specification 3.1.1.5 is preserved.

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32 and 51.35, an environmental assessment and finding of no significant impact have been prepared and published in the Federal Register on June 9, 1989(54 FR 24774). Accordingly, based upon the environmental assessment, we have determined that the issuance of he amendment will not have a significant effect on the quality of the human environment.

5.0 CONCLUSION

The staff has reviewed the request by the Indiana and Michigan Power Company to operate the Donald C. Cook Nuclear Plant Unit 1 at the reduced temperatures and pressures of the RTP program. Reactor operation is restricted to an upper limit on T_{avg} of 567.8°F because the steamline break mass and energy release inside containment was not reanalyzed as part of the RTP program. Although the safety analysis was performed at power ratings which would support a possible power uprating for Cook Unit 1, power uprating is not addressed in the staff's review. The power of D.C. Cook Nuclear Plant Unit 1 is limited to the present rated thermal power of 3250 MWt. Based on its review, the staff concludes that appropriate material was submitted and that normal operation and the transients and accidents that were evaluated and analyzed are acceptable. The Technical Specifications submitted for this license amendment suitably reflect the necessary modifications for the operation of Cook Unit 1.

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

Date: June 9, 1989

Principal Contributors: Dan Fieno
John Stang, NRR
Anthony Gody, NRR

6.0 REFERENCES

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2. "Reduced Temperature and Pressure Operation for Donald C. Cook Nuclear Plant Unit 1 - Licensing Report," D. L. Cecchett and D. B. Augustine, WCAP-11902, October 1988.

3. Ellenberger S.L., et al., "Design Bases for the Thermal Overpower-Delta T and Thermal Overtemperature-Delta T Trip Functions," WCAP-8746, March 1977.

4. Chelemer, H.; Boman, L.H.; Sharp, D.R., "Improved Thermal Design Procedures," WCAP-8567, July 1975.

5. Butler, J.C., and Love, D.S., "Steamline Break Mass/Energy Releases for Equipment Qualification Outside Containment," WCAP-10961, Rev. 1 (proprietary) and WCAP-11184 (nonproprietary), October 1985.

6. Morita, T., et al., "Dropped Rod Methodology for Negative Flux Rate Trip Plants," WCAP-10297-P-A (proprietary) and WCAP-10298-A (nonproprietary), June 1983.

7. Letter (AEP:NRC:1067B) from M. P. Alexich (Indiana and Michigan Power Company) to the USNRC, dated February 6, 1989.

8. "American National Standard for Decay Heat Power in Light Water Reactors," ANSI/ANS-5.1-1979, August 1979.

9. Letter (AEP:NRC:1067A) from M. P. Alexich (Indiana and Michigan Power Company) to the USNRC, dated December 30, 1988.

10. Letter (AEP:NRC:1067C) from M. P. Alexich (Indiana and Michigan Power Company) to the USNRC, dated March 14, 1989.

11. Letter (AEP:NRC:1024D) from M. P. Alexich to T. E. Murley (NRC), dated August 22, 1988. Includes WCAP-11908, "Containment Integrity Analysis for Donald C. Cook Nuclear Plants, Units 1 and 2."

12. Letter, J. F. Stang (NRC) to M. P. Alexich (IMECo), dated January 30, 1989.