April 19, 1989

Docket No. 50-483

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Mr. Donald F. Schnell
Senior Vice President - Nuclear
Union Electric Company
Post Office Box 149
St. Louis, Missouri 63166

Dear Mr. Schnell:

SUBJECT: AMENDMENT NO. 44 TO FACILITY OPERATING LICENSE NO. NPF-30 (TAC NO. 69875)

The Commission has issued the enclosed Amendment No. 44 to Facility Operating License No. NPF-30 for the Callaway Plant, Unit 1. This amendment revises the Technical Specifications (TS) in response to your application dated October 25, 1988.

The amendment revises the TS to support a core reload for Cycle 4. The revised TS include increased peaking factors, a positive moderator temperature coefficient, increased refueling water storage tank and accumulator boron concentrations, and increased spray additive tank sodium hydroxide concentration.

A copy of the Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

Thomas W. Alexion, Project Manager
Project Directorate III-3
Division of Reactor Projects - III, IV, V and Special Projects

Enclosures:
1. Amendment No. 44 to License No. NPF-30
2. Safety Evaluation

cc w/enclosures:
See next page

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Callaway Plant
Unit No. 1

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

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT 1

DOCKET NO. STN 50-483

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 44
License No. NPF-30

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

   A. The application for amendment filed by Union Electric Company (UE, the licensee) dated October 25, 1988 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. NPF-30 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 44 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into the license. UE shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

John N. Hannon

John N. Hannon, Director
Project Directorate III-3
Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: April 19, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 44

OPERATING LICENSE NO. NPF-30

DOCKET NO. 50-483

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change. Corresponding overleaf pages are provided to maintain document completeness.

REMOVE	INSERT
B 2-1	B 2-1
B 2-2	B 2-2
3/4 1-4	3/4 1-4
3/4 1-11	3/4 1-11
3/4 1-12	3/4 1-12
3/4 2-1	3/4 2-1
3/4 2-2(a)	3/4 2-2(a)
3/4 2-4	3/4 2-4
3/4 2-5	3/4 2-5
3/4 2-6	3/4 2-6
3/4 2-7	3/4 2-7
3/4 2-7(a)	3/4 2-7(a)
3/4 2-7(b)	3/4 2-7(b)
3/4 2-8	3/4 2-8
3/4 5-1	3/4 5-1
3/4 5-10	3/4 5-10
3/4 6-14	3/4 6-14
B 3/4 1-2	B 3/4 1-2
B 3/4 1-3	B 3/4 1-3
B 3/4 2-1	B 3/4 2-1
B 3/4 2-4	B 3/4 2-4
B 3/4 2-5	B 3/4 2-5
B 3/4 2-6	B 3/4 2-6
B 3/4 5-2	B 3/4 5-2

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. This relation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio (DNBR) defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, 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-1 correlation for Optimized fuel (OFA) and the WRB-2 correlation for VANTAGE 5 fuel in this application). The correlation DNBR limit is 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 both the WRB-1 and WRB-2 correlations).

In meeting this design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95% probability with 95% confidence level that the minimum DNBR for the limiting rod is greater than or equal to the DNBR limit. The uncertainties in the above plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit, establishes a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties. For Callaway, the design DNBR values are 1.33 and 1.35 for thimble and typical cells, respectively, for OFA, and 1.33 and 1.34 for thimble and typical cells, respectively, for VANTAGE 5 fuel. In addition, margin has been maintained in both fuel designs by meeting safety analysis DNBR limits of 1.42 and 1.45 for thimble and typical cells, respectively, for OFA, and 1.61 and 1.69 for thimble and typical cells, respectively, for VANTAGE 5 fuel.

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 value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

2.1.1 REACTOR CORE (Continued)

The curves are based on a measured nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$, of 1.49 for OFA and 1.59 for VANTAGE 5 fuel, and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expressions:

$$F^N_{\Delta H} = 1.49\ [1 + 0.3\ (1-P)] \text{ for OFA, and}$$
$$F^N_{\Delta H} = 1.59\ [1 + 0.3\ (1-P)] \text{ for VANTAGE 5 fuel}$$

where P is the fraction of RATED THERMAL POWER.

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

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this safety limit protects the integrity of the Reactor Coolant System (RCS) from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor vessel, pressurizer, and the RCS piping and valves are designed to Section III of the ASME Code for Nuclear Power Plants which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated Code requirements.

The entire RCS is hydrotested at greater than or equal to 125% (3110 psig) of design pressure to demonstrate integrity prior to initial operation.

SHUTDOWN MARGIN - $T_{avg} \leq 200°F$

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 1% Δk/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 1% Δk/k, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MAR is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1% Δk/k:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inopera 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); and

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

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.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than +5 pcm/°F for power levels up to 70% RATED THERMAL POWER and a linear ramp from that point to 0 pcm/°F at 100% RATED THERMAL POWER for the all rods withdrawn, beginning of cycle life (BOL) condition; and

b. Less negative than -4.1×10^{-4} $\Delta k/k/°F$ for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.

APPLICABILITY: Specification 3.1.1.3a. - MODES 1 and 2*#.
Specification 3.1.1.3b. - MODES 1, 2 and 3#.

ACTION:

a. With the MTC more positive than the limit of Specification 3.1.1.3a. above, operation in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to within the above limits within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6;

2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition; and

3. A Special Report is prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the limit of Specification 3.1.1.3b. above, be in HOT SHUTDOWN within 12 hours.

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A Boric Acid Storage System with:

1) A minimum contained borated water volume of 2968 gallons,

2) Between 7000 and 7700 ppm of boron, and

3) A minimum solution temperature of 65°F.

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water volume of 55,416 gallons,

2) A minimum boron concentration of 2350 ppm, and

3) A minimum solution temperature of 37°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration of the water,

2) Verifying the contained borated water volume, and

3) Verifying the Boric Acid Storage System solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water and the outside air temperature is less than 37°F.

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 As a minimum, the following borated water sources shall be OPERABLE as required by Specification 3.1.2.2 for MODES 1, 2 and 3 and one of the following borated water sources shall be OPERABLE as required by Specification 3.1.2.1 for MODE 4:

a. A Boric Acid Storage System with:

   1) A minimum contained borated water volume of 17,658 gallons,

   2) Between 7000 and 7700 ppm of boron, and

   3) A minimum solution temperature of 65°F.

b. The refueling water storage tank (RWST) with:

   1) A minimum contained borated water volume of 394,000 gallons,

   2) Between 2350 and 2500 ppm of boron,

   3) A minimum solution temperature of 37°F, and

   4) A maximum solution temperature of 100°F.

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

ACTION:

a. With the Boric Acid Storage System inoperable and being used as one of the above required borated water sources in MODE 1, 2, or 3, 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 RWST inoperable in MODE 1, 2, or 3, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With no borated water source OPERABLE in MODE 4, restore one borated water source to OPERABLE status within 6 hours or be in COLD SHUTDOWN within the following 30 hours.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the following target band (flux difference units) about the target flux difference:

a. +3%, -12% for Normal Operation

b. $\pm$3% for RESTRICTED AFD OPERATION

The indicated AFD may deviate outside the applicable required target band at greater than or equal to 50% but less than 0.9 APL^{ND}** or 90% of RATED THERMAL POWER, whichever is less, provided the indicated AFD is within the Acceptable Operation Limits of Figure 3.2-1 and the cumulative penalty deviation times does not exceed 1 hour during the previous 24 hours.

The indicated AFD may deviate outside the applicable required target band at greater than 15% but less than 50% of RATED THERMAL POWER provided the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

APPLICABILITY: MODE 1, above 15% of RATED THERMAL POWER*,#

ACTION:

a. With the indicated AFD outside of the applicable required target band and with THERMAL POWER greater than or equal to 0.9 APL^{ND}** or 90% of RATED THERMAL POWER, whichever is less, within 15 minutes, either:

   1. Restore the indicated AFD to within the applicable required target band limits, or

---

* See Special Test Exception Specification 3.10.2.

\# Surveillance testing of the Power Range Neutron Flux channel may be performed pursuant to Specification 4.3.1.1 provided the indicated AFD is maintained within the Acceptable Operation Limits of Figure 3.2-1 and THERMAL POWER $\leq APL^{NO}$***. A total of 16 hours operation may be accumulated with the AFD outside of the applicable required target band during testing without penalty deviation.

** APL^{ND} is the minimum allowable power level for RESTRICTED AFD OPERATION and will be provided in the Peaking Factor Limit Report per Specification 6.9.1.9.

*** APL^{NO} is equal to the

$$\underset{\text{over } Z}{\text{minimum}} \left[\frac{2.50 \, * \, K(Z)}{F_Q^M(Z) \, * \, W(Z)_{NO}} * 100 \right]$$

and $F_Q^M(Z)$ and $W(Z)_{NO}$ are defined in 4.2.2.2.c.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

2. Reduce THERMAL POWER to less than 0.9 APL^{ND}** or 90% of RATED THERMAL POWER, whichever is less, and discontinue RESTRICTED AFD OPERATION (if applicable).

b. With the indicated AFD outside of the applicable required target band for more than 1 hour of cumulative penalty deviation time during the previous 24 hours or outside the Acceptable Operation Limits of Figure 3.2-1 and with THERMAL POWER less than 0.9 APL^{ND}** or 90%, whichever is less, but equal to or greater than 50% of RATED THERMAL POWER, reduce:

1. THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes, and

2. The Power Range Neutron Flux-High Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

c. With the indicated AFD outside of the applicable required target band for more than 1 hour of cumulative penalty deviation time during the previous 24 hours and with THERMAL POWER less than 50% but greater than 15% of RATED THERMAL POWER, the THERMAL POWER shall not be increased equal to or greater than 50% of RATED THERMAL POWER until the indicated AFD is within the applicable required target band.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AFD 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.

b. Monitoring and logging the indicated AFD 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 AFD Monitor Alarm is inoperable. The logged values of the indicated AFD shall be assumed to exist during the interval preceding each logging.

SURVEILLANCE REQUIREMENTS

4.2.1.2 The indicated AFD shall be considered outside of its target band when two or more OPERABLE excore channels are indicating the AFD to be outside the target band. Penalty deviation outside of the above required target band shall be accumulated on a time basis of:

a. One minute penalty deviation for each 1 minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

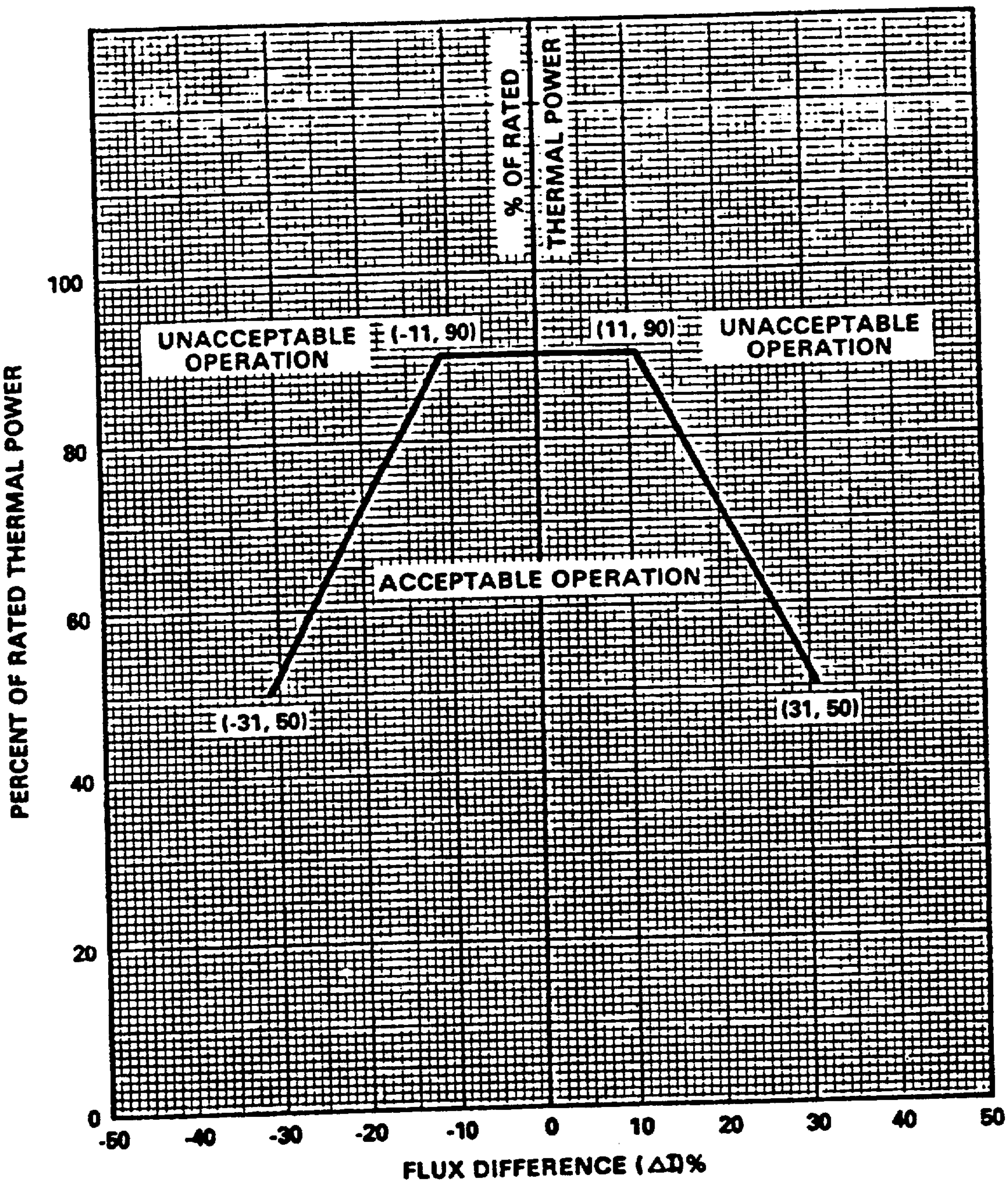
b. One-half minute penalty deviation for each 1 minute of POWER OPERATION outside of the target band at THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER.

4.2.1.3 The target flux difference of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full Power Days. The provisions of Specification 4.0.4 are not applicable.

4.2.1.4 The target flux difference shall be updated at least once per 31 Effective Full Power Days by either determining the target flux difference pursuant to Specification 4.2.1.3 above or by linear interpolation between the most recently measured value and the calculated value at the end of the cycle life. The provisions of Specification 4.0.4 are not applicable.

FIGURE 3.2-1

AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER



POWER DISTRIBUTION LIMITS

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

LIMITING CONDITION FOR OPERATION

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

$F_Q(Z) \leq \frac{[2.50]}{P} [K(Z)]$ for P > 0.5, and

$F_Q(Z) \leq [5.00] [K(Z)]$ for $p \leq 0.5$.

Where:

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

K(Z) = the function obtained from Figure 3.2-2 for a given core height location.

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(Z)$ exceeding its limit:

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

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

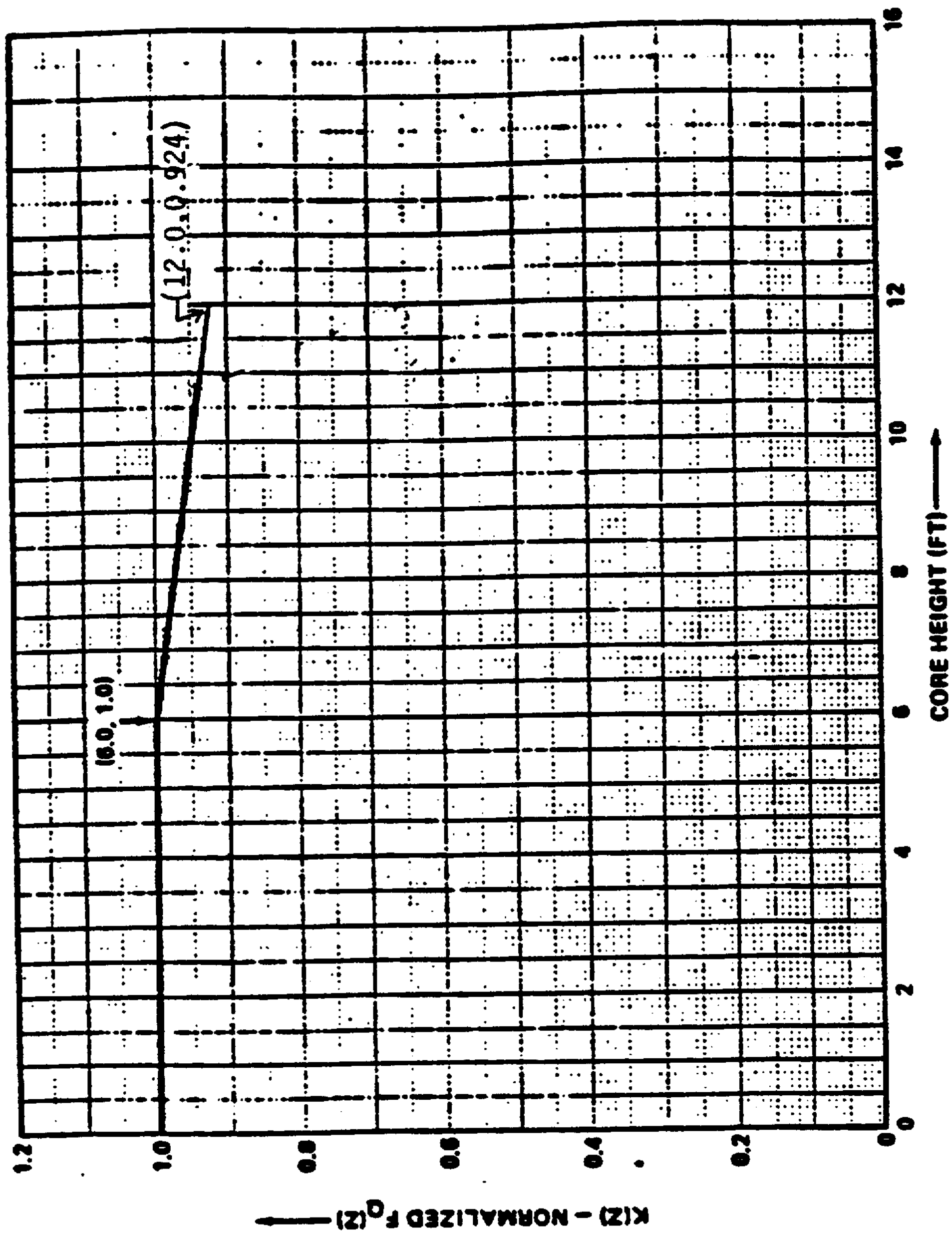


FIGURE 3.2-2.
K(Z) – NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 For Normal Operation, $F_Q(z)$ shall be evaluated to determine if $F_Q(z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured $F_Q(z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties.

c. Satisfying the following relationship:

$$F_Q^M(z) \leq \frac{2.50 \times K(z)}{P \times W(z)_{NO}} \text{ for } P > 0.5$$

$$F_Q^M(z) \leq \frac{2.50 \times K(z)}{W(z)_{NO} \times 0.5} \text{ for } P \leq 0.5$$

where $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, 2.50 is the F_Q limit, K(z) is given in Figure 3.2-2, P is the relative THERMAL POWER, and $W(z)_{NO}$ is the cycle dependent, Normal Operation function that accounts for power distribution transients encountered during Normal Operation. This function is given in the Peaking Factor Limit Report as per Specification 6.9.1.9.

d. Measuring $F_Q^M(z)$ according to the following schedule:

1. Upon achieving equilibrium conditions after exceeding, by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(z)$ was last determined,* or

2. At least once per 31 Effective Full Power Days (EFPD), whichever occurs first.

*During power escalation at the beginning of each cycle, power level may be increased until a power level for extended operation has been achieved and a power distribution map obtained.

4.2.2.2 (Continued)

e. With measurements indicating

$$\underset{\text{over } z}{\text{maximum}} \left(\frac{F_Q^M(z)}{K(z)} \right)$$

has increased since the previous determination of $F_Q^M(z)$, either of the following actions shall be taken:

1. $F_Q^M(z)$ shall be increased by 2% over that specified in Specification 4.2.2.2c., or

2. $F_Q^M(z)$ shall be measured at least once per 7 Effective Full Power Days until two successive maps indicate that

$$\underset{\text{over } z}{\text{maximum}} \left(\frac{F_Q^M(z)}{K(z)} \right) \text{ is not increasing.}$$

f. With the relationships specified in Specification 4.2.2.2c. above not being satisfied:

1. Calculate the percent $F_Q(z)$ exceeds its limit by the following expression:

$$\left[\left(\text{max. over } z \text{ of} \left(\frac{F_Q^M(z) \times W(z)\ NO}{\frac{2.50}{P} \times K(z)} \right) \right) - 1 \right] \times 100 \quad \text{for} \quad P \geq 0.5$$

$$\left[\left(\text{max. over } z \text{ of} \left(\frac{F_Q^M(z) \times W(z)NO}{\frac{2.50}{0.5} \times K(z)} \right) \right) - 1 \right] \times 100 \quad \text{for} \quad P < 0.5$$

2. Either one of the following actions shall be taken:

(a) Comply with the requirements of Specification 3.2.2 for $F_Q(z)$ exceeding its limit by the percent calculated above, or

(b) Verify that the requirements of Specification 4.2.2.3 for RESTRICTED AFD OPERATION are satisfied and enter RESTRICTED AFD OPERATION.

g. The limits specified in Specifications 4.2.2.2.c., 4.2.2.2.e., and 4.2.2.2.f. above are not applicable in the following core plane regions:

1. Lower core region from 0 to 15%, inclusive.
2. Upper core region from 85 to 100%, inclusive.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.2.3 RESTRICTED AFD OPERATION (RAFDO) is permitted at powers above APL^{ND} if the following conditions are satisfied:

a. Prior to entering RAFDO, maintain THERMAL POWER above APL^{ND} and less than or equal to that allowed by Specification 4.2.2.2 for at least the previous 24 hours. Maintain RAFDO surveillance (AFD within $\pm 3\%$ of target flux difference) during this time period. RAFDO is then permitted providing THERMAL POWER is maintained between APL^{ND} and APL^{RAFDO} or between APL^{ND} and 100% (whichever is more limiting) and F_Q surveillance is maintained pursuant to Specification 4.2.2.4. APL^{RAFDO} is defined as:

$$APL^{RAFDO} = \underset{\text{over } z}{\text{minimum}} \left[\frac{2.50 \times K(z)}{F_Q^M(z) \times W(z)_{RAFDO}} \right] \times 100\%$$

where: $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty. The F_Q limit is 2.50. K(z) is given in Figure 3.2-2. $W(z)_{RAFDO}$ is the cycle dependent function that accounts for limited power distribution transients encountered during RAFDO. This function is given in the Peaking Factor Limit Report as per Specification 6.9.1.9.

b. During RAFDO, if the THERMAL POWER is decreased below APL^{ND} then the conditions of 4.2.2.3.a shall be satisfied before re-entering RAFDO.

4.2.2.4 During RAFDO, $F_Q(z)$ shall be evaluated to determine if $F_Q(z)$ is within its limits by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER above APL^{ND}.

b. Increasing the measured $F_Q(z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties.

c. Satisfying the following relationship:

$$F_Q^M(z) \leq \frac{2.50 \times K(z)}{P \times W(z)_{RAFDO}} \quad \text{for } P > APL^{ND}$$

where: $F_Q^M(z)$ is the measured $F_Q(z)$. The F_Q limit is 2.50. K(z) is given in Figure 3.2-2. P is the relative THERMAL POWER. $W(z)_{RAFDO}$ is the cycle dependent function that accounts for limited power distribution transients encountered during RAFDO. This function is given in the Peaking Factor Limit Report as per Specification 6.9.1.9.

4.2.2.4 (Continued)

d. Measuring $F_Q^M(z)$ in conjunction with target flux difference determination according to the following schedule:

   1. Prior to entering RAFDO after satisfying Section 4.2.2.3 unless a full core flux map has been taken in the previous 31 EFPD with the relative thermal power having been maintained above APL^{ND} for the 24 hours prior to mapping, and

   2. At least once per 31 Effective Full Power Days.

e. With measurements indicating

$$\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M(z)}{K(z)} \right]$$

has increased since the previous determination of $F_Q^M(z)$ either of the following actions shall be taken:

   1. $F_Q^M(z)$ shall be increased by 2 percent over that specified in 4.2.2.4.c, or

   2. $F_Q^M(z)$ shall be measured at least once per 7 EFPD until two successive maps indicate that

$$\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M(z)}{K(z)} \right] \text{ is not increasing.}$$

f. With the relationship specified in 4.2.2.4.c above not being satisfied, comply with the requirements of Specification 3.2.2 for $F_Q(z)$ exceeding its limit by the percent calculated with the following expression:

$$\left[\left(\text{max. over z of} \left(\frac{F_Q^M(z) \times W(z)_{RAFDO}}{\frac{2.50}{P} \times K(z)} \right) \right) -1 \right] \times 100 \text{ for } P \geq APL^{ND}$$

g. The limits specified in 4.2.2.4.c, 4.2.2.4.e, and 4.2.2.4.f above are not applicable in the following core plane regions:

   1. Lower core region from 0 to 15 percent, inclusive.

   2. Upper core region from 85 to 100 percent, inclusive.

4.2.2.5 When $F_Q(z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2.2 or 4.2.2.4, an overall measured $F_Q(z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

POWER DISTRIBUTION LIMITS

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F^N_{\Delta H}$

LIMITING CONDITION FOR OPERATION

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

$$F^N_{\Delta H} \leq 1.59\ [1 + 0.3\ (1-P)]$$

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

$F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F^N_{\Delta H}$ shall be used since an uncertainty of 4% for incore measurement of $F^N_{\Delta H}$ has been included in the above limit.

APPLICABILITY: MODE 1

ACTION:

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

a. Within 2 hours either:

1. Restore the $F^N_{\Delta H}$ to within the above limits, or
2. Reduce THERMAL POWER TO LESS THAN 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux-High Trip Setpoint to ≤ 55% of RATED THERMAL POWER within the next 4 hours.

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

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

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System accumulator shall be OPERABLE with:

a. The isolation valve open and power removed,

b. A contained borated water volume of between 6061 and 6655 gallons,

c. A boron concentration of between 2300 and 2500 ppm, and

d. A nitrogen cover-pressure of between 602 and 648 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 1 hour or be in at least HOT STANDBY within the next 6 hours and reduce RCS pressure to less than 1000 psig within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed#, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and reduce RCS pressure to less than 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying, by the absence of alarms, the contained borated water volume and nitrogen cover-pressure in the tanks, and

2) Verifying that each accumulator isolation valve is open.

* RCS pressure above 1000 psig.

One accumulator isolation valve may be closed for up to 2 hours in mode 3* for surveillance testing per 4.0.5 or 4.4.6.2.2.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 70 gallons by verifying the boron concentration of the accumulator solution; and

c. At least once per 31 days when the RCS pressure is above 1000 psig by verifying that the circuit breaker supplying power to the isolation valve operator is open.

4.5.1.2 Each accumulator water level and pressure channel shall be demonstrated OPERABLE at least once per 18 months by the performance of a CHANNEL CALIBRATION.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.4 ECCS SUBSYSTEMS - $T_{avg} \leq 200°F$

LIMITING CONDITION FOR OPERATION

3.5.4 All Safety Injection pumps shall be inoperable.

APPLICABILITY: MODE 5 with the water level above the top of the reactor vessel flange, and MODE 6 with the reactor vessel head on and with the water level above the top of the reactor vessel flange.

ACTION:

With a Safety Injection pump OPERABLE, restore all Safety Injection pumps to an inoperable status within 4 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 All Safety Injection pumps shall be demonstrated inoperable* by verifying that the motor circuit breakers are secured in the open position at least once per 31 days.

*An inoperable pump may be energized for testing or for filling accumulators provided the discharge at the pump has been isolated from the RCS by a closed isolation valve with power removed from the valve operator, or by a manual isolation valve secured in the closed position.

3/4.5.5 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A minimum contained borated water volume of 394,000 gallons,

b. A boron concentration of between 2350 and 2500 ppm of boron,

c. A minimum solution temperature of 37°F, and

d. A maximum solution temperature of 100°F.

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

ACTION:

With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the contained borated water volume in the tank, and

2) Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is either less than 37°F or greater than 100°F.

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent Containment Spray Systems shall be OPERABLE with each Containment Spray System capable of taking suction from the RWST and transferring suction to the containment sump.

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

ACTION:

With one Containment Spray System inoperable, restore the inoperable Containment Spray System to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the inoperable Containment Spray System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each Containment Spray System shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position;

b. By verifying, that on recirculation flow, each pump develops a discharge pressure of greater than or equal to 250 psig when tested pursuant to Specification 4.0.5;

c. 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 Containment Pressure-High-3 (CSAS) test signal, and

   2)# Verifying that each spray pump starts automatically on a Containment Pressure-High-3 (CSAS) test signal.

d. At least once per 5 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

#The specified 18 month frequency may be waived for Cycle I provided the surveillance is performed prior to restart following the first refueling outage or June 1, 1986, whichever occurs first. The provisions of Specification 4.0.2 are reset from performance of this surveillance.

CONTAINMENT SYSTEMS

SPRAY ADDITIVE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The Spray Additive System shall be OPERABLE with:

a. A spray additive tank containing a volume of between 4340 and 4540 gallons of between 31% and 34% by weight NaOH solution, and

b. Two spray additive eductors each capable of adding NaOH solution from the chemical additive tank to a Containment Spray System pump flow.

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

ACTION:

With the Spray Additive System inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the Spray Additive System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.2.2 The Spray Additive System shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position;

b. At least once per 6 months by:

   1) Verifying the contained solution volume in the tank, and

   2) Verifying the concentration of the NaOH solution by chemical analysis.

c. At least once per 18 months during shutdown, by verifying that each automatic valve in the flow path actuates to its correct position on a Containment Pressure-High-3 (CSAS) test signal; and

d. At least once per 5 years by verifying

   1) Each eductor flow rate is greater than or equal to 52 gpm using the RWST as the test source throttled to 17 psig at the eductor inlet, and

   2) The lines between the spray additive tank and the eductors are not blocked by verifying flow.

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 1.3% $\Delta k/k$ is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. 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.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

MODERATOR TEMPERATURE COEFFICIENT (Continued)

The most negative MTC value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting MTC value -4.1×10^{-4} $\Delta k/k/°F$. The MTC value of -3.2×10^{-4} $\Delta k/k/°F$ represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting MTC value of -4.1×10^{-4} $\Delta k/k/°F$.

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 551°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the trip instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) the reactor vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The Boration Systems ensure that negative reactivity control is available during each MODE of facility operation. The components required to perform this function include: (1) borated water sources, (2) centrifugal charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, and (5) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature equal to or greater than 350°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The Boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1.3% $\Delta k/k$ after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 17,658 gallons of 7000 ppm borated water from the boric acid storage tanks or 83,745 gallons of 2350 ppm borated water from the RWST. With the RCS average temperature less than 350°F, only one boron injection flow path is required.

BORATION SYSTEMS (Continued)

With the RCS temperature below 200°F, one Boration 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 changes in the event the single Boron Injection System becomes inoperable.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE pump to be inoperable in MODES 4, 5, and 6 provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV or an RHR suction relief valve.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% $\Delta k/k$ after xenon decay and cooldown from 200°F to 140°F. This condition requires either 2968 gallons of 7000 ppm borated water from the boric acid storage tanks or 14,076 gallons of 2350 ppm borated water from the RWST.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 11.0 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 one Boration System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

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) the potential effects of rod misalignment on associated accident analyses are limited. 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. Verification that the Digital Rod Position Indicator agrees with the demanded position within ± 12 steps at 24, 48, 120 and 228 steps withdrawn for the Control Banks and 18, 210 and 228 steps withdrawn for the Shutdown Banks provides assurance that the Digital Rod Position Indicator is operating correctly over the full range of indication. Since the Digital Rod Position System does not indicate the actual shutdown rod position between 18 steps and 210 steps, only points in the indicated ranges are picked for verification of agreement with demanded position. Shutdown and control rods are positioned at 225 steps or higher for fully withdrawn.

MOVABLE CONTROL ASSEMBLIES (Continued)

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 and a restriction in THERMAL POWER. These restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety 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 safety analyses. Measurement with T_{avg} greater than or equal to 551°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 LCOs are satisfied.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) maintaining the minimum DNBR in the core at or above the safety analysis DNBR limits during normal operation and in short-term transients, and (2) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definition of certain hot channel and peaking factors as used in these specifications are as follows:

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

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

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelopes of 2.32 and 2.50 for OFA and VANTAGE 5, respectively, times the normalized axial peaking factor are not exceeded during either normal operation or in the event of xenon redistribution following power changes.

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

The limits on AXIAL FLUX DIFFERENCE (AFD) are given in Specification 3.2.1. Two modes of operation are permissible. One mode is Normal Operation, where the applicable AFD limit is defined by Specification 3.2.1.a. The AFD limit for this mode of operation is a +3, -12% target band about the target flux difference. After extended load following maneuvers, the AFD limits may result in restrictions in the maximum allowed power to quarantee operation with $F_Q(Z)$ less than its limiting value. To prevent this occurrence, another operating mode which

3/4.2.1 AXIAL FLUX DIFFERENCE (Continued)

restricts the AFD to a relatively small target band and does not allow significant changes in power level has been defined. This mode is called RESTRICTED AFD OPERATION, which restricts the AFD to a $\pm 3\%$ target band about the target flux difference and restricts power levels to between APL^{ND} and either APL^{RAFDO} or 100% of RATED THERMAL POWER, whichever is less. Prior to entering RESTRICTED AFD OPERATION, a 24-hour waiting period at a power level ($\pm 2\%$) above APL^{ND} and below that allowed by Normal Operation is necessary. During this time period load changes and control rod motion are restricted to that allowed by the RESTRICTED AFD OPERATION procedure. After the waiting period, RESTRICTED AFD OPERATION is permitted.

Although it is intended that the plant will be operated with the AFD within the target band required by Specification 3.2.1 about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1-hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits of Figure 3.2-1 while at THERMAL POWER levels between 50% and 90% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the 1 minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 90% of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and 90% and between 15% and 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

Figure B 3/4.2-1 shows a typical monthly target band.

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.

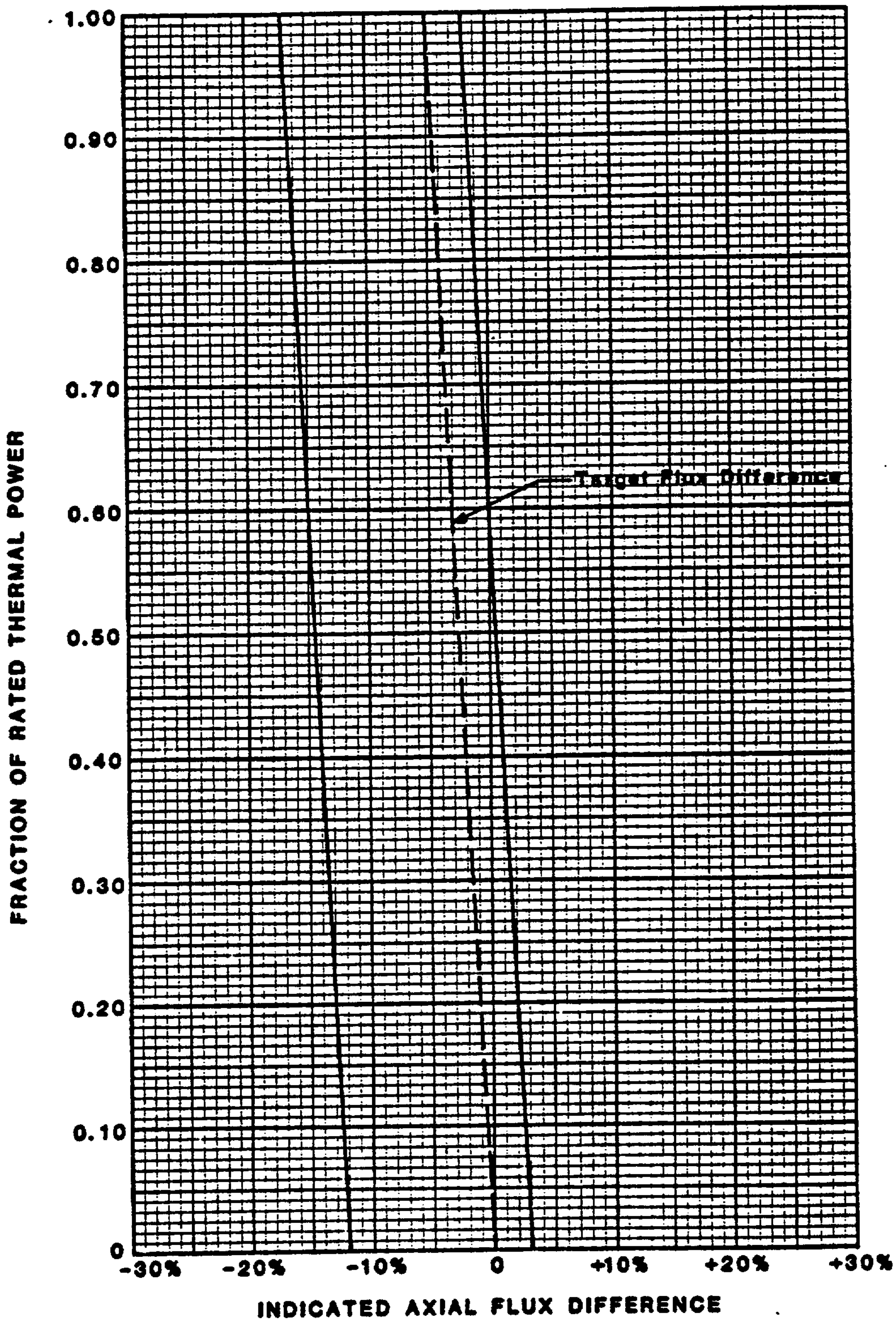


FIGURE B 3/4.2-1

TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. 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 $\pm$ 12 steps, indicated, from the group demand position.

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

c. The control rod insertion limits of Specification 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 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.

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

When $F^N_{\Delta H}$ is measured (i.e., inferred), no additional allowances are necessary prior to comparison with the limits of Section 3.2.3. An error allowance of 4% has been included in the limits of Section 3.2.3.

Specifications 3.2.2 and 3.2.3 contain the F_Q and F-delta-H limits applicable to VANTAGE 5 fuel. The OFA fuel is analyzed to lower limits since it will have experienced burnup, thereby reducing the attainable OFA-specific hot channel factors such that the expected peak power levels and peak radial power of the OFA fuel will be much less than that necessary to approach the OFA F_Q and F-delta-H analysis limits.

Margin between the safety analysis DNBR limits (1.42 and 1.45 for the Optimized fuel thimble and typical cells, respectively, and 1.61 and 1.69 for the VANTAGE 5 thimble and typical cells) and the design DNBR limits (1.33 and 1.35 for the Optimized fuel thimble and typical cells and 1.33 and 1.34 for the VANTAGE 5 thimble and typical cells, respectively) is maintained. A fraction of this margin is utilized to accommodate the transition core DNBR penalty

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

(12½% for VANTAGE 5 fuel) and the appropriate fuel rod bow DNBR penalty (less than 1.5% per WCAP-8691, Rev. 1). The margin between design and safety analysis DNBR limits of 6.3% for Optimized fuel and 17.4% for VANTAGE 5 fuel includes greater than 3% margin for both Optimized fuel and VANTAGE 5 fuel for plant design flexibility.

The hot channel factor $F_Q^M(z)$ is measured periodically and increased by a cycle and height dependent power factor appropriate to either Normal Operation or RESTRICTED AFD OPERATION, $W(z)_{NO}$ or $W(z)_{RAFDO}$, to provide assurance that the limit on the hot channel factor, $F_Q(z)$, is met. $W(z)_{NO}$ accounts for the effects of normal operation transients and was determined from expected power control maneuvers over the full range of burnup conditions in the core. $W(z)_{RAFDO}$ accounts for the more restrictive operating limits required by RESTRICTED AFD OPERATION which result in less severe transient values. The W(z) functions are provided in the Peaking Factor Limit Report per Specification 6.9.1.9.

Provisions to account for the possibility of decreases in margin to the $F_Q(z)$ limit during intervals between surveillances are provided. Any decrease in the minimum margin to the $F_Q(z)$ limit compared to the minimum margin determined from the previous flux map is determined by comparing the ratio of:

$$\underset{\text{over } z}{\text{maximum}} \left(\frac{F_Q^M(z)}{K(z)} \right)$$

taken from the current map to the same ratio from the previous map. The ratios to be compared from the two flux maps do not need to be calculated at identical z locations. Increases in this ratio indicate that the minimum margin to the $F_Q(z)$ limit has decreased and that additional penalties must be applied to the measured $F_Q(z)$ to account for further decreases in margin that could occur before the next surveillance. More frequent surveillances may also be substituted for the additional penalty.

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. A

3/4.2.4 QUADRANT POWER TILT RATIO (Continued)

limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2-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 control rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the movable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8.

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters is maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain the safety analysis DNBR limit throughout each analyzed transient. The indicated T_{avg} value of 592.6°F and the indicated pressurizer pressure value of 2220 psig correspond to analytical limits of 595.2°F and 2202 psig respectively, with allowance for measurement uncertainty.

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.

When RCS flow rate is measured, no additional allowances are necessary prior to comparison with the limits of Section 3.2.5. A measurement uncertainty of 2.2% (including 0.1% for feedwater venturi fouling) for RCS total flow rate has been allowed for in determination of the design DNBR value. The measurement uncertainty for the RCS total flow rate is based upon performing a precision heat balance and using the result to normalize the RCS flow rate indicators. Potential fouling of the feedwater venturi which might not be detected could bias the result from the precision heat balance in a non-conservative manner. Therefore, an inspection is performed on the feedwater venturi each refueling outage.

3/4.5.1 ACCUMULATORS

The OPERABILITY of each Reactor Coolant System (RCS) accumulator ensures that a sufficient volume of borated water will be immediately forced into the core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one accumulator is not available and prompt action is required to place the reactor in a MODE where this capability is not required. In order to perform check valve surveillance testing per 4.0.5 or 4.4.6.2.2 above 1000 psig RCS pressure, one accumulator isolation valve may be closed for up to 2 hours in mode 3 only.

The requirement to verify accumulator isolation valves shut with power removed from the valve operator when the pressurizer is solid ensures the accumulators will not inject water and cause a pressure transient when the Reactor Coolant System is on solid plant pressure control.

3/4.5.2, 3/4.5.3, and 3/4.5.4 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

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

ECCS SUBSYSTEMS (Continued)

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE charging pump to be inoperable in MODES 4 and 5 and in MODE 6 with the reactor vessel head on, provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV or RHR suction relief valve. In addition, the requirement to verify all Safety Injection pumps to be inoperable in MODE 4, in MODE 5 with the water level above the top of the reactor vessel flange, and in MODE 6 with the reactor vessel head on and with the water level above the top of the reactor vessel flange, provides assurance that the mass addition can be relieved by a single PORV or RHR suction relief valve.

With the water level not above the top of the reactor vessel flange and with the vessel head on, Safety Injection pumps may be available to mitigate the effects of a loss of decay heat removal during partially drained conditions.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensure, that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance Requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses. The Surveillance Requirements for leakage testing of ECCS check valves ensure that a failure of one valve will not cause an inter-system LOCA. The Surveillance Requirement to vent the ECCS pump casings and accessible, i.e., can be reached without personnel hazard or high radiation dose, discharge piping ensures against inoperable pumps caused by gas binding or water hammer in ECCS piping.

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the refueling water storage tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that: (1) sufficient water is available within containment to permit recirculation cooling flow to the core, and (2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes assuming all the control rods are out of the core. These assumptions are consistent with the LOCA analyses.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 44 TO FACILITY OPERATING LICENSE NO. NPF-30

UNION ELECTRIC COMPANY
CALLAWAY PLANT, UNIT 1
DOCKET NO. STN 50-483

1.0 INTRODUCTION

By letter dated October 25, 1988, Union Electric Company made application to modify the Technical Specifications (TS) for the Callaway plant to permit Cycle 4 operation. The significant changes incorporate increased peaking factors (F-delta H and Fq) and a positive moderator temperature coefficient (MTC). Other changes include an increase in the boron concentration of the refueling water storage tank (RWST) and the reactor coolant system (RCS) accumulators, and an increased concentration of sodium hydroxide in the spray additive tank.

The proposed Cycle 4 TS change values for the increased peaking factors (F-delta H and Fq) and a positive moderator temperature coefficient (MTC) were used in the previous Cycle 3 reload safety analyses which were reviewed and accepted by the NRC staff. The proposed values, however, were not incorporated in the TS in Cycle 3.

The Cycle 3 core contained three different types of fuel: low parasitic (LOPAR), optimized fuel assemblies (OFA's) and Vantage 5 (V-5). It was the first application of V-5 fuel at the Callaway plant.

2.0 RELOAD DESCRIPTION

The Callaway Cycle 4 reload will retain 9 OFA's and 96 V-5 fuel assemblies from the previous cycle and add 88 new, 17 x 17, V-5 fuel assemblies. Sixty-four of the new assemblies will have a 3.6 percent enrichment of U-235 and 24 assemblies will have with an enrichment of 4.0 percent U-235. The 88 added assemblies will have 10,112 integral fuel burnable absorbers (IFBA's).

Axial blankets are part of the design of V-5 fuel assemblies. This design feature was approved generically via the staff review of Westinghouse Topical Report WCAP-10444-P-A (Reference 1). The axial blankets were not used in the Cycle 3 reload and this will be their first use at the Callaway plant.

The reference cycle for this reload is Cycle 3.

3.0 EVALUATION

The Cycle 3 safety analysis, as a result of the hydraulic flow characteristics of three different fuel assembly types in the core, included a transition core departure from nucleate boiling ratio (DNBR) penalty of 2 percent for OFA, and 12.5 percent for V-5 assemblies. This circumstance results from the tendency of an increased core flow to the LOPAR assemblies, which, of the three fuel assembly designs, have a relatively lower hydraulic resistance .

LOPAR assemblies will not be in the Cycle 4 core. Therefore, the V-5 assemblies and the OFA's will have less flow diversion and hence an improved performance with regard to DNBR and peak clad temperature (PCT). With a mixture of two fuel designs, no penalty in DNBR is assigned to the fuel of lower hydraulic resistance, OFA, in Cycle 4. The 12.5 percent transition core DNBR penalty for the V-5 fuel is assumed in Cycle 4 and is conservative. The staff finds this penalty acceptable for addressing mixed core effects.

The licensee has also advised that debris-filter bottom nozzles (DFBN) will be used in Cycle 4. The DFBN is a revised version of the current 17 x 17 nozzle design and includes an improved pattern of flow holes that:

1. Reduces the passage of debris into the fuel assembly,

2. Maintains the structural integrity of the current nozzle design,

3. Maintains the hydraulic performance of the current design.

The staff finds this use acceptable as the hydraulic performance of the current nozzles is maintained.

The Cycle 3 safety analysis included a full power enthalpy rise hot channel factor (F-delta H) of 1.65 (with measurement uncertainty) for the V-5 assemblies, a maximum heat flux hot channel factor (Fq) of 2.5, and a positive moderator temperature coefficient of +5 pcm/F from 0-70 percent power and decreasing linearly to 0 pcm/F at 100 percent power. This safety analysis was reviewed and accepted by the NRC staff in Reference 2. The staff finds the proposed peaking factors and positive MTC acceptable for the Cycle 4 reload.

With a positive MTC, the reactor coolant system (RCS) boron concentration is expected to increase. The Cycle 3 safety analysis included a reanalysis of the boron dilution transient to incorporate this increase. For the Cycle 4 reload, the licensee reanalyzed the boron dilution transient. This reanalysis incorporates a revised methodology that includes the effects of density compensation on the dilution flow rates. The licensee stated that the applicable safety criteria were met. The staff finds this approach acceptable and further notes that this improvement would not have affected Cycle 3 as this cycle's TS for the MTC was not changed.

To support longer cycles and the implementation of a positive MTC while retaining the required shutdown margin, it has been proposed that the TS for the RWST and the RCS accumulators boron concentration ranges be increased to 2350-2500 ppm and 2300-2500 ppm, respectively. The accident analysis in the Final Safety Analysis Report (FSAR) was evaluated for the above-mentioned increased boron concentrations. The licensee determined that there was no adverse effect on the FSAR results caused by increasing the RWST and the RCS accumulator boron concentration bands. The staff has reviewed the evaluation and agrees with the conclusion.

The containment spray system reduces the iodine and particulate product inventories in the containment post-LOCA atmosphere. A minimum pH of 8.5 in the containment sump is necessary to ensure long-term retention of iodine in solution. On the basis of the increased boron concentrations, higher sodium hydroxide concentrations (31-34 percent by weight) are needed to achieve the minimum long-term post-LOCA containment sump pH of 8.5. The licensee has considered the post-accident effects of the increased sodium hydroxide on the following:

Hydrogen generation in containment
Environmental qualification of electrical equipment in containment
Radiological consequences
Potential precipitation of boric acid

It was determined that the increased sodium hydroxide did not adversely affect these considerations. The staff has reviewed the evaluation and find it acceptable.

4.0 TECHNICAL SPECIFICATION CHANGES

The TS proposed changes are described below:

TS 3.1.1.3 is revised to include a positive MTC.

TS 3.1.2.5 and TS 3.5.5 are revised to include a new minimum boron concentration for the RWST of 2350 ppm.

TS 3.1.2.6 is revised to include a new boron concentration range for the RWST.

TS 3.2.1, TS 3.2.2, TS Fig 3.2.2, TS 4.2.2, TS 4.2.2.2, TS 4.2.2.3 and TS 4.2.2.4 are revised to include the increased heat flux hot channel factor, Fq.

TS 4.2.1.4 is revised to include a calculated value, rather than zero value, at the end-of-cycle life in determining axial offset. The use of the calculated value is more realistic and represents a small change.

TS 3.2.3 is revised to include the increased F-delta H.

TS 3.5.1 is revised to include a new boron concentration range for each reactor coolant system accumulator.

TS 3.6.2.2 is revised to reflect a new weight percent range of sodium hydroxide in the spray additive tank.

The proposed TS are consistent with the safety analyses and are therefore acceptable.

5.0 SUMMARY

The staff has reviewed the accident analysis evaluation presented in the Callaway Cycle 4 reload submittal and has concluded that the proposed reload and associated TS are acceptable.

6.0 ENVIRONMENTAL CONSIDERATION

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

7.0 CONCLUSION

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

8.0 REFERENCES

1. Davidson, S.L. and Kramer, W.R.; (Ed.) "Reference Core Report VANTAGE 5 Fuel Assembly," WCAP-10444-P-A, September 1985.

2. License Amendment No. 28, dated October 9, 1987.

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