August 27, 1990

Docket Nos. 50-315
and 50-316

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| SMeador | ACRS(10) |

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

Dear Mr. Alexich:

SUBJECT: AMENDMENT NOS. 148 AND 134 TO FACILITY OPERATING LICENSE NOS. DPR-58 AND DPR-74: (TAC NOS. 75395, 75396 AND 76816)

The Commission has issued the enclosed Amendment No. 148 to Facility Operating License No. DPR-58 and Amendment No. 134 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Units Nos. 1 and 2. The amendments consist of changes to the Technical Specifications in response to your applications dated February 6, 1990 (as supplemented May 29, 1990 and July 23, 1990) and May 11, 1990.

These amendments modify the Technical Specifications for Unit 2 Cycle 8 to allow for a transition to Westinghouse 17 x 17 VANTAGE 5 fuel as a replacement for some 17 x 17 Advanced Nuclear Fuels Corporation (ANF) fuel. They also propose administrative changes and corrections to the Technical Specifications for both Units 1 and 2. These amendments further modify Units 1 and 2 Technical Specifications to achieve consistency between both units where the changes are clearly approved for the other unit.

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

Timothy G. Colburn, Project Manager
Project Directorate III-1
Division of Reactor Projects - III,
IV, V & Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 148 to DPR-58
2. Amendment No. 134 to DPR-74
3. Safety Evaluation

cc w/enclosures:
See next page

| LA/PD31:DRSP | PM/PD31:DRSP | D/PD31:DRSP | OGC |
|---|---|---|---|
| SMeador | TColburn | RPierson | |
| 8/21/90 | 8/20/90 | 8/21/90 | 8/24/90 |

9009060148 900827
PDR ADOCK 05000315
P PDC

COOK AMEND 75395/6 & 75816

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

August 27, 1990

Docket Nos. 50-315
and 50-316

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

Dear Mr. Alexich:

SUBJECT: AMENDMENT NOS.148 AND 134TO FACILITY OPERATING LICENSE NOS. DPR-58 AND DPR-74: (TAC NOS. 75395, 75396 AND 76816)

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These amendments modify the Technical Specifications for Unit 2 Cycle 8 to allow for a transition to Westinghouse 17 x 17 VANTAGE 5 fuel as a replacement for some 17 x 17 Advanced Nuclear Fuels Corporation (ANF) fuel. They also propose administrative changes and corrections to the Technical Specifications for both Units 1 and 2. These amendments further modify Units 1 and 2 Technical Specifications to achieve consistency between both units where the changes are clearly approved for the other unit.

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Timothy G. Colburn, Project Manager
Project Directorate III-1
Division of Reactor Projects - III,
IV, V & Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 148 to DPR-58
2. Amendment No. 134 to DPR-74
3. Safety Evaluation

cc w/enclosures:
See next page

Mr. Milton Alexich
Indiana Michigan Power Company

Donald C. Cook Nuclear Plant

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Columbus, Ohio 43216

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. 148
License No. DPR-58

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

   A. The applications for amendment by Indiana Michigan Power Company (the licensee) dated February 6, 1990 (as supplemented May 29, 1990) and May 11, 1990, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the 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.

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PDR ADOCK 05000315
P PDC

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. 148 , 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

Timothy G. Colburn for

Robert Pierson, Director
Project Directorate III-1
Division of Reactor Projects - III,
IV, V & Special Projects
Office of Nuclear Reactor Regulation

Attachment: August 27, 1990
Changes to the Technical
Specifications

Date of Issuance:

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

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 134
License No. DPR-74

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

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated February 6, 1990, and as supplemented May 29, 1990 and July 23, 1990, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 134 , 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

Timothy R. Colburn for
Robert Pierson, 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: August 27, 1990

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 148 TO FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

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

| REMOVE | INSERT |
|---|---|
| 3/4 1-1 | 3/4 1-1 |
| 3/4 1-2 | 3/4 1-2 |
| 3/4 1-3 | 3/4 1-3 |
| 3/4 1-3a | 3/4 1-3a |
| 3/4 1-3b | 3/4 1-3b |
| 3/4 2-2 | 3/4 2-2 |
| 3/4 3-11 | 3/4 3-11 |
| 3/4 5-5 | 3/4 5-5 |
| B3/4 1-1 | B3/4 1-1 |

## 3/4.1 REACTIVITY CONTROL SYSTEMS

### 3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - TAVG GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.6% Delta k/k.

APPLICABILITY: MODES 1, 2*, 3, and 4.

ACTION:

With the SHUTDOWN MARGIN less than 1.6% Delta k/k, immediately initiate and continue boration at greater than or equal to 10 gpm of a solution containing greater than or equal to 20,000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.6% Delta k/k:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s).

b. When in MODE 1 or MODE 2 with Keff greater than or equal to 1.0, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.5.

c. When in MODE 2 with Keff less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.5.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit of Specification 3.1.3.5.

*See Special Test Exception 3.10.1.

SURVEILLANCE REQUIREMENTS (Continued)

e. When in MODES 3 or 4, at least once per 24 hours by consideration of the following factors:

1. Reactor coolant system boron concentration,

2. Control rod position,

3. Reactor coolant system average temperature,

4. Fuel burnup based on gross thermal energy generation,

5. Xenon concentration, and

6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within plus or minus 1% Delta k/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

SHUTDOWN MARGIN - TAVG LESS THAN OR EQUAL TO 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 1.0% Delta k/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 1.0% Delta k/k, immediately initiate and continue boration at greater than or equal to 10 gpm of a solution containing greater than or equal to 20,000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.0% Delta k/k:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s).

b. At least once per 24 hours by consideration of the following factors:

   1. Reactor coolant system boron concentration,

   2. Control rod position,

   3. Reactor coolant system average temperature,

   4. Fuel burnup based on gross thermal energy generation,

   5. Xenon concentration, and

   6. Samarium concentration.

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LIMITING CONDITION FOR OPERATION (Continued)

b. THERMAL POWER shall not be increased above 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER unless the indicated AFD is within the target band and ACTION 2.a) 1), above has been satisfied.

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD has not been outside of the target band for more than 1 hour penalty deviation cumulative during the previous 24 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AXIAL FLUX DIFFERENCE shall be determined to be within its limits during POWER OPERATION above 15% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

1. At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

2. At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status if the AFD has been outside of the target band for any period of time in the previous 24 hours of operation.

b. Monitoring and logging the indicated AXIAL FLUX DIFFERENCE for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AXIAL FLUX DIFFERENCE Monitor Alarm is inoperable. The logged values of the indicated AXIAL FLUX DIFFERENCE shall be assumed to exist during the interval preceding each logging.

SURVEILLANCE REQUIREMENTS (Continued)

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 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 | Greater than or equal to 2405 psig |
| 2. | Safety Injection pump | Greater than or equal to 1409 psig |
| 3. | Residual heat removal pump | Greater than or equal to 190 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.

TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

| FUNCTIONAL UNIT | RESPONSE TIME |
|---|---|
| 12. Loss of Flow - Single Loop (Above P-8) | ≤ 1.0 seconds |
| 13. Loss of Flow - Two loops (Above P-7 and below P-8) | ≤ 1.0 seconds |
| 14. Steam Generator Water Level--Low-Low | ≤ 1.5 seconds |
| 15. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level | NOT APPLICABLE |
| 16. Undervoltage-Reactor Coolant Pumps | ≤ 1.2 seconds |
| 17. Underfrequency-Reactor Coolant Pumps | ≤ 0.6 seconds |
| 18. Turbine Trip | |
| A. Low Fluid Oil Pressure | NOT APPLICABLE |
| B. Turbine Stop Valve | NOT APPLICABLE |
| 19. Safety Injection Input from ESF | NOT APPLICABLE |
| 20. Reactor Coolant Pump Breaker Position Trip | NOT APPLICABLE |

## 3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

### 3/4.1.1 BORATION CONTROL

### 3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS $T_{avg}$. The most restrictive condition occurs at EOL, with $T_{avg}$ at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.6% Delta k/k is initially required to control the reactivity transient and automatic ESF is assumed to be available. With Tavg less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% Delta k/k SHUTDOWN MARGIN provides adequate protection for this event.

The SHUTDOWN MARGIN requirements are based upon the limiting conditions described above and are consistent with FSAR safety analysis assumptions.

### 3/4.1.1.3 BORON DILUTION

A minimum flow rate of at least 2000 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 2000 GPM will circulate an equivalent Reactor Coolant System volume of 12,612 plus or minus 100 cubic feet in approximately 30 minutes. The reactivity change rate associated with boron reductions will therefore be within the capability for operator recognition and control.

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

The limitations on MTC are provided to ensure that the assumptions used in the accident and transient analyses remain valid through each fuel cycle. The surveillance requirement for measurement of the MTC at the beginning, and near the end of each fuel cycle is adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO.134 TO FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

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

| REMOVE | INSERT |
|---|---|
| 2-2 | 2-2 |
| 2-5 through 2-9 | 2-5 through 2-9 |
| 3/4 1-1 through 3/4 1-3b | 3/4 1-1 through 3/4 1-3b |
| 3/4 1-16 | 3/4 1-16 |
| 3/4 1-23 | 3/4 1-23 |
| 3/4 2-2 | 3/4 2-2 |
| 3/4 2-15 through 3/4 2-19 | 3/4 2-15 through 3/4 2-19 |
| 3/4 3-9 through 3/4 3-10 | 3/4 3-9 through 3/4 3-10 |
| 3/4 3-20 through 3/4 3-21 | 3/4 3-20 through 3/4 3-21 |
| 3/4 3-23 through 3/4 3-26 | 3/4 3-23 through 3/4 3-26 |
| 3/4 3-28 | 3/4 3-28 |
| 3/4 3-30 through 3/4 3-33 | 3/4 3-30 through 3/4 3-33 |
| 3/4 4-6 | 3/4 4-6 |
| 3/4 5-1 | 3/4 5-1 |
| 3/4 5-5 through 3/4 5-6 | 3/4 5-5 through 3/4 5-6 |
| B 2-1 | B 2-1 |
| B 2-4 through B 2-5 | B 2-4 through B 2-5 |
| B 2-7 | B 3/4 2-7 |
| B 3/4 1-1 | B 3/4 1-1 |
| B 3/4 1-3 | B 3/4 1-3 |
| - | B 3/4 1-4 |
| B 3/4 2-4 and B 3/4 2-5 | B 3/4 2-4 through B 3/4 2-6 |
| B 3/4 7-1 | B 3/4 7-1 |

DESIGN FLOW = 91,600 GPM/LOOP

DESCRIPTION OF SAFETY LIMITS

| Pressure (psia) | Power (frac) | Tavg (°F) | Power (frac) | Tavg (°F) | Power (frac) | Tavg (°F) | Power (frac) | Tavg (°F) |
|---|---|---|---|---|---|---|---|---|
| 1775 | 0.00 | 615.4 | 0.98 | 583.8 | 1.02 | 580.9 | 1.2 | 558.1 |
| 2000 | 0.00 | 631.8 | 0.86 | 605.8 | 0.96 | 597.5 | 1.2 | 568.5 |
| 2100 | 0.00 | 639.1 | 0.82 | 614.0 | 0.96 | 601.6 | 1.2 | 573.1 |
| 2250 | 0.00 | 649.2 | 0.72 | 628.6 | 0.98 | 605.2 | 1.2 | 580.4 |
| 2400 | 0.00 | 659.0 | 0.62 | 642.0 | 1.1 | 599.0 | 1.2 | 588.1 |

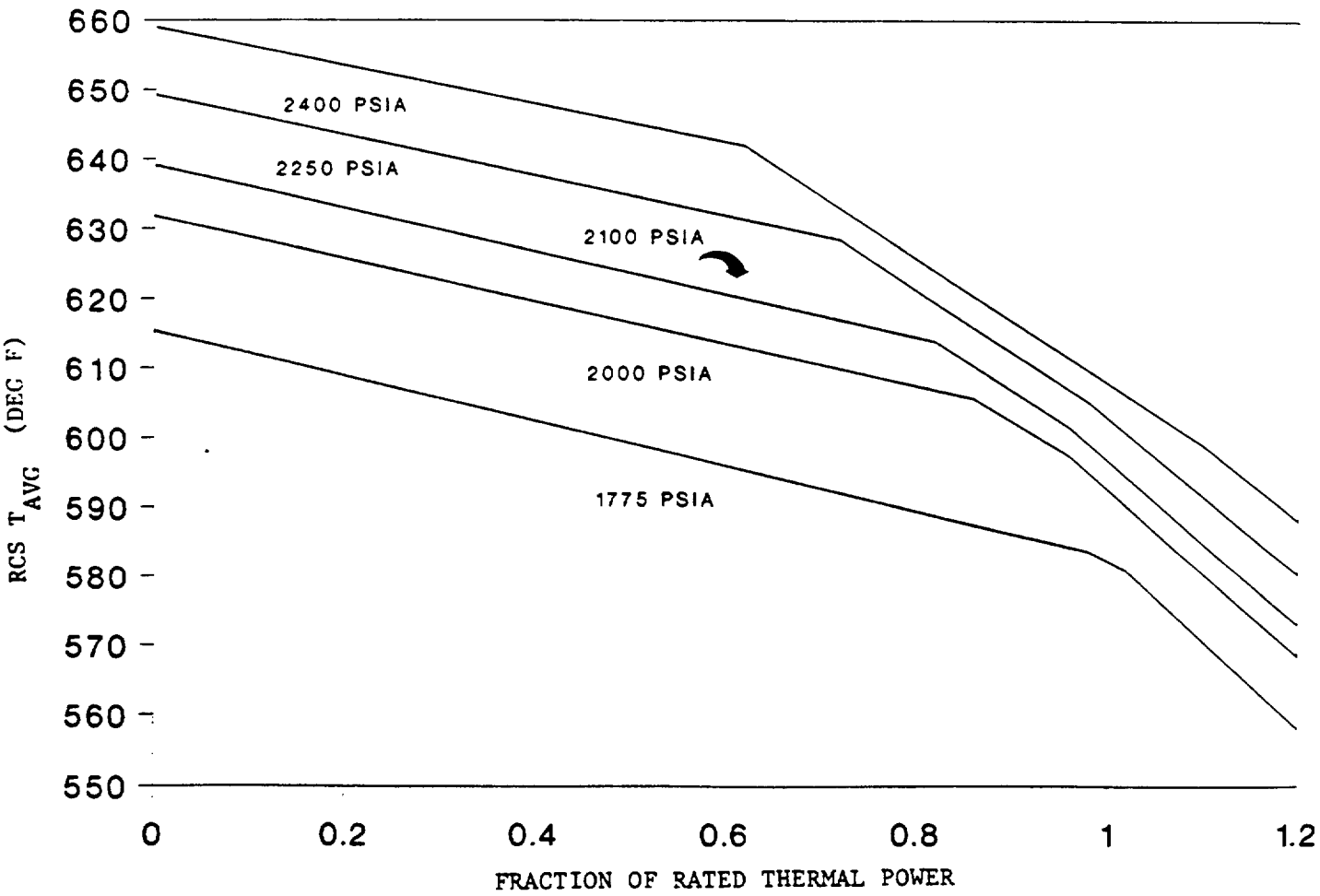


Figure 2.1-1 Reactor Core Safety Limits
Four Loops in Operation

## 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 - Less than or equal to 25% of RATED THERMAL POWER | Low Setpoint - Less than or equal to 26% of RATED THERMAL POWER |
| | High Setpoint - Less than or equal to 109% of RATED THERMAL POWER | High Setpoint - Less than or equal to 110% of RATED THERMAL POWER |
| 3. Power Range, Neutron Flux, High Positive Rate | Less than or equal to 5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds | Less than or equal to 5.5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds |
| 4. Power Range, Neutron Flux, High Negative Rate | Less than or equal to 5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds | Less than or equal to 5.5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds |
| 5. Intermediate Range, Neutron Flux | Less than or equal to 25% of RATED THERMAL POWER | Less than or equal to 30% of RATED THERMAL POWER |
| 6. Source Range, Neutron Flux | Less than or equal to $10^5$ counts per second | Less than or equal to 1.3 x $10^5$ counts per second |
| 7. Overtemperature Delta T | See Note 1 | See Note 3 |
| 8. Overpower Delta T | See Note 2 | See Note 4 |
| 9. Pressurizer Pressure -- Low | Greater than or equal to 1950 psig | Greater than or equal to 1940 psig |
| 10. Pressurizer Pressure -- High | Less than or equal to 2385 psig | Less than or equal to 2395 psig |
| 11. Pressurizer Water Level -- High | Less than or equal to 92% of instrument span | Less than or equal to 93% of instrument span |
| 12. Loss of Flow | Greater than or equal to 90% of design flow per loop* | Greater than or equal to 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

| FUNCTIONAL UNIT | TRIP SETPOINT | ALLOWABLE VALUES |
|---|---|---|
| 13. Steam Generator Water Level-Low-Low | Greater than or equal to 21% of narrow range instrument span - each steam generator | Greater than or equal to 19.2% of narrow range instrument span - each steam generator |
| 14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level | Less than or equal to 1.47 x $10^6$ lbs/hr of steam flow at RATED THERMAL POWER coincident with steam generator water level greater than or equal to 25% of narrow range instrument span - each steam generator | Less than or equal to 1.56 x $10^6$ lbs/hr of steam flow at RATED THERMAL POWER coincident with steam generator water level greater than or equal to 24% of narrow range instrument span - each steam generator |
| 15. Undervoltage - Reactor Coolant Pumps | Greater than or equal to 2905 volts - each bus | Greater than or equal to 2870 volts - each bus |
| 16. Underfrequency - Reactor Coolant Pumps | Greater than or equal to 57.5 Hz - each bus | Greater than or equal to 57.4 Hz - each bus |
| 17. Turbine Trip | | |
| A. Low Trip System Pressure | Greater than or equal to 58 psig | Greater than or equal to 57 psig |
| B. Turbine Stop Valve Closure | Greater than or equal to 1% open | Greater than or equal to 1% open |
| 18. Safety Injection Input from ESF | Not Applicable | Not Applicable |
| 19. Reactor Coolant Pump Breaker Position Trip | Not Applicable | Not Applicable |

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

Note 1:

Overtemperature $\Delta T \leq \Delta T_o \; [K_1 - K_2[(1 + \tau_1 S)/(1 + \tau_2 S)](T-T') + K_3(P-P') - f_1(\Delta I)]$

Where:

| | |
|---|---|
| $\Delta T_o$ | - Indicated $\Delta T$ at RATED THERMAL POWER |
| $T$ | - Average temperature, $^oF$ |
| $T'$ | - Indicated $T_{avg}$ at RATED THERMAL POWER less than or equal to 576.0 $^oF$ |
| $P$ | - Pressurizer Pressure, psig |
| $P'$ | - 2235 psig (indicated RCS nominal operating pressure) |
| $\frac{1 + \tau_1 S}{1 + \tau_2 S}$ | - The function generated by the lead-lag controller for $T_{avg}$ dynamic compensation |
| $\tau_1$, $\tau_2$ | - Time constants utilized in the lead-lag controller for $T_{avg}$; $\tau_1$ = 28 secs, $\tau_2$ = 4 secs. |
| $S$ | - Laplace transform operator |

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

4 Loops in Operation

K1 = 1.09

K2 = 0.01331

K3 = 0.00058

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 -33 percent and +6 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 -33 percent, the ΔT trip setpoint shall be automatically reduced by 3.5 percent of its value at RATED THERMAL POWER.

(iii) For each percent that the magnitude of $(q_t - q_b)$ exceeds +6 percent, the ΔT trip setpoint shall be automatically reduced by 1.0 percent of its value at RATED THERMAL POWER.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATIONS (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_o \; [K_4 - K_5[\tau_3 S/(1+\tau_3 S)]T - K_6[T-T''] - f_2(\Delta I)]$

Where:

| | |
|---|---|
| $\Delta T_o$ | = Indicated $\Delta T$ at rated power |
| $T$ | = Average temperature, $^oF$ |
| $T''$ | = Indicated $T_{avg}$ at RATED THERMAL POWER less than or equal to 576.0 $^oF$ |
| $K_4$ | = 1.08 |
| $K_5$ | = 0.02/$^oF$ for increasing average temperature and 0 for decreasing average temperature |
| $K_6$ | = 0.00197 for T greater than T"; $K_6$ = 0 for T less than or equal to T" |
| $\tau_3 S/(1+\tau_3 S)$ | = The function generated by the rate lag controller for $T_{avg}$ dynamic compensation |
| $\tau_3$ | = Time constant utilized in the rate lag controller for $T_{avg}$; $\tau_3$ = 10 secs. |
| $S$ | = Laplace transform operator |
| $f_2(\Delta I)$ | = 0.0 |

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 1.3 percent $\Delta T$ span.

Note 4: The channel's maximum trip point shall not exceed its computed trip point by more than 3.0 percent $\Delta T$ span.

## 3/4.1 REACTIVITY CONTROL SYSTEMS

### 3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - $T_{AVG}$ GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.6% Delta k/k.

APPLICABILITY: MODES 1, 2*, 3, and 4.

ACTION:

With the SHUTDOWN MARGIN less than 1.6% Delta k/k, immediately initiate and continue boration at greater than or equal to 10 gpm of a solution containing greater than or equal to 20,000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.6% Delta k/k:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s).

b. When in MODE 1 or MODE 2 with $K_{eff}$ greater than or equal to 1.0, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6.

c. When in MODE 2 with $K_{eff}$ less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit of Specification 3.1.3.6.

*See Special Test Exception 3.10.1

SURVEILLANCE REQUIREMENTS (Continued)

e. When in MODES 3 or 4, at least once per 24 hours by consideration of the following factors:

1. Reactor coolant system boron concentration,

2. Control rod position,

3. Reactor coolant system average temperature,

4. Fuel burnup based on gross thermal energy generation,

5. Xenon concentration, and

6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within plus or minus 1% Delta k/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

SHUTDOWN MARGIN - $T_{AVG}$ LESS THAN OR EQUAL TO 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 1.0% Delta k/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 1.0% Delta k/k, immediately initiate and continue boration at greater than or equal to 10 gpm of a solution containing greater than or equal to 20,000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.0% Delta k/k:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s).

b. At least once per 24 hours by consideration of the following factors:

   1. Reactor coolant system boron concentration,

   2. Control rod position,

   3. Reactor coolant system average temperature,

   4. Fuel burnup based on gross thermal energy generation,

   5. Xenon concentration, and

   6. Samarium concentration.

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BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.8 Each of the following borated water sources shall be OPERABLE:

a. A boric acid storage system and associated heat tracing with:

1. A minimum contained borated water volume of 7715 gallons,

2. Between 20,000 and 22,500 ppm of boron, and

3. A minimum solution temperature of 145°F.

b. The refueling water storage tank with:

1. A minimum contained borated water volume of 350,000 gallons of water,

2. Between 2400 and 2600 ppm of boron, and

3. A minimum solution temperature of 80°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the boric acid storage system inoperable, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% Delta k/k at 200°F; restore the boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 Each borated water source shall be demonstrated OPERABLE:

## ROD DROP TIME

### LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) rod drop time from the fully withdrawn position (specified in the COLR) shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. $T_{avg}$ greater than or equal to 541°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 AND 2

ACTION:

With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

### SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of full length rods shall be demonstrated through measurement prior to entering MODE 2:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. At least once per 18 months.

ACTION: (Continued)

b. THERMAL POWER shall not be increased above 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER unless the indicated AFD is within the target band and ACTION 2.a) 1), above has been satisfied.

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD has not been outside of the target band for more than 1 hour penalty deviation cumulative during the previous 24 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AXIAL FLUX DIFFERENCE shall be determined to be within its limits during POWER OPERATION above 15% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

1. At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

2. At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status if the AFD has been outside of the target band for any period of time in the previous 24 hours of operation.

b. Monitoring and logging the indicated AXIAL FLUX DIFFERENCE for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AXIAL FLUX DIFFERENCE Monitor Alarm is inoperable. The logged values of the indicated AXIAL FLUX DIFFERENCE shall be assumed to exist during the interval preceding each logging.

POWER DISTRIBUTION LIMITS

DNB AND Tavg OPERATING PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the following operational indicated limits:

a. DNB

| | |
|---|---|
| 1. Reactor Coolant System $T_{avg}$ | Less than or equal to 578.7°F* |
| 2. Pressurizer Pressure | Greater than or equal to 2200 psig*/** |
| 3. Reactor Coolant System Total Flow Rate | Greater than or equal to 366,400 gpm*** |

b. $T_{avg}$

| | |
|---|---|
| 1. Reactor Coolant System $T_{avg}$ | Greater than or equal to 543.9°F* |

APPLICABILITY: MODE 1

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the above parameters shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 The indicators used to determine RCS total flow shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

4.2.5.3 The RCS total flow rate shall be determined by a power balance around the steam generators at least once per 18 months.

4.2.5.4 The provisions of Specification 4.0.4 shall not apply to primary flow surveillances.

---

* Indicated average of at least three OPERABLE instrument loops.

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

*** Indicated value

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POWER DISTRIBUTION LIMITS

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 relationships:

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

- o CFQ is the $F_Q$ limit at RATED THERMAL POWER specified in the COLR for Westinghouse or Exxon fuel.
- o $F_Q(Z)$ is the measured hot channel factor, including a 3% manufacturing tolerance uncertainty and a 5% measurement uncertainty.
- o V(Z) is the function specified in the COLR.

- $F_p$ = 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 following 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 2 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 | Less than or equal to 0.5 seconds* |
| 3. Power Range, Neutron Flux, High Positive Rate | NOT APPLICABLE |
| 4. Power Range, Neutron Flux High Negative Rate | Less than or equal to 0.5 seconds* |
| 5. Intermediate Range, Neutron Flux | NOT APPLICABLE |
| 6. Source Range, Neutron Flux | NOT APPLICABLE |
| 7. Overtemperature Delta T | Less than or equal to 6.0 seconds* |
| 8. Overpower Delta T | NOT APPLICABLE |
| 9. Pressurizer Pressure--Low | Less than or equal to 2.0 seconds |
| 10. Pressurizer Pressure--High | Less than or equal to 2.0 seconds |
| 11. Pressurizer Water Level--High | Less than or equal to 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 3.3-2 (Continued)
REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

| FUNCTIONAL UNIT | RESPONSE TIME |
|---|---|
| 12. Loss of Flow - Single Loop (Above P-8) | Less than or equal to 1.0 seconds |
| 13. Loss of Flow - Two Loops (Above P-7 and below P-8) | Less than or equal to 1.0 seconds |
| 14. Steam Generator Water Level--Low-Low | Less than or equal to 2.0 seconds |
| 15. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level | NOT APPLICABLE |
| 16. Undervoltage-Reactor Coolant Pumps | Less than or equal to 1.5 seconds |
| 17. Underfrequency-Reactor Coolant Pumps | Less than or equal to 0.6 seconds |
| 18. Turbine Trip | |
| A. Low Fluid Oil Pressure | NOT APPLICABLE |
| B. Turbine Stop Valve | NOT APPLICABLE |
| 19. Safety Injection Input from ESF | NOT APPLICABLE |
| 20. Reactor Coolant Pump Breaker Position Trip | NOT APPLICABLE |

TABLE 3.3-3 (Continued)
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| Three Loops Operating | 1 $T_{avg}$/ operating loop | 1### $T_{avg}$ in any operating loop | 1 $T_{avg}$ in any two operating loops | 3## | 15 |
| e. Steam Line Pressure-Low | | | | | |
| Four Loops Operating | 1 pressure/ loop | 2 pressures any loops | 1 pressure any 3 loops | 1,2,3## | 14* |
| Three Loops Operating | 1 pressure/ operating loop | 1### pressure in any operating loop | 1 pressure in any 2 operating loops | 3## | 15 |
| 5. TURBINE TRIP & FEEDWATER ISOLATION | | | | | |
| a. Steam Generator Water Level--High-High | 3/loop | 2/loop in any operating loop | 2/loop in each operating loop | 1,2 and 3 | 14* |

TABLE 3.3-3 (Continued)
TABLE NOTATION

#Trip function may be bypassed in this MODE below P-11.

##Trip function may be bypassed in this MODE below P-12.

###The channel(s) associated with the protective functions derived from the out of service Reactor Coolant Loop shall be placed in the tripped mode.

####Manually trip all bistables which would be automatically tripped in the event pressure in the associated active loop were less than the pressure in the inactive loop. For example, if loop 1 is the inactive loop then the bistables which indicate low pressure in loops 2, 3, and 4 relative to loop 1 should be tripped.

*The provisions of Specification 3.0.4 are not applicable.

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | TRIP SETPOINTS | ALLOWABLE VALUES |
|---|---|---|
| 1. SAFETY INJECTION, TURBINE TRIP, FEEDWATER ISOLATION, AND MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS | | |
| a. Manual Initiation | Not Applicable | Not Applicable |
| b. Automatic Actuation Logic | Not Applicable | Not Applicable |
| c. Containment Pressure--High | Less than or equal to 1.1 psig | Less than or equal to 1.2 psig |
| d. Pressurizer Pressure--Low | Greater than or equal to 1900 psig | Greater than or equal to 1890 psig |
| e. Differential Pressure Between Steam Lines--High | Less than or equal to 100 psi | Less than or equal to 112 psi |
| f. Steam Line Pressure--Low | Greater than or equal to 600 psig steam line pressure | Greater than or equal to 585 psig steam line pressure |

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | TRIP SETPOINTS | ALLOWABLE VALUES |
|---|---|---|
| 2. CONTAINMENT SPRAY | | |
| a. Manual Initiation | Not Applicable | Not Applicable |
| b. Automatic Actuation Logic | Not Applicable | Not Applicable |
| c. Containment Pressure--High-High | Less than or equal to 2.9 psig | Less than or equal to 3.0 psig |
| 3. CONTAINMENT ISOLATION | | |
| a. Phase "A" Isolation | | |
| 1. Manual | Not Applicable | Not Applicable |
| 2. From Safety Injection Automatic Actuation Logic | Not Applicable | Not Applicable |
| b. Phase "B" Isolation | | |
| 1. Manual | Not Applicable | Not Applicable |
| 2. Automatic Actuation Logic | Not Applicable | Not Applicable |
| 3. Containment Pressure-High-High | Less than or equal to 2.9 psig | Less than or equal to 3.0 psig |
| c. Purge and Exhaust Isolation | | |
| 1. Manual | Not Applicable | Not Applicable |
| 2. Containment Radioactivity--High Train A (VRS-2101, ERS-2301, ERS-2305) | See Table 3.3-6 | Not Applicable |
| 3. Containment Radioactivity--High Train B (VRS-2201, ERS-2401, ERS-2405) | See Table 3.3-6 | Not Applicable |

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | TRIP SETPOINTS | ALLOWABLE VALUES |
|---|---|---|
| 4. STEAM LINE ISOLATION | | |
| a. Manual | Not Applicable | Not Applicable |
| b. Automatic Actuation Logic | Not Applicable | Not Applicable |
| c. Containment Pressure--High-High | Less than or equal to 2.9 psig | Less than or equal to 3.0 psig |
| d. Steam Flow in Two Steam Lines--High Coincident with $T_{avg}$--Low-Low | Less than or equal to a function defined as follows: A Delta-p corresponding to $1.6 \times 10^6$ lbs/hr steam flow between 0% and 20% load and then a Delta-p increasing linearly to a Delta-p corresponding to $4.5 \times 10^6$ lbs/hr at full load. | Less than or equal to a function defined as follows: A Delta-p corresponding to $1.75 \times 10^6$ lbs/hr steam flow between 0% and 20% load and then a Delta-p increasing linearly to a Delta-p corresponding to $4.55 \times 10^6$ lbs/hr at at full load. |
| | $T_{avg}$ greater than or equal to 541°F | $T_{avg}$ greater than or equal to 539°F |
| e. Steam Line Pressure--Low | Greater than or equal to 600 psig steam line pressure | Greater than or equal to 585 psig steam line pressure |
| 5. TURBINE TRIP AND FEEDWATER ISOLATION | | |
| a. Steam Generator Water Level--High-High | Less than or equal to 67% of narrow range instrument span each steam generator | Less than or equal to 68% of narrow range instrument span each steam generator |

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | TRIP SETPOINT | ALLOWABLE VALUES |
|---|---|---|
| 6. MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS | | |
| a. Steam Generator Water Level--Low-Low | Greater than or equal to 21% of narrow range instrument span each steam generator | Greater than or equal to 19.2% of narrow range instrument span each steam generator |
| b. 4 kV Bus Loss of Voltage | 3196 volts with a 2 second delay | 3280 ± 120 volts with a 2 + - 0.2 second delay |
| c. Safety Injection | Not Applicable | Not Applicable |
| d. Loss of Main Feedwater Pumps | Not Applicable | Not Applicable |
| 7. TURBINE DRIVEN AUXILIATY FEEDWATER PUMPS | | |
| a. Steam Generator Water Level--Low-Low | Greater than or equal to 21% of narrow range instrument span each steam generator | Greater than or equal to 19.2% of narrow range instrument span each steam generator |
| b. Reactor Coolant Pump Bus Undervoltage | Greater than or equal to 2750 Volts--each bus | Greater than or equal 2725 Volts--each bus |
| 8. LOSS OF POWER | | |
| a. 4 kV Bus Loss of Voltage | 3196 volts with a 2 second delay | 3280 ± 120 volts with a 2 + - 0.2 second delay |
| b. 4 kV Bus Degraded Voltage | 3596 volts with a 2.0 minute time delay | 3638 ± 60 volts with a 2.0 minute + - 6 second time delay |

## TABLE 3.3-5

## ENGINEERED SAFETY FEATURES RESPONSE TIMES

| INITIATING SIGNAL AND FUNCTION | RESPONSE TIME IN SECONDS |
|---|---|
| 1. Manual | |
| a. Safety Injection (ECCS) | Not Applicable |
| Feedwater Isolation | Not Applicable |
| Reactor Trip (SI) | Not Applicable |
| Containment Isolation-Phase "A" | Not Applicable |
| Containment Purge and Exhaust Isolation | Not Applicable |
| Auxiliary Feedwater Pumps | Not Applicable |
| Essential Service Water System | Not Applicable |
| Containment Air Recirculation Fan | Not Applicable |
| b. Containment Spray | Not Applicable |
| Containment Isolation-Phase "B" | Not Applicable |
| Containment Purge and Exhaust Isolation | Not Applicable |
| c. Containment Isolation-Phase "A" | Not Applicable |
| Containment Purge and Exhaust Isolation | Not Applicable |
| d. Steam Line Isolation | Not Applicable |
| 2. Containment Pressure-High | |
| a. Safety Injection (ECCS) | Less than or equal to 27.0* |
| b. Reactor Trip (from SI) | Less than or equal to 3.0 |
| c. Feedwater Isolation | Less than or equal to 8.0 |
| d. Containment Isolation-Phase "A" | Not Applicable |
| e. Containment Purge and Exhaust Isolation | Not Applicable |
| f. Auxiliary Feedwater Pumps | Not Applicable |
| g. Essential Service Water System | Not Applicable |

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

| INITIATING SIGNAL AND FUNCTION | RESPONSE TIME IN SECONDS |
|---|---|
| 6. Steam Line Pressure--Low | |
| a. Safety Injection (ECCS) | Less than or equal to 12.0#/24.0## |
| b. Reactor Trip (from SI) | Less than or equal to 2.0 |
| c. Feedwater Isolation | Less than or equal to 8.0 |
| d. Containment Isolation-Phase "A" | Less than or equal to 18.0#/28.0## |
| e. Containment Purge and Exhaust Isolation | Not Applicable |
| f. Motor Driven Auxiliary Feedwater Pumps | Less than or equal to 60.0 |
| g. Essential Service Water System | Less than or equal to 14.0#/48.0## |
| h. Steam Line Isolation | Less than or equal to 8.0 |
| 7. Containment Pressure--High-High | |
| a. Containment Spray | Less than or equal to 45.0 |
| b. Containment Isolation-Phase "B" | Not Applicable |
| c. Steam Line Isolation | Less than or equal to 7.0 |
| d. Containment Air Recirculation Fan | Less than or equal to 600.0 |
| 8. Steam Generator Water Level--High-High | |
| a. Turbine Trip | Less than or equal to 2.5 |
| b. Feedwater Isolation | Less than or equal to 11.0 |
| 9. Steam Generator Water Level--Low-Low | |
| a. Motor Driven Auxiliary Feedwater Pumps | Less than or equal to 60.0 |
| b. Turbine Driven Auxiliary Feedwater Pump | Less than or equal to 60.0 |
| 10. 4160 volt Emergency Bus Loss of Voltage | |
| a. Motor Driven Auxiliary Feedwater Pumps | Less than or equal to 60.0 |
| 11. Loss of Main Feedwater Pumps | |
| a. Motor Driven Auxiliary Feedwater Pumps | Less than or equal to 60.0 |
| 12. Reactor Coolant Pump Bus Undervoltage | |
| a. Turbine Driven Auxiliary Feedwater Pump | Less than or equal to 60.0 |

TABLE 4.3-2 (Continued)
ENGINEERED SAFETY FEATURED ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

| FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | CHANNEL FUNCTIONAL TEST | MODES IN WHICH SURVEILLANCE REQUIRED |
|---|---|---|---|---|
| 1. SAFETY INJECTION, TURBINE TRIP, FEEDWATER ISOLATION, AND MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS | | | | |
| a. Manual Initiation | N.A. | N.A. | M(1) | 1,2,3,4 |
| b. Automatic Actuation Logic | N.A. | N.A. | M(2) | 1,2,3,4 |
| c. Containment Pressure-High | S | R | M(3) | 1,2,3 |
| d. Pressurizer Pressure-Low | S | R | M | 1,2,3 |
| e. Differential Pressure Between Steam Lines--High | S | R | M | 1,2,3 |
| f. Steam Line Pressure--Low | S | R | M | 1,2,3 |
| 2. CONTAINMENT SPRAY | | | | |
| a. Manual Initiation | N.A. | N.A. | M(1) | 1,2,3,4 |
| b. Automatic Actuation Logic | N.A. | N.A. | M(2) | 1,2,3,4 |
| c. Containment Pressure-High-High | S | R | M(3) | 1,2,3 |
| 3. CONTAINMENT ISOLATION | | | | |
| a. Phase "A" Isolation | | | | |
| 1) Manual | N.A. | N.A. | M(1) | 1,2,3,4 |
| 2) From Safety Injection Automatic Actuation Logic | N.A. | N.A. | M(2) | 1,2,3,4 |
| b. Phase "B" Isolation | | | | |
| 1) Manual | N.A. | N.A. | M(1) | 1,2,3,4 |
| 2) Automatic Actuation Logic | N.A. | N.A. | M(2) | 1,2,3,4 |
| 3) Containment Pressure-High-High | S | R | M(3) | 1,2,3 |

TABLE 4.3-2 (Continued)
ENGINEERED SAFETY FEATURED ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

| FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | CHANNEL FUNCTIONAL TEST | MODES IN WHICH SURVEILLANCE REQUIRED |
|---|---|---|---|---|
| c. Purge and Exhaust Isolation | | | | |
| 1) Manual | N.A. | N.A. | M(1) | 1,2,3,4 |
| 2) Containment Radio-activity-High | S | R | M | 1,2,3,4 |
| 4. STEAM LINE ISOLATION | | | | |
| a. Manual | N.A. | N.A. | M(1) | 1,2,3 |
| b. Automatic Actuation Logic | N.A. | N.A. | M(2) | 1,2,3 |
| c. Containment Pressure--High-High | S | R | M(3) | 1,2,3 |
| d. Steam Flow in Two Steam Lines--High Coincident with Tavg--Low-Low | S | R | M | 1,2,3 |
| e. Steam Line Pressure--Low | S | R | M | 1,2,3 |
| 5. TURBINE TRIP AND FEEDWATER ISOLATION | | | | |
| a. Steam Generator Water Level--High-High | S | R | M | 1,2,3 |
| 6. MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS | | | | |
| a. Steam Generator Water Level--Low-Low | S | R | M | 1,2,3 |
| b. 4 kV Bus Loss of Voltage | S | R | M | 1,2,3 |
| c. Safety Injection | N.A. | N.A. | M(2) | 1,2,3 |
| d. Loss of Main Feed Pumps | N.A. | N.A. | R | 1,2 |

TABLE 4.3-2 (Continued)
ENGINEERED SAFETY FEATURED ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

| FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | CHANNEL FUNCTIONAL TEST | MODES IN WHICH SURVEILLANCE REQUIRED |
|---|---|---|---|---|
| 7. TURBINE DRIVEN AUXILIARY FEEDWATER PUMP | | | | |
| a. Steam Generator Water Level--Low-low | S | R | M | 1,2,3 |
| b. Reactor Coolant Pump Bus Undervoltage | N.A. | R | M | 1,2,3 |
| 8. LOSS OF POWER | | | | |
| a. 4 kv Bus Loss of Voltage | S | R | M | 1,2,3,4 |
| b. 4 kv Bus Degraded Voltage | S | R | M | 1,2,3,4 |

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TABLE 4.3-2 (Continued)

TABLE NOTATION

(1) Manual actuation switches shall be tested at least once per 18 months during shutdown. All other circuitry associated with manual safeguards actuation shall receive a CHANNEL FUNCTIONAL TEST at least once per 31 days.

(2) Each train or logic channel shall be tested at least every other 31 days.

(3) The CHANNEL FUNCTIONAL TEST shall include exercising the transmitter by applying either a vacuum or pressure to the appropriate side of the transmitter.

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 limit 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.

## 3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

### 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 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 12 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 12 hours.

### SURVEILLANCE REQUIREMENTS

4.5.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying the contained borated water volume and nitrogen cover-pressure in the tanks, and

2. Verifying that each accumulator isolation valve is open.

---

*Pressurizer Pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS (Continued)

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 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:

| | | |
|---|---|---|
| 1. | Centrifugal charging pump | Greater than or equal to 2405 psig |
| 2. | Safety Injection pump | Greater than or equal to 1445 psig |
| 3. | Residual heat removal pump | Greater than or equal to 195 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.

2. At least once per 18 months.

| Boron Injection Throttle Valves | Safety Injection Throttle Valves |
|---|---|
| Valve Number | Valve Number |
| 1. 2-SI-141 L1 | 1. 2-SI-121 N |
| 2. 2-SI-141 L2 | 2. 2-SI-121 S |
| 3. 2-SI-141 L3 | |
| 4. 2-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 greater than or equal to 300 gpm |
| Loop 2 Boron Injection Flow 117.5 gpm | Loop 2 and 3 Cold Leg Flow greater than or equal to 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 into each loop. Under these conditions there is zero mini-flow 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 through the three most conservative (lowest flow) branch lines must not be less than 300 gpm,

d) the charging pump discharge resistance (2.31*Pd/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).

## 2.1 SAFETY LIMITS

## BASES

### 2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and reactor coolant temperature and pressure have been related to DNB through the WRB-2 correlation and W-3 correlation for conditions outside the range of WRB-2. The DNB correlations have been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, and is indicative of the margin to DNB.

The DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the WRB-2 correlation for Vantage-5 fuel, and the W-3 correlation for ANF fuel and conditions which fall outside the range of applicability of the WRB-2). The correlation DNBR limits are established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit (1.17 for WRB-2 and 1.3 for the W-3).

In meeting the DNB design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are statistically combined with the DNBR correlation statistics such that there is at least a 95 percent probability with a 95 percent confidence level that the minimum DNBR for the limiting rod is greater than or equal to a calculated design limit DNBR. The uncertainties in the above plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the DNBR correlation statistics, establishes a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties. For Cook Nuclear Plant Unit 2, the design DNBR values are 1.23 and 1.22 for Vantage-5 fuel typical and thimble cells, respectively, and 1.39 and 1.36 for typical and thimble cells for the ANF fuel. In addition, margin has been maintained in both fuel types by performing safety analyses to a safety analysis limit DNBR. The margin between the design and safety analysis limit DNBR is used to offset known DNBR penalties (i.e., transition core penalties, rod bow, etc.) and provide DNBR margin for operating and design flexibility.

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

BASES

The Power Range Negative Rate trip provides protection to ensure that the calculated DNBR is maintained above the design DNBR value for multiple control rod drop accidents. The analysis of a single control rod drop (or some multiple rod drops) accident indicates a return to full power may be initiated by the automatic control system in response to a continued full power turbine load demand or by the negative moderator temperature feedback.

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. This reactor trip limit is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are more severe 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.

BASES

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. No credit was taken for operation of this trip in the accident analyses; however, the functional capability of the Overpower Delta T trip 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 control rod assembly bank withdrawal at power event.

BASES

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump bus trips provide reactor core protection against DNB as a result of loss of voltage or underfrequency to more than one reactor coolant pump. The specified set points assure a reactor trip signal is generated before the low flow trip set point is reached. Time delays are incorporated in the underfrequency and undervoltage trips to prevent spurious reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed 1.2 seconds. For underfrequency, the delay is set so that the time required for a signal to reach the reactor trip breakers after the underfrequency trip set point is reached shall not exceed 0.3 seconds. The total response times for these functional units include an additional 0.3 seconds for trip breaker operation and CRDM release.

Turbine Trip

A Turbine Trip causes a direct reactor trip when operating above P-7. Each of the turbine trips provide turbine protection and reduce the severity of the ensuing transient. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the specified trip settings is required to enhance the overall reliability of the Reactor Protection System.

## 3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

### 3/4.1.1 BORATION CONTROL

### 3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS $T_{avg}$. The most restrictive condition occurs at EOL, with $T_{avg}$ at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.6% Delta k/k is initially required to control the reactivity transient and automatic ESF is assumed to be available.

With $T_{avg}$ less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% Delta k/k SHUTDOWN MARGIN provides adequate protection for this event.

The SHUTDOWN MARGIN requirements are based upon the limiting conditions described above and are consistent with FSAR safety analysis assumptions.

### 3/4.1.1.3 BORON DILUTION

A minimum flow rate of at least 2000 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 2000 GPM will circulate an equivalent Reactor Coolant System volume of 12,612 cubic feet in approximately 30 minutes. The reactivity change rate associated with boron reductions will therefore be within the capability for operator recognition and control.

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

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

The boration capability of either system is sufficient to provide the required SHUTDOWN MARGIN from expected operating conditions after xenon decay and cooldown to 200°F. The maximum expected boration capability usable volume requirement is 7715 gallons of 20,000 ppm borated water from the boric acid storage tanks or 160,122 gallons of borated water from the refueling water storage tank. The required RWST volume is based on an assumed boron concentration of 2400 ppm. The minimum RWST boron concentration required by the post-LOCA long-term cooling analysis is 2400 ppm. The minimum contained RWST volume is based on ECCS considerations. See Section B 3/4.5.5.

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

The boron capability required below 200°F is sufficient to provide the required MODE 5 SHUTDOWN MARGIN after xenon decay and cooldown from 200°F to 140°F. This condition requires usable volumes of either 4300 gallons of 20,000 ppm borated water from the boric acid storage tanks or 90,000 gallons of borated water from the refueling water storage tank. The value for the boric acid storage tank volume includes sufficient boric acid to borate to 2190 ppm. The required RWST volume is based on an assumed boron concentration of 2400 ppm. The minimum RWST boron concentration required by the post-LOCA long-term cooling analysis is 2400 ppm.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.6 and 9.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The OPERABILITY of boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

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

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

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

The boron capability required below 200°F is sufficient to provide the required MODE 5 SHUTDOWN MARGIN after xenon decay and cooldown from 200°F to 140°F. This condition requires usable volumes of either 4300 gallons of 20,000 ppm borated water from the boric acid storage tanks or 90,000 gallons of borated water from the refueling water storage tank. The value for the boric acid storage tank volume includes sufficient boric acid to borate to 2190 ppm. The required RWST volume is based on an assumed boron concentration of 2400 ppm. The minimum RWST boron concentration required by the post-LOCA long-term cooling analysis is 2400 ppm.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.6 and 9.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The OPERABILITY of boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

BORATION SYSTEMS (Continued)

The charging flowpath of Unit 2 required for Unit 1 shutdown support ensures that flow is available to Unit 1 and addresses the requirements of 10 CFR 50 Appendix R. The flowpath consists of a charging pump powered from an electrical bus and associated water supplies and delivery system. Fire watches posted in the affected opposite unit areas (i.e., Unit 1 areas requiring use of the Unit 2 charging system in the event of a fire) may serve as the equivalent shutdown capability specified in the action statements of Specification 3.1.2.3. In the affected areas, either establish continuous fire watches or verify the OPERABILITY of fire detectors per Specification 4.3.3.7 and establish hourly fire watch patrols. The required opposite unit equipment along with the surveillance requirements necessary to ensure that this equipment is capable of fulfilling its intended Appendix R alternate safe shutdown function have been established and are included in a plant procedure. An additional plant procedure details how the above noted fire watches will be implemented.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

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

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors or a restriction in THERMAL POWER; either of these restrictions provide assurance of fuel rod integrity during continued operation. In addition, those accident analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

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

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

## 3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, and nuclear enthalpy rise hot channel factor ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2.1, 4.2.2.2, 4.2.3, 4.2.6.1 and 4.2.6.2. This periodic surveillance is sufficient to ensure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than plus or minus 12 steps from the group demand position.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6.

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits as specified in the COLR provided conditions a. through d. above are maintained. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. The form of this relaxation for DNBR limits is discussed in Section 2.1.1 of this basis.

When an $F_Q$ measurement is taken, both experimental error and manufacturing tolerance must be allowed for. 5% is the appropriate allowance on $F_Q$ for a full core map taken with the incore detector flux mapping system and 3% in the appropriate allowance for manufacturing tolerance.

BASES: (Continued)

When RCS flow rate and $F^N_{\Delta H}$ are measured, no additional allowances are necessary prior to comparison with the limits of Specification 3.2.3. Measurement errors of 2.1% for RCS flow total flow rate and 4% for $F^N_{\Delta H}$ have been allowed for in determination of the design DNBR value and in the determination of the LOCA/ECCS limit.

Margin between the safety analysis DNBRs and the design limit DNBRs is maintained. (Safety analyses DNBRs: 1.69 and 1.61 for the Vantage 5 typical and thimble cells, respectively, and 1.43 and 1.40 for the ANF fuel typical and thimble cells. Design limit DNBRs: 1.23 and 1.22 for the Vantage 5 typical and thimble cells, respectively, and 1.39 and 1.36 for the ANF fuel typical and thimble cells.) A fraction of this margin is utilized to accomodate applicable transition core penalties and the appropriate fuel rod bow DNBR penalty for the Vantage 5 fuel (equal to 1.3% per WCAP-8691, Rev. 1). The remainder of the margin between design and safety analysis DNBR limits can be used for plant design flexibility.

## 3/4.2 POWER DISTIBUTION LIMITS
## BASES

### 3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02 at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts.

The two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on $F_Q$ is reinstated by reducing the power by 3 percent from RATED THERMAL POWER for each percent of tilt in excess of 1.0.

### 3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters ensure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The $T_{avg}$ less than or equal to 578.7°F and pressurizer pressure greater than or equal to 2200 psig are consistent with the UFSAR assumptions and have been analytically demonstrated adequate to maintain the core at or above the design DNBR throughout each analyzed transient with allowance for measurement uncertainty. The $T_{avg}$ greater than or equal to 543.9°F is conservative to a safety analysis performed to demonstrate that the plant may operate on a linear control program where the analytical limit of $T_{avg}$ at 100% RATED THERMAL POWER may range from 541.4°F to 580.1°F. The limit of 543.9°F contains a margin of 1.1°F. The core may be operated with indicated vessel average temperature at any value between the upper and lower limits. Pressurizer pressure is limited to a single nominal setpoint, with the lower limit of the indicated value setpoint set forth in the specifications. The T/S value was selected for consistency with Unit 1 and contains a margin of 6 psi. The limits are consistent with the UFSAR assumptions and have been analytically demonstrated to be adequate to maintain the core at or above the applicable design limit DNBR values for each fuel type (which are listed in the bases for Section 2.1.1) throughout each analyzed transient.

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 12-hour surveillance of the RCS flow measurement is adequate to detect flow degradation. The CHANNEL CALIBRATION performed after refueling ensures the accuracy of the shiftly flow measurement. The total flow is measured after each refueling based on a secondary side calorimetric and measurements of primary loop temperatures.

3/4.2.6 ALLOWABLE POWER LEVEL - APL

Constant Axial Offset Control (CAOC) operation manages core power distributions such that Technical Specification limits on $F_Q(Z)$ are not violated during normal operation and limits on MDNBR are not violated during steady-state, load-follow, and anticipated transients. The V(Z) factor given in the Peaking Factor Limit Report and applied by the Technical Specifications provides the means for predicting the maximum $F_Q(Z)$ distribution anticipated during operation using CAOC taking into account the incore measured equilibrium power distribution. A comparison of the maximum $F_Q(Z)$ with the Technical Specification limit determines the power level (APL) below which the Technical Specification limit can be protected by CAOC. This comparison is done by calculating APL, as defined in Specification 3.2.6.

## 3/4.7.1 TURBINE CYCLE

### 3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% of its design pressure of 1085 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity of all safety valves on all of the steam lines is 17,153,800 lbs/hr which is at least 105 percent of the maximum secondary steam flow rate at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are derived on the following bases:

For 4 loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times (109)$$

Where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER

V = maximum number of inoperable safety valves per steam line

X = total relieving capacity of all safety valves per steam line in lbs./hours = 4,288,450

Y = maximum relieving capacity of any one safety valve in lbs./hour = 857,690

109 = Power Range Neutron Flux-High Trip Setpoint for 4 loop operation

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 148 TO FACILITY OPERATING LICENSE NO. DPR-58

AND AMENDMENT NO. 134 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNITS NOS. 1 AND 2

DOCKETS NOS. 50-315 AND 50-316

## 1.0 INTRODUCTION

By letter dated February 6, 1990 (Ref. 1), Indiana Michigan Power Company (the licensee) proposed changes to the Technical Specifications (TS) for the Donald C. Cook Nuclear Plant Unit No. 2. Additional information was provided in the letter of May 29, 1990 (Ref. 2) which also included a revised page 2-2 for Unit 2, correcting a typographical error from the earlier submittal. The proposed changes would modify technical specifications for Unit 2 Cycle 8 to allow for a transition to Westinghouse 17x17 VANTAGE 5 fuel as a replacement for 17x17 Advanced Nuclear Fuels Corporation (ANF) fuel. The majority of the proposed TS changes are related to the transition to Westinghouse fuel and reload analysis methodology. Additional information on the radiological consequences of the locked-rotor analysis for the reactor coolant pumps was provided by letter dated July 23, 1990. Neither the May 29, 1990 letter nor the July 23, 1990 letter changed the technical content of the proposed Technical Specification, nor did they change the staff's proposed no significant hazards consideration determination.

Certain related Unit 1 TS changes were also proposed. These changes were made where justification for a proposed Unit 1 TS change is essentially identical to a justification for a similar change to the Unit 2 TS. This is to maintain consistent TSs between both units.

In parallel with this submittal the licensee submitted proposed TS changes for Unit 2 in response to Generic Letter 88-16, "Removal of Cycle Specific Parameter Limits from Technical Specifications." A Core Operating Limits Report (COLR) was submitted for Unit 2 which covers TSs used for: (1) moderator temperature coefficient (MTC), (2) 300 ppm MTC surveillance acceptance criterion, (3) all rods out (ARO) shutdown rod position and control rod insertion limits, (4) axial flux difference allowable deviation, and axial flux difference target band, (5) $F_Q$ and K(Z) and, (6) $F_{\Delta H}^N$ and $F_{\Delta H}^N$ slope. The staff has approved the implementation of the COLR in accordance with the requirements of Generic Letter 88-16 (Ref. 3). The staff's review is contained in the Safety Evaluation supporting Amendment No. 122 to Facility Operating License No. DPR-74 for Unit 2 dated May 23, 1990.

By letter dated May 11, 1990 (Ref. 4), the licensee also made application to revise the TS for Unit 1 by modifying TS Table 3.3-2, "Reactor Trip System Response Times," to make it consistent with changes proposed for Unit 2 in Reference 1.

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The licensee proposed a Tavg window mode of operation over a range of primary temperatures and requested TS upper and lower limits of 543.9°F and 578.7°F, respectively. A similar Tavg window was previously addressed for Unit 1. Also, a lower TS pressure limit of 2200 psig was proposed.

The licensee also proposed removal of transition mode Technical Specifications associated with the Unit 2 Cycle 6 reanalysis. These were incorporated in the Unit 2 Technical Specifications by Amendment No. 82 to Facility Operating License No. DPR-74. They resulted from a review of abnormal operating occurrences and postulated accidents in the transition, shutdown, and refueling modes of operation. The Cycle 6 analysis, performed after the licensee switched fuel vendors from Westinghouse to ANF, included four additional events not analyzed in the original Final Safety Analysis Report (FSAR) and considered three other events that were bounded by the FSAR analyses.

In a meeting held on June 12, 1989, the staff and licensee met with Westinghouse to discuss proposed Cycle 8 Technical Specifications changes. The staff determined that the seven additional events analyzed for Cycle 6 and currently discussed in Section 14 of the Updated Final Safety Analysis Report (UFSAR) are not part of the licensing basis and were not required to be analyzed for Cycle 8. The staff's acceptance of this position is documented in a letter dated August 3, 1989.

The licensee proposes to remove the requirements associated with the Cycle 6 transitional mode analyses from the Technical Specifications. These requirements will be implemented instead by administrative controls. The staff finds this proposal to be acceptable.

The licensee also proposes to make a change to the Unit 2 Technical Specifications to achieve consistency with Unit 1. Specifically, the licensee proposes to increase the maximum pressurizer level to 92% of span. This change was approved for Unit 1 by Amendment No. 126 to Facility Operating License No. DPR-58 dated June 9, 1989. The staff finds this change to be acceptable. Our basis for acceptance is contained in the Safety Evaluation supporting Amendment No. 126 to Unit 1.

The Donald C. Cook Nuclear Plant Unit 2 is currently operating in Cycle 7 with ANF fuel in the core. Beginning with Cycle 8, the licensee plans to refuel and operate with the Westinghouse VANTAGE 5 improved fuel design. As a result, future transition core loadings will range from approximately 40 percent VANTAGE 5 and 60 percent ANF to eventually an all VANTAGE 5 fueled core. The VANTAGE 5 fuel assembly was designed as a modification to the current Westinghouse Optimized Fuel Assembly (OFA) design (Ref. 5). The VANTAGE 5 features were previously reviewed and approved by NRC in the Westinghouse topical report WCAP-10444-P-A, "Reference Core Report VANTAGE 5 Fuel Assembly" (Ref. 6). The features of the VANTAGE 5 fuel assembly include Intermediate Flow Mixer Grids, Reconstitutable Top Nozzles and Debris Filter Bottom Nozzles (DFBNs). In addition, the VANTAGE 5 fuel uses Integral Fuel Burnable Absorber and Axial Blanket design features and is designed for extended burnup using the design basis in Reference 7. The Technical Specification changes for use of VANTAGE 5 fuel include (1) use of the Westinghouse WRB-2 DNBR correlation, and (2) a VANTAGE 5 (V-5) core evaluation/analysis, which were performed to support an

uprate to a core thermal power level of 3588 MWt unless specifically indicated. However, Cycle 8 will operate at the currently licensed core power of 3411 MWt. The following assumptions were made in the safety evaluation: (1) a full power $F\Delta_H^N$ of 1.62 (with uncertainties) for the VANTAGE 5 fuel and 1.55 for the ANF fuel; (2) maximum $F_Q$ of 2.22 for VANTAGE 5 fuel and 2.10 for ANF fuel; and (3) a 15 percent peak and 10 percent average steam generator tube plugging level.

The approved Westinghouse Revised Thermal Design Procedure (RTDP) was used in the DNB analyses of both VANTAGE 5 and ANF fuel assemblies for all DNB-related accidents, excluding transients such as the steamline break where RTDP methodology is not applicable. For such transients, standard DNB design methods are used. The WRB-2 DNBR correlation was used in the VANTAGE 5 DNB analyses. The ANF fuel was analyzed by using the W-3 DNB correlation.

The standard Westinghouse reload design methods described in Reference 8 were used by the licensee for this evaluation. The licensee presented results of the Mechanical, Nuclear, Thermal and Hydraulic and Accidents Evaluations in Reference 1. Also included were the Technical Specification changes for transition to VANTAGE 5 fuel. The licensee reviewed all the non-LOCA accidents to address any impact from the V-5 fuel. The V-5 design features that were considered were:

Intermediate Flow Mixer (IFM) Grids

Axial Blankets

Integral Fuel Burnable Absorbers (IFBAs)

Debris Filter Bottom Nozzle

Reconstitutable Top Nozzle

## 2.0 EVALUATION

### 2.1 Mechanical Design

The VANTAGE 5 fuel assembly was designed to be compatible with the Westinghouse designed LOPAR and Optimized Fuel Assemblies (OFA), reactor internals interfaces, the fuel handling equipment, and refueling equipment.

The VANTAGE 5 design dimensions are essentially equivalent to the ANF 17x17 design from an exterior assembly envelope and reactor internals standpoint. The design basis and design limits for VANTAGE 5 are essentially the same as those for the Westinghouse LOPAR design.

The significant new mechanical features of the VANTAGE 5 design relative to the current ANF 17x17 fuel assembly include the following: (a) integral fuel burnable absorber, (b) intermediate flow mixer grids, (c) reconstitutable top nozzle, (d) extended burnup capability, and (e) axial blankets. Also, the VANTAGE 5 fuel design for Cook Nuclear Plant Unit 2 Cycle 8 includes the debris filter bottom (DFBN) nozzle which reduces the possibility of fuel rod damage due to debris-induced fretting. The DFBN is structurally and hydraulically equivalent to the existing low profile bottom nozzle.

### 2.1.1 Fuel Rod Performance

Fuel rod design evaluations for VANTAGE 5 fuel were performed using approved models in References 9 and 10 and approved extended burnup design methods in Reference 11 to demonstrate that the design bases are satisfied. Fuel performance evaluations are completed for each fuel region to demonstrate that the design criteria is satisfied for all fuel rod types under planned operating conditions.

### 2.1.2 Grid and Guide Thimble Assemblies

The design bases and evaluations of the grid and guide thimble assemblies were provided in Reference 6 which was approved by the staff. The VANTAGE 5 thimble rods provide adequate diametrical clearance for the control rods. For accident analyses, a 2.7-second scram time to the dashpot is used for the VANTAGE 5 assembly. There is a 0.5 second greater drop time for VANTAGE 5 fuel as compared to LOPAR due to the larger fuel assembly pressure drop attributed to the VANTAGE 5 IFM grids. The increase in pressure drop results in increased Rod Cluster Control Assembly (RCCA) resistance during rod drops. The Technical Specification for scram time has therefore been increased by 0.5 second to 2.7 seconds. This is consistent with the value used in the accident analysis.

### 2.1.3 Mechanical Compatibility of Fuel Assemblies

Based on the evaluation of design differences in Reference 4, it is concluded that VANTAGE 5 is mechanically compatible with both ANF and Westinghouse LOPAR fuel assemblies. The VANTAGE 5 fuel rod mechanical design bases remain unchanged from the Westinghouse LOPAR fuel assemblies used previously in the Cook Nuclear Plant Unit 2.

### 2.1.4 Rod Bow

Rod bow in the VANTAGE 5 fuel rods containing IFBAs is not expected to differ in magnitude or frequency from that currently observed in both LOPAR and OFA fuel rods under similar operating conditions. The VANTAGE 5 fuel rod bow magnitudes are predicted to be bounded by the LOPAR assembly rod bow data. The approved methodology for comparing rod bow for different assembly designs is given in Reference 11.

### 2.1.5 Fuel Rod Wear

Fuel rod wear is dependent on both the support provided by the fuel assembly and the flow environment to which it is subjected. Because the VANTAGE 5 fuel assembly has a different guide thimble tube diameter than the ANF fuel and IFM grids, an unequal pressure distribution results between the ANF and VANTAGE 5 fuel assembly designs. Evaluations were performed, taking into account the hardware differences between ANF 17x17 and VANTAGE 5 design, to evaluate the hydraulic compatibility of the two designs. The evaluations demonstrated that the VANTAGE 5 and ANF fuel were hydraulically compatible as each was shown to be compatible with OFA fuel. Also direct analyses were made to show hydraulic compatibility between the VANTAGE 5 and ANF fuel.

VANTAGE 5 fuel rod wear predictions were extrapolated from a full scale hydraulic test of a VANTAGE 5 assembly adjacent to a 17x17 OFA assembly. Vibration test results indicated that crossflow effects produced by this fuel assembly combination would have the most detrimental effect on fuel rod wear. From results of wear inspection and analysis, it was concluded that the VANTAGE 5 fuel rod wear would be less than the maximum wear depth established in Reference 12 for the 17x17 OFA at end-of-life condition.

### 2.1.6 Seismic/LOCA Impact on Fuel Assemblies

An evaluation of the VANTAGE 5 fuel assembly structural integrity considering the lateral effect of LOCA and seismic loading was performed by the licensee. The VANTAGE 5 fuel assembly is structurally equivalent to the LOPAR and ANF fuel designs. Based on the grid load tests, the 17x17 VANTAGE Zircaloy-4 is capable of maintaining core coolable geometry under design basis earthquake and asymmetric pipe rupture transients in either all VANTAGE 5 or transition core operations. The grids of either fuel type will not buckle due to combined impact loads of seismic and LOCA events. The stresses in the fuel assembly components resulting from seismic and LOCA induced deflections were found to be within acceptable limits.

### 2.1.7 Core Components

The core components for Cook Nuclear Plant Unit 2 are designed to be compatible with both LOPAR and VANTAGE 5 fuel assemblies. The thimble plugs included in the plugging devices for Cook Nuclear Plant Unit 2 have been designed to be compatible with the LOPAR, ANF and VANTAGE 5 designs.

## 2.2 Nuclear Design

Evaluations for transition and equilibrium cycle VANTAGE 5 cores in general and for Cook Nuclear Plant Unit 2 have shown that the impact of using V-5 fuel does not cause a significant change in physics parameters beyond the normal range of variation seen from cycle to cycle. Methods and core models used in the Cook Nuclear Plant Unit 2 transition are the standard approved ones.

Power distributions and peaking factors show slight changes as a result of the use of axial blankets and reduced length burnable absorbers, in addition to normal variations from different loading patterns. Technical Specifications changes are proposed for increased $F\Delta_H^N$ and $F_Q$ limits. The increased $F\Delta_H^N$ limit will allow a loading pattern design with lower leakage which in turn allows longer cycles. The increased $F_Q$ limit will provide greater flexibility with regard to the axial heterogeneous cores (blankets and short burnable absorbers). As is current practice, each reload core design will be evaluated to assure that design and safety limits are satisfied.

The staff has reviewed the nuclear design and concludes that it is acceptable because; (1) approved codes and methodologies have been used, (2) acceptable core parameters have been obtained, and (3) appropriate changes to the Technical Specifications have been proposed.

## 2.3 Thermal and Hydraulic Design

The calculational methods used for the thermal-hydraulic analysis of the ANF and VANTAGE 5 fuel used the approved Revised Thermal Design Procedure (RTDP) (Ref. 13). The ANF fuel analysis used the W-3 DNB correlation and the VANTAGE 5 fuel used the WRB-2 DNB correlation. The core was analyzed at the thermal-hydraulic conditions shown below:

| Thermal and Hydraulic Design Parameters | Bounding Parameters for Mixed Cores (Cycles 8 & 9) | Bounding Parameters for Homogeneous VANTAGE 5 Cores Cycles 10 & Beyond |
|---|---|---|
| Reactor Core Heat Output, MWt | 3588 | 3588 |
| Core Pressure, Nominal, psia | 2280 | 2130 |
| Nuclear Enthalpy Rise Hot Channel Factor, $F\Delta H_N$ | (ANF) 1.49<br>(V-5) 1.59 | 1.59 |
| Safety Analysis Limit DNBR | | |
| Typical Flow Channel | (ANF) 1.43<br>(V-5) 1.69 | 1.69 |
| Thimble Cold Wall Channel | (ANF) 1.40<br>(V-5) 1.61 | 1.61 |
| Vessel Average Temperature, °F | 576.0 | 581.3 |
| Vessel Minimum Measured Flow, GPM | 366,400 | 366,400 |

The design limit DNBRs are 1.23 and 1.22 for typical and thimble cells, respectively, for VANTAGE 5 fuel, and 1.39 and 1.36 for typical and thimble cells, respectively, for ANF fuel. Standard Thermal Design Procedure (STDP) was used where the RTDP methodology was not applicable. The parameters used in the STDP method were treated in a conservative way so as to give the lowest minimum DNBR. The differences between the design and safety analysis limit DNBR result in DNBR margin. A fraction of the margin was utilized to accommodate the transition core penalty. For VANTAGE 5 fuel, an additional margin of 1.3% for 20-inch grid spans and 0% for 10-inch grid spans was used to compensate for the rod bow penalty. There is no rod bow penalty required for the ANF fuel. The remaining margin is reserved for flexibility in the design.

The results of hydraulic tests performed on the ANF fuel assembly were compared to hydraulic test data for the VANTAGE 5 fuel assembly (Ref. 6). The data showed that the ANF and VANTAGE 5 fuel assemblies are hydraulically compatible.

The fuel temperatures used in the safety analysis for the V-5 fuel were calculated using the approved PAD code (Ref. 10). These fuel temperatures were used as initial conditions for the LOCA and non-LOCA transient.

The staff has reviewed the thermal-hydraulic design of the reference core and concludes that it is acceptable because; (1) approved codes and methodologies have been used, (2) rod bow penalty is offset by the DNBR margin available in the safety limit DNBR, and (3) all of the current thermal-hydraulic design criteria are satisfied.

## 2.4 Non-LOCA Accidents

### 2.4.1 Introduction

The analyses or evaluations of fifteen (15) non-LOCA accidents for the Cook Nuclear Plant Unit 2 licensing basis, as reported in the original FSAR (Ref. 14), were reviewed to address any impact resulting from the VANTAGE 5 fuel reload. The licensee has reviewed all the non-LOCA accidents to find the impact of using the V-5 fuel. Fourteen transients which were impacted by one or more of the V-5 design features or modified safety analysis assumptions as listed in Section 1 of this report were reanalyzed. Another accident, the Startup of an Inactive Loop, was not reanalyzed. The Technical Specification for Cook Nuclear Plant Unit 2 does not permit operation with less than four loops when operating in Modes 1 and 2. The fourteen accidents that were reanalyzed are:

(1) Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition
(2) Uncontrolled RCCA Bank Withdrawal at Power
(3) Rod Cluster Control Assembly (RCCA) Misalignment
(4) Rod Cluster Control Assembly (RCCA) Drop
(5) Uncontrolled Boron Dilution
(6) Loss of Forced Reactor Coolant Flow
(7) Loss of External Electric Load or Turbine Trip
(8) Loss of Normal Feedwater
(9) Excessive Heat Removal due to Feedwater System Malfunction
(10) Excessive Load Increase
(11) Loss of Offsite Power (LOOP) to the Station Auxiliaries
(12) Rupture of a Steamline (Steamline Break)
(13) Rupture of a Control Rod Drive Mechanism (CRDM) Housing (Rod Cluster Control Assembly Ejection)
(14) Major Rupture of Main Feedwater Pipe (Feedline Break)

### 2.4.2 Modified Safety Analysis Assumptions

Listed below are the analysis assumptions which represent a departure from that currently used for Cook Nuclear Plant Unit 2.

(1) Revised Maximum Moderator Density Coefficient
(2) Increased Design Enthalpy Rise Hot Channel Factors ($F\Delta_H^N$) and $F_Q$ for the Westinghouse VANTAGE 5 fuel
(3) Increased $F\Delta_H^N$ Part Power Multiplier on Westinghouse VANTAGE 5 fuel
(4) Decreased Shutdown Margin
(5) Revised Thermal Design Procedure (RTDP)
(6) Increased Core Power
(7) Reduced Temperature and Pressure (RTP) Operation
(8) 0 ppm boron concentration in the Boron Injection Tank (BIT)
(9) Constant Steam Generator Level Program
(10) System Performance Degradation

A brief description of each of these assumptions follows:

(1) Revised Maximum Moderator Density Coefficient:

The analyses considered an End-of-Cycle (EOC) most positive Moderator Density Coefficient (MDC) of 0.54 k/gm/cc. The Moderator Temperature Coefficient (MTC) varied as a function of vessel average temperature.

(2) Increased $F\Delta_H^N$ and $F_Q$ :

The design $F\Delta_H^N$ for the ANF and VANTAGE 5 fuel is 1.55 and 1.65, respectively. The non-LOCA calculations applicable for the VANTAGE 5 core assumed a full power $F\Delta_H^N$ of 1.65 which is a conservative safety analysis assumption.

The increase in the Technical Specification maximum LOCA $F_Q$ from 2.1 to 2.22 is conservatively bounded in the non-LOCA transients. A maximum $F_Q$ of 2.5 was assumed in the non-LOCA safety analyses.

(3) Increased $F\Delta_H^N$ Part Power Multipliers:

The $F\Delta_H^N$ part power multipliers are 0.2 for ANF fuel and 0.3 for VANTAGE 5 fuel. These values were considered in the generation of the core thermal limits for both fuel types. The changes in the core thermal safety limits result in a change to the Overtemperature and Overpower ΔT (OTΔT/OPΔT) reactor protection trip setpoints. Two sets of OTΔT/OPΔT setpoints were calculated. The first set of these setpoints was calculated based on ANF core thermal limits and is applicable for transition cycles. The second set of these setpoints was calculated based on VANTAGE 5 core thermal limits and is applicable for full VANTAGE 5 core (Cycles 10 and beyond). DNB analyses which are performed using LOFTRAN alone were analyzed twice, once for mixed core cycles and once for full VANTAGE 5 core. The remaining DNB analyses accounted for the variation in $F\Delta_H^N$ part power multipliers between a mixed core and a full VANTAGE 5 core.

(4) Decreased Shutdown Margin (SDM):

A change in the shutdown margin from 2.0% Δk/k (used for ANF fuel) to 1.3% Δk/k (standard values used for Westinghouse fuel) was used in the non-LOCA safety analyses.

(5) Revised Thermal Design Procedure (RTDP):

The calculational method utilized to meet the DNB design basis was the RTDP. Uncertainties in the plant operating parameters are statistically treated such that there is at least a 95 percent probability at a 95 percent confidence level that the minimum DNBR will be greater than the applicable limits. Since the parameter uncertainties are considered in determining the design DNBR value, the plant safety analyses were performed using nominal initial conditions without uncertainties. The ANF fuel analyses used the W-3 correlation, while the VANTAGE 5 fuel analyses use the WRB-2 correlation.

(6) Increased Core Thermal Power:

An increase in the nominal core thermal power from 3411 MWt to 3588 MWt was used in the non-LOCA safety analyses for the potential rerating of the Cook Nuclear Plant Unit 2. The non-LOCA safety analyses performed at 3588 MWt conservatively bounds the current nominal core thermal power level of 3411 MWt.

(7) Reduced Temperature and Pressure (RTP) Operation:

Reduced temperature and pressure operation for Cook Nuclear Plant Unit 2 was used in the non-LOCA safety analyses. The full power vessel average temperature range of 547°F to 581.3°F at either of two values of pressurizer pressure (2100 psia or 2250 psia) was used. However, because of the DNB constraints associated with the presence of ANF fuel during transition cycles (Cycles 8 and 9), limitations on pressure and temperature conditions were applied. These included a full power vessel average temperature range of 547°F to 576°F, and a pressurizer pressure of 2250 psia. Generating an acceptable nominal setpoint for the OTΔT reactor trip setpoint during transition cycles has resulted in this limitation. This limitation does not apply when a full VANTAGE 5 core is in place. The non-LOCA safety analyses provided support for a "full-window" of operation in the assumed range of RTDP operation when a full VANTAGE 5 core is in place at Cook Nuclear Plant Unit 2.

(8) BIT Boron Concentration:

A zero (0) ppm BIT boron concentration was assumed in the non-LOCA analyses to support BIT removal at Cook Nuclear Plant Unit 2. This is a conservative safety analysis assumption.

(9) Steam Generator Water Level Program:

A change in the steam generator water level program was used in the non-LOCA safety analyses. The existing steam generator water level program is a ramp function from 33% narrow range span (NRS) to 44% NRS from 0% power to 20% power and a constant level at 44% NRS between 20% power and 100% power. The steam generator water level program to be implemented at the beginning of Cycle 8 is a constant level at 44% NRS between 0% power and 100% power.

(10) System Performance Degradation:

The system performance degradation assumptions made for the non-LOCA safety analyses are as follows:

(a) A 10% average steam generator tube plugging level. This is a conservative safety analysis assumption for the non-LOCA analyses and bounds a 0% tube plugging level.

(b) An increase in the Main Steamline Isolation Valve (MSIV) closure time from 5 seconds to 8 seconds with a corresponding increase in total response times.

(c) A 10% Safety Injection Flow degradation.

(d) A 15 percent degradation in auxiliary feedwater pump capacity.

### 2.4.3 Conclusions for Non-LOCA Accidents

The staff has evaluated the licensee's evaluation and analysis of the non-LOCA events presented in Section 2.4.1, using the revised safety analysis assumptions associated with the VANTAGE 5 fuel presented in Section 2.4.2 and concludes that they are acceptable because (1) approved methodologies and computer codes have been used, and (2) all applicable safety criteria have been met.

## 2.5 Large and Small Break LOCA Analysis

### 2.5.1 Large Break LOCA

The large break analysis was performed with the December 1981 version of the Evaluation Model modified to incorporate the BASH (Ref. 15) computer code. A range of reactor operating temperatures was analyzed in order to justify plant operation at a reactor power level of 3588 MWt between 582.2°F to 615.2°F in the hot legs and 511.7°F to 547.6°F in the cold legs. In addition to the temperature range analyzed, initial RCS pressurizer pressure was also varied to justify plant operation between 2037 and 2313 psia. A full spectrum break analysis was done at the high pressure/high temperature RCS conditions (initial RCS pressurizer pressure, with uncertainty, of 2313 psia and initial hot leg temperature of 615.2°F) from which the limiting break size ($C_D$ = 0.6) was determined. The limiting break was then reanalyzed for low temperature and high RCS pressure. The limiting break was also reanalyzed for the high temperature and low initial RCS pressure of 2037 psia. The analysis also considered plant operation at reduced power level with the RHR cross tie valve closed.

The 17x17 ANF fuel assembly is nearly identical to the Westinghouse 17x17 OFA assembly in terms of hydraulic and geometric characteristics. Therefore, the analyses reported in Reference 6, which demonstrated that the 17x17 VANTAGE 5 fuel features result in a fuel assembly that is more limiting than a Westinghouse 17x17 OFA fuel assembly with respect to large break LOCA ECCS performance, remain valid as applied at Cook Nuclear Plant Unit 2. A large break LOCA transition core penalty was assigned to the transition from 17x17 ANF fuel to Westinghouse 17x17 VANTAGE 5 fuel assemblies. The various fuel assembly specific transition core analyses performed resulted in peak cladding temperature increases of up to 50°F for core axial elevations that bound the location of the PCT. Therefore, the maximum PCT penalty possible for VANTAGE 5 fuel residing in a transition core is 50°F. Once a full core of VANTAGE 5 fuel is achieved, the large break LOCA analysis will apply without the transition core penalty.

Of the seven cases reported, the largest peak clad temperature calculated for a large break was 2140°F, which is less than the acceptance criterion limit of 2200°F. The maximum local metal-water reaction was 6.80 percent, which is well below the embrittlement limit of 17 percent as required by 10 CFR 50.46. The total metal-water reaction was less than 0.3 percent for all breaks, corresponding to less than 0.3 percent hydrogen generation, as compared with the 1 percent criterion of 10 CFR 50.46. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. As a

result, the core temperature will continue to drop and the ability to remove decay heat generated in the fuel for an extended period of time will be provided. Therefore, the applicable acceptance criteria have been met.

2.5.2 Small Break LOCA

As stated in the VANTAGE 5 Westinghouse core reference report (Ref. 6), the mixed core hydraulic resistance mismatch is not a significant factor for small break LOCA analysis and therefore no transition core penalty was required. The small break analysis used the NOTRUMP (Ref. 16) computer code to calculate the transient depressurization of the RCS as well as describe the mass and enthalpy of the fluid flow through the break. Peak clad temperature calculations were performed using the LOCTA-IV (Ref. 17) code. The limiting small break from the LOCA analysis (as determined by the highest calculated peak clad temperature) for a range of break sizes and RCS pressures and temperatures at a core power of 3588 MWt was found to be a 4-inch diameter cold leg break initiated at reduced RCS pressurizer pressure (2100 psia) and high temperature (core Tavg = 581.3°F) conditions. The peak clad temperature attained during the transient was 1357°F. The maximum local metal-water reaction was 0.15 percent and the total metal-water reaction was less than 0.3 percent for all cases analyzed. There was no rod burst. All other acceptance criteria were met.

Additional calculations were made to support closure of the high head safety injection cross tie valves. Since the amount of pumped injection flow would be reduced with the high head cross tie valves closed, it was necessary to lower core power to 3413 MWt in order to maintain the peak clad temperature within the 10 CFR 50.46 limit. It was concluded that, with the high head cross tie valves closed and with reduced reactor power, the limiting break would be shifted from the 4-inch diameter cold leg break to the 3-inch diameter break size. The small break LOCA analysis for this condition resulted in a peak clad temperature of 2124°F. The maximum local metal-water reaction was 8.65 percent and the total metal-water reaction was less than 0.3 percent for all cases analyzed. There was no rod burst. All other acceptance criteria were met.

2.5.3 Steamline Break Mass and Energy Releases

2.5.3.1 Steamline Break Mass and Energy Release Inside Containment

WCAP-11902, Supplement 1, Section S-3.3.4.1, documents the steamline break mass and energy release inside of containment for Units 1 and 2. However, in the transition to VANTAGE 5 fuel, the analysis does not consider the 10% safety injection flow degradation or the increase in RCCA insertion time to dashpot increase to 2.7 seconds from 2.4 seconds. An evaluation performed by the licensee concludes that these differences would have an insignificant effect on the calculated mass and energy releases. Therefore, we find the analysis supports the transition to VANTAGE 5 fuel.

2.5.3.2 Steamline Break and Energy Release Outside Containment

The mass and energy releases were evaluated for VANTAGE 5 fuel in References 19 and 20. It was found that the energy releases documented in Reference 19 are bounding for Unit 2 provided that the following conditions are satisfied: (1) Maximum allowable NSSS power of 3425 MWt, (2) EOC life most positive MDC not

more positive than 0.43 ΔK/gm/cc, (3) minimum shutdown margin of 1.6% ΔK/K, (4) maximum allowable steamline isolation valve closure time no greater than 5.0 seconds, and (5) the compensated nominal setpoint for low steamline pressure no less than 520 psig. This setpoint corresponds to the analysis setpoint of 379 psig. These conditions are consistent with the Technical Specifications proposed for Cycle 8 and are acceptable.

2.5.4 Containment Response

2.5.4.1 Short Term Containment Analysis

The short-term containment analysis documented in Section 3.4.1 of WCAP-11902 (Reference 19) applies to both Units 1 and 2. We have reviewed the input parameters for the WCAP-11902 analysis and the operating parameters for Cycle 8 of D. C. Cook, Unit 2 to confirm the applicability of the previous analysis. Based on the confirmation, we conclude that the NRC Safety Evaluation Report (SER) on the short term containment analysis documented in Section 2.6 of Reference 22 for D. C. Cook Unit 1 also applies to Unit 2 Cycle 8 operation.

2.5.4.2 Long Term LOCA Containment Analysis

The long-term LOCA containment analysis for Units 1 and 2 was documented in WCAP-11908 (Reference 23). Since the temperatures of the reactor coolant system (RCS) used in WCAP-11908, with the exception of the temperature at the vessel inlet, were not explicitly stated, it was not clear that the analysis described in WCAP-11908 applied to Unit 2, Cycle 8 operation.

By telecopy of July 31, 1990, the licensee clarified the following points:

(1) the operating temperature of the reactor coolant system will be higher in Cycle 8 than in Cycle 7;

(2) the RCS temperature of the reactor coolant system will be higher in Cycle 8 than in Cycle 7;

(3) Tavg for the WCAP-11908 analysis, Cycle 8 operation, and Cycle 7 operations, are 581.3, 578.7, and 576.3 degrees F, respectively.

Based on this clarification, we find that the calculated containment responses in WCAP-11908 bound the response from both Cycle 8 and Cycle 7 operations at D. C. Cook Unit 2. Furthermore, we find the safety evaluation documented in Reference 24 to be applicable for Unit 2, Cycle 8 operation.

2.5.4.3 Containment Analysis for Steam Line Break

Steamline breaks inside containment were analyzed in Reference 19. We have reviewed the assumptions, methodology, and input parameters used for the analysis. The containment responses from a spectrum of breaks associated with different break sizes, break types, power levels, protection system designs, and single failures were analyzed. The LOFTRAN compute code was used to calculate the mass and energy releases. The blowdown was assumed to be dry saturated steam. Using this assumption, the break flow could not become superheated following a steam generator dryout. The licensee indicated that the same

assumption of dry saturated steam was used as the design basis in the FSAR. The concern of superheated steam releases for other ice-condenser containments did not apply to D. C. Cook, Units 1 and 2 because of the containment spray in the low compartment. The spray would eliminate the temperature effects resulting from the superheated steam releases. Therefore, we find the assumption of dry saturated steam for D. C. Cook steam line isolation was assumed to be 11 seconds, which allowed 8 seconds for valve closure and 3 seconds for electric delays and signal processing. The results of this analysis were bounded by the current FSAR results. Therefore, we conclude that the analyses for steamline break inside containment as documented in WCAP-11902 are acceptable.

## 2.6 Conditions on Use of VANTAGE 5 Fuel

During the staff review of the VANTAGE 5 fuel designs in WCAP-10444-P-A, the staff identified several conditions to be resolved for use of VANTAGE 5 fuel. Our review of the conditions listed in the safety evaluation of WCAP-10444-P-A follows.

### 2.6.1 Statistical Convolution Method

In our SER on WCAP-10444, we stated that the statistical method should not be used in VANTAGE 5 for evaluating the fuel rod shoulder gap. The licensee stated that worst case fabrication tolerances and fuel rod and assembly growth were used to determine the initial fuel rod to nozzle growth gaps. We consider this acceptable.

### 2.6.2 Seismic and LOCA Loads

In our SER on WCAP-10444, we stated that, for each plant application, it must be demonstrated that the fuel assembly will maintain its coolable geometry under combined seismic and LOCA loads. The licensee stated that an evaluation has been performed and that the grid load results are well below the grid strengths. This is also true for transition cores with ANF, LOPAR and V-5 fuel. The stresses in the fuel assembly components resulting from seismic and LOCA induced deflections are within acceptable limits and all criteria are met. Thus, we consider this acceptable.

### 2.6.3 Irradiation Demonstration Program

In our SER on WCAP-10444 we required that an irradiation program be performed to confirm the V-5 fuel performance. The demonstration of 17x17 V-5 fuel rod performance has been successfully illustrated by irradiation at the V. C. Summer plant Unit 2. Also V-5 has been irradiated in Turkey Point 3 and 4 for two reactor cycles and the IFM grid feature has been successfully demonstrated at McGuire Unit 1. We consider this acceptable.

### 2.6.4 Improved Thermal Design Procedure (ITDP)

In our SER on WCAP-10444, we stated that those restrictions in approving the use of Westinghouse improved thermal design procedure (ITDP) should be applied to the V-5 fuel design. The licensee stated that revised thermal design procedure (RTDP) (WCAP-1197-P-A), an extension of ITDP, was used for D. C. Cook Unit 2. The licensee also submitted a report, WCAP-12576, "Westinghouse Revised Thermal

Design Procedure Instrument Uncertainty Methodology for American Electric Power D. C. Cook Unit 2" from which the flow measurement uncertainty was 1.9%, not including a 1% fouling penality. However, the licensee will continue to use the existing more conservative 3.5% flow measurement uncertainty. We thus consider the use of the ITDP acceptable as the restrictions were followed.

2.6.5 DNBR Limit

In our SER on WCAP-10444, we stated that the WRB-2 correlation with a DNBR limit of 1.17 is acceptable for application to 17x17 VANTAGE 5 fuel when used in the proper applicability range. The licensee has verified that D. C. Cook Unit 2 application, using the WRB-2 correlation with a DNBR limit of 1.17, is within the approved range of applicability for use with ITDP methodology. Therefore, we conclude that this is acceptable.

2.6.6 Transition Core Penalty

In our SER on WCAP-10444, we stated that a separate analysis will be required to determine a transitional mixed core penalty. The licensee has addressed the issue of transitional core penalty due to potential flow mismatch. The transitional core DNBR penalty is a function of the number of VANTAGE 5 fuel assemblies in the core and is addressed in References 19 and 20. We find this acceptable.

2.6.7 Higher Peaking Factor

In our SER on WCAP-10444, we stated that a plant-specific analysis should be performed to demonstrate that the DNBR limit will not be violated with higher $F\Delta_H$ and $F_Q$. The licensee's analysis shows that the DNBR limits are met with the higher peaking factors. This is acceptable.

2.6.8 Steam System Pipe Failure with Loss of Offsite Power

In our SER on WCAP-10444, we stated that a specific-plant analysis should be performed to demonstrate that the steam system piping failure event can be satisfied with the assumption of loss of offsite power if that is the most conservative case. The D. C. Cook Unit 2 plant-specific analysis for main steam piping rupture was performed for the VANTAGE 5 fuel transition and included scenarios both with and without loss of offsite power. Based on meeting the acceptable DNB design basis criterion, we find the results acceptable.

2.6.9 Reactor Coolant Pump Shaft Seizure

In our SER on WCAP-10444, we stated that the mechanistic approach in determining the fraction of fuel failures during the reactor coolant pump shaft seizure accident was unacceptable. Consistent with the original design basis, the licensee used criteria which require a maximum clad temperature limit of 2700°F and total metal-water reaction at the hot spot less than 16% for the RCS pump shaft seizure (locked rotor) for the VANTAGE 5 transition. This limit is to ensure that a coolable geometry is maintained. As indicated above, we disapprove of the mechanistic approach based on 2700°F peak clad temperature in determining fuel failure. We require that the fuel failure criterion be based on the

approved 95/95 DNBR limit and that the DNBR analysis is based on this criterion. Westinghouse performed a separate analysis using the LOFTRAN, FACTRAN and THINC computer codes to determine the number of rods that experience DNB during the locked rotor accident. Any rods which violated the 95/95 DNBR limit were assumed to fail in this analysis. The results of this analysis showed that 11% of the rods were predicted to be below the 95/95 DNBR limit. The amount of fuel failures will be used in the dose calculation for radiological impact assessment.

We have reviewed the licensee's radiological assumptions and parameters used in their calculations and we have determined that they are in accordance with Standard Review Plan (SRP) Sections 15.3.3 and 15.6.3. The licensee's calculated offsite doses are well within the acceptance criteria specified in SRP Section 15.3.3. Therefore, we conclude that the licensee's reactor coolant pump locked rotor offsite radiological dose analysis is acceptable.

### 2.6.10 Positive Moderator Temperature Coefficient (MTC)

In our SER on WCAP-10444, we stated that, if a positive MTC is intended for VANTAGE 5, the same positive MTC consistent with the plant Technical Specification should be used in the plant-specific safety analysis. The licensee affirmed that the above condition was used in the safety analysis for VANTAGE 5 fuel. Therefore, we find the positive MTC condition satisfactorily addressed.

## 2.7 Technical Specification Changes

The proposed Technical Specification (TS) changes for Unit 2 Cycle 8 allow for a transition to Westinghouse 17x17 VANTAGE 5 fuel as a replacement for 17x17 ANF fuel. The majority of the proposed changes are related to the transformation to Westinghouse fuel and reload analysis methodology. The TS changes are also based on modified safety assumptions as discussed in Section 2.4.2 of this SER. The majority (approximately 100) of the proposed TS changes are for Unit 2. Certain Unit 1 TS changes were proposed for essentially identical changes to the Unit 2 TS so that both units can be maintained more nearly alike.

These changes include the proposal to remove special shutdown margin requirements when operating on residual heat removal, the modification of the axial flux difference surveillance, and the new acceptance criteria for safety injection pumps and residual heat removal pumps (a full description of these changes is contained in Reference 1, Attachment 3 and Reference 4). These changes have an identical or essentially identical justification for both units and the Unit 2 justification contained in Reference 1 was evaluated in Section 2 of this SER. Therefore, the staff finds these changes acceptable for Unit 1 as well.

The licensee also proposed numerous administrative or editorial changes. These include rotating print 90 degrees, replacing mathematical symbols with written words, and moving sections of text from one location to another. A full description of these changes contained in Attachment 3 of Reference 1. We have reviewed each of these changes and find them to be purely administrative or editorial and, therefore, acceptable.

As stated previously, the May 29, 1990 submittal also included a change to proposed page 2-2 to correct a typographical error from a previous submittal. This change is acceptable. In addition, the staff modified several Technical Specification pages, included in the February 6, 1990 submittal to reflect other amendments issued during the pendency of the request and determined that changes proposed to TS pages B 3/4 2-2 and B 3/4 2-4b for Unit 2 (also in the February 6, 1990 submittal) need not be issued because of recent license amendments which incorporated those changes. The licensee has agreed to this updating of its submittal.

We have reviewed each of the changes to the TSs as proposed by the licensee, and have determined that they are acceptable because they reflect the design changes made to the plant and the assumptions used in the safety analysis as discussed in Section 2, or are administrative or editorial in nature.

We have reviewed the request by Indiana Michigan Power Company to operate the Donald C. Cook Nuclear Power Plant Unit 2 with VANTAGE 5 fuel and to change some Unit 1 and Unit 2 Technical Specifications to maintain both units more nearly alike. Based on this review, we conclude that appropriate material was submitted and that normal and transient operation and accidents were properly reevaluated/ reanalyzed. The Technical Specifications submitted reflect the necessary modifications for operation following future reloads. The licensee's request is, therefore, acceptable.

## 3.0 ENVIRONMENTAL CONSIDERATION

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

## 4.0 CONCLUSION

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

## 5.0 REFERENCES

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