① See correction ltr of 10/30/87
② See correction ltr of 12/10/87

October 9, 1987

Docket No. 50-483

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

Dear Mr. Schnell:

The Commission has issued the enclosed Amendment No. 28 to Facility Operating License No. NPF-30 for the Callaway Plant, Unit 1. This amendment is in response to your application dated March 31, 1987, as supplemented by letters dated April 15, June 5, June 18, July 16, July 28, August 7, August 13, August 31, September 9, and October 6, 1987.

The amendment modifies the technical specifications to support a transition from a Westinghouse 17 x 17 low-parasitic fuel assembly and optimized fuel assembly fueled core to a Westinghouse 17 x 17 Vantage 5 (V-5) fuel assembly fueled core.

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,

/s/

Thomas W. Alexion, Project Manager
Project Directorate III-3
Division of Reactor Projects

Enclosures:
1. Amendment No. 28 to License No. NPF-30
2. Safety Evaluation

cc w/enclosures:
See next page

*SEE PREVIOUS CONCURRENCE

| | | | | |
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| Date: | 09/26/87 | 09/28/87 | 09/29/87 | 10/ 9/87 |

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| Date: | 10/9 /87 |

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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. 28
License No. NPF-30

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

   A. The application for amendment filed by Union Electric Company (the licensee) dated March 31, 1987, as supplemented by letters dated April 15, June 5, June 18, July 16, July 28, August 7, August 13, August 31, September 9 and October 6, 1987 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. 28, 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

David L. Wigginton, Acting Director
Project Directorate III-3
Division of Reactor Projects

Attachment:
Changes to the Technical Specifications

Date of Issuance: October 9, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 28

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 |
|---|---|
| I | I |
| II | II |
| V | V |
| XX | XX |
| 1-5 | 1-5 |
| 1-6 | 1-6 |
| 1-7 | 1-7 |
| 2-2 | 2-2 |
| 2-4 | 2-4 |
| 2-7 | 2-7 |
| 2-8 | 2-8 |
| 2-9 | 2-9 |
| 2-10 | 2-10 |
| B 2-1 | B 2-1 |
| B 2-5 | B 2-5 |
| B 2-6 | B 2-6 |
| - | B 2 2-6(a) |
| 3/4 1-19 | 3/4 1-19 |
| 3/4 2-1 | 3/4 2-1 |
| 3/4 2-2 | 3/4 2-2 |
| - | 3/4 2-2(a) |
| 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(b) |
| 3/4 2-14 | 3/4 2-14 |
| 3/4 3-12 | 3/4 3-12 |
| 3/4 3-12a | 3/4 3-12a |
| 3/4 5-1 | 3/4 5-1 |
| B 3/4 2-1 | B 3/4 2-1 |
| B 3/4 2-2 | B 3/4 2-2 |
| 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 |
| 6-21 | 6-21 |
| 6-22 | 6-22 |

DEFINITIONS

| SECTION | PAGE |
|---|---|
| **1.0 DEFINITIONS** | |

SECTION | PAGE

DEFINITIONS (Continued)

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

# INDEX

## LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

# INDEX

## ADMINISTRATIVE CONTROLS

ADMINISTRATIVE CONTROLS

QUADRANT POWER TILT RATIO

1.25 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.26 RATED THERMAL POWER shall be a total core heat transfer rate to the reactor coolant of 3411 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.27 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its Trip Setpoint at the channel sensor until loss of stationary gripper coil voltage.

REPORTABLE EVENT

1.28 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

RESTRICTED AFD OPERATION

1.29 RESTRICTED AFD OPERATION (RAFDO) limits the AXIAL FLUX DIFFERENCE (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% RATED THERMAL POWER, whichever is less. $APL^{ND}$ and $APL^{RAFDO}$ are defined in Specifications 3.2.1 and 4.2.2.3, respectively. RAFDO may be entered at the discretion of the licensee.

SHUTDOWN MARGIN

1.30 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full-length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.31 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

SLAVE RELAY TEST

1.32 A SLAVE RELAY TEST shall be the energization of each slave relay and verification of OPERABILITY of each relay. The SLAVE RELAY TEST shall include a continuity check, as a minimum, of associated testable actuation devices.

SOLIDIFICATION

1.33 SOLIDIFICATION shall be the conversion of wet wastes into a form that meets shipping and burial ground requirements.

SOURCE CHECK

1.34 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

STAGGERED TEST BASIS

1.35 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

THERMAL POWER

1.36 THERMAL POWER shall be the total core heat transfer rate to the reactor coolant.

TRIP ACTUATING DEVICE OPERATIONAL TEST

1.37 A TRIP ACTUATING DEVICE OPERATIONAL TEST shall consist of operating the Trip Actuating Device and verifying OPERABILITY of alarm, interlock and/or trip functions. The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include adjustment, as necessary, of the Trip Actuating Device such that it actuates at the required Setpoint within the required accuracy.

UNIDENTIFIED LEAKAGE

1.38 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

UNRESTRICTED AREA

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

VENTILATION EXHAUST TREATMENT SYSTEM

1.40 A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Features (ESF) Atmospheric Cleanup Systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.41 VENTING shall be any controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

WASTE GAS HOLDUP SYSTEM

1.42 A WASTE GAS HOLDUP SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting Reactor Coolant System off-gases from the Reactor Coolant System and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

TABLE 1.1

FREQUENCY NOTATION

| NOTATION | FREQUENCY |
|---|---|
| S | At least once per 12 hours. |
| D | At least once per 24 hours. |
| W | At least once per 7 days. |
| M | At least once per 31 days. |
| Q | At least once per 92 days. |
| SA | At least once per 184 days. |
| R | At least once per 18 months. |
| S/U | Prior to each reactor startup. |
| N.A. | Not applicable. |
| P | Completed prior to each release. |

# 2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

## 2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature ($T_{avg}$) shall not exceed the limits shown in Figure 2.1-1 for four loop operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.1.

MODES 3, 4, and 5:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.1.

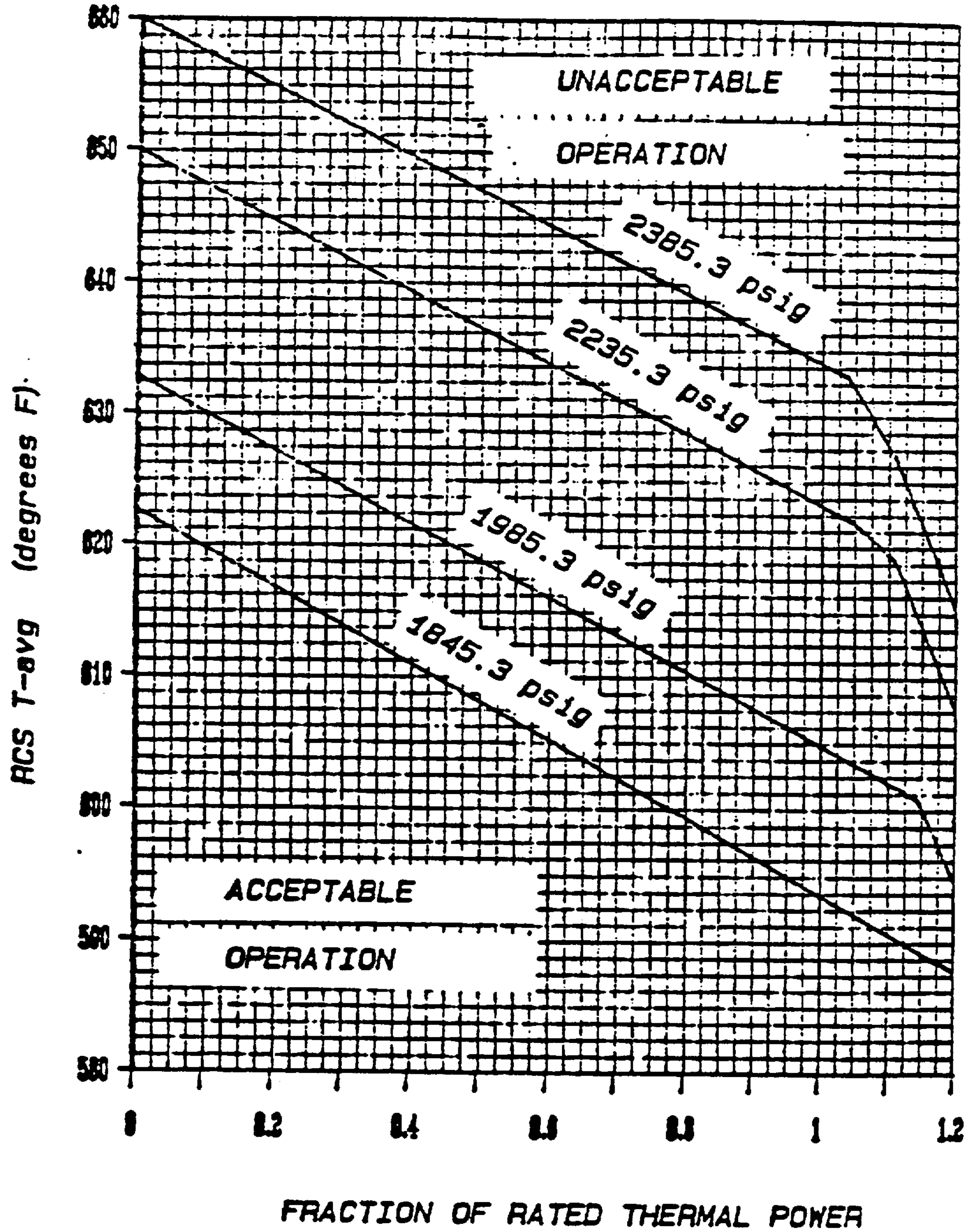


FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION

## 2.2 LIMITING SAFETY SYSTEM SETTINGS

### REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The Reactor Trip System Instrumentation and Interlocks Setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

a. With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 2.2-1, adjust the Setpoint consistent with the Trip Setpoint value.

b. With the Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, either:

   1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 2.2-1 and determine within 12 hours that Equation 2.2-1 was satisfied for the affected channel, or

   2. Declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

   Equation 2.2-1 $$Z + R + S \leq TA$$

   Where:

   Z = The value from Column Z of Table 2.2-1 for the affected channel,

   R = The "as measured" value (in percent span) of rack error for the affected channel,

   S = Either the "as measured" value (in percent span) of the sensor error, or the value from Column S (Sensor Error) of Table 2.2-1 for the affected channel, and

   TA = The value from Column TA (Total Allowance) of Table 2.2-1 for the affected channel.

## REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | TOTAL ALLOWANCE (TA) | SENSOR ERROR Z | (S) | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 1. Manual Reactor Trip | N.A. | N.A. | N.A. | N.A. | N.A. |
| 2. Power Range, Neutron Flux | | | | | |
| a. High Setpoint | 7.5 | 4.56 | 0 | $\leq$109% of RTP* | $\leq$112.3% of RTP* |
| b. Low Setpoint | 8.3 | 4.56 | 0 | $\leq$25% of RTP* | $\leq$28.3% of RTP* |
| 3. Power Range, Neutron Flux, High Positive Rate | 2.4 | 0.5 | 0 | $\leq$4% of RTP* with a time constant $\geq$2 seconds | $\leq$6.3% of RTP* with a time constant $\geq$2 seconds |
| 4. Power Range, Neutron Flux, High Negative Rate | 2.4 | 0.5 | 0 | $\leq$4% of RTP* with a time constant $\geq$2 seconds | $\leq$6.3% of RTP* with a time constant $\geq$2 seconds |
| 5. Intermediate Range, Neutron Flux | 17.0 | 8.41 | 0 | $\leq$25% of RTP* | $\leq$35.3% of RTP* |
| 6. Source Range, Neutron Flux | 17.0 | 10.01 | 0 | $\leq 10^5$ cps | $\leq 1.6 \times 10^5$ cps |
| 7. Overtemperature ΔT | 9.3 | 6.47 | 1.83 +1.24*** | See Note 1 | See Note 2 |
| 8. Overpower ΔT | 5.7 | 1.46 | 1.8 | See Note 3 | See Note 4 |
| 9. Pressurizer Pressure-Low | 5.0 | 2.21 | 2.0 | $\geq$1885 psig | $\geq$1874 psig |
| 10. Pressurizer Pressure-High | 7.5 | 4.96 | 1.0 | $\leq$2385 psig | $\leq$2400 psig |
| 11. Pressurizer Water Level-High | 8.0 | 2.18 | 2.0 | $\leq$92% of instrument span | $\leq$93.8% of instrument span |
| 12. Reactor Coolant Flow-Low | 2.5 | 1.38 | 0.6 | $\geq$90% of loop minimum measured flow** | $\geq$88.8% of loop minimum measured flow** |

*RTP = RATED THERMAL POWER
**Minimum Measured Flow = 95,660 gpm
***Two Allowances (temperature and pressure, respectively)

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE $\Delta T$

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \left\{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

Where:

| | | |
|---|---|---|
| $\Delta T$ | = | Measured $\Delta T$ by RTD Manifold Instrumentation; |
| $\frac{1 + \tau_1 S}{1 + \tau_2 S}$ | = | Lead-lag compensator on measured $\Delta T$; |
| $\tau_1$, $\tau_2$ | = | Time constants utilized in lead-lag compensator for $\Delta T$, $\tau_1$ = 8 s, $\tau_2$ = 3 s; |
| $\frac{1}{1 + \tau_3 S}$ | = | Lag compensator on measured $\Delta T$; |
| $\tau_3$ | = | Time constant utilized in the lag compensator for $\Delta T$, $\tau_3$ = 0 s; |
| $\Delta T_0$ | = | Indicated $\Delta T$ at RATED THERMAL POWER: |
| $K_1$ | = | 1.15; |
| $K_2$ | = | 0.0251/°F; |
| $\frac{1 + \tau_4 S}{1 + \tau_5 S}$ | = | The function generated by the lead-lag compensator for $T_{avg}$ dynamic compensation; |
| $\tau_4$, $\tau_5$ | = | Time constants utilized in the lead-lag compensator for $T_{avg}$, $\tau_4$ = 28 s, $\tau_5$ = 4 s; |
| T | = | Average temperature, °F; |
| $\frac{1}{1 + \tau_6 S}$ | = | Lag compensator on measured $T_{avg}$; |
| $\tau_6$ | = | Time constant utilized in the measured $T_{avg}$ lag compensator, $\tau_6$ = 0 s; |

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

$T'$ $\leq$ 588.4°F (Referenced $T_{avg}$ at DESIGN THERMAL POWER);

$K_3$ = 0.00116;

$P$ = Pressurizer pressure, psig;

$P'$ = 2235 psig (Nominal RCS operating pressure);

$S$ = Laplace transform operator, $s^{-1}$;

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron 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 -35% and + 6%, $f_1(\Delta I) = 0$, where $q_t$ and $q_b$ are percent DESIGN THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of DESIGN THERMAL POWER;

(ii) For each percent that the magnitude of $q_t - q_b$ exceeds -35%, the $\Delta T$ Trip Setpoint shall be automatically reduced by 1.91% of its value at DESIGN THERMAL POWER; and

(iii) For each percent that the magnitude of $q_t - q_b$ exceeds +6%, the $\Delta T$ Trip Setpoint shall be automatically reduced by 1.89% of its value at DESIGN THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.3% of $\Delta T$ span.

## TABLE 2.2-1 (Continued)

## TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_o \left\{K_4 - K_5 \left(\frac{\tau_7 S}{1 + \tau_7 S}\right) \left(\frac{1}{1 + \tau_6 S}\right) T - K_6 \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T''\right] - f_2(\Delta I)\right\}$$

Where:

| | | |
|---|---|---|
| $\Delta T$ | = | Measured ΔT by RTD Manifold Instrumentation; |
| $\frac{1 + \tau_1 S}{1 + \tau_2 S}$ | = | Lead-lag compensator on measured ΔT; |
| $\tau_1$, $\tau_2$ | = | Time constants utilized in lead-lag compensator for ΔT, $\tau_1$, = 8 s., $\tau_2$ = 3 s; |
| $\frac{1}{1 + \tau_3 S}$ | = | Lag compensator on measured ΔT; |
| $\tau_3$ | = | Time constant utilized in the lag compensator for ΔT, $\tau_3$ = 0 s; |
| $\Delta T_o$ | = | Indicated ΔT at RATED THERMAL POWER; |
| $K_4$ | = | 1.080; |
| $K_5$ | = | 0.02/°F for increasing average temperature and 0 for decreasing average temperature; |
| $\frac{\tau_7 S}{1 + \tau_7 S}$ | = | The function generated by the rate-lag compensator for $T_{avg}$ dynamic compensation; |
| $\tau_7$ | = | Time constant utilized in the rate-lag compensator for $T_{avg}$, $\tau_7$ = 10 s; |
| $\frac{1}{1 + \tau_6 S}$ | = | Lag compensator on measured $T_{avg}$; |
| $\tau_6$ | = | Time constant utilized in the measured $T_{avg}$ lag compensator, $\tau_6$ = 0 s; |

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

| | | |
|---|---|---|
| $K_6$ | = | 0.0065/°F for T > T" and $K_6$ = 0 for T ≤ T"; |
| T | = | Average Temperature, °F; |
| T" | = | Indicated $T_{avg}$ at DESIGN THERMAL POWER (Calibration temperature for ΔT instrumentation, ≤ 588.4°F); |
| S | = | Laplace transform operator, $s^{-1}$; and |
| $f_2(\Delta I)$ | = | 0 for all ΔI. |

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 3.3% of ΔT span.

## 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 1.32 and 1.34 for thimble and typical cells, respectively, for OFA, and 1.32 and 1.33 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 nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$, of 1.49 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 expression:

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

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 $f_1$ ($\Delta I$) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature $\Delta T$ trips will reduce the setpoints to provide protection consistent with 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.

## Intermediate and Source Range, Neutron Flux

The Intermediate and Source Range, Neutron Flux trips provide core protection during reactor startup to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. 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 equivalent to approximately 25% of RATED THERMAL POWER unless manually blocked when P-10 becomes active.

## 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 Pressurizer High and Low Pressure trips. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water and includes dynamic compensation for piping delays from the core to the loop temperature detectors, (2) pressurizer pressure, and (3) axial power distribution. With normal axial power distribution, this Reactor trip limit is always below the core Safety Limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the Reactor trip is automatically reduced according to the notations in Table 2.2-1.

Delta-$T_0$, as used in the Overtemperature and Overpower $\Delta T$ trips, represents the 100% RTP value as measured by the plant for each loop. This normalizes each loop's $\Delta T$ trips to the actual operating conditions existing at the time of measurement, thus forcing the trip to reflect the equivalent full power conditions as assumed in the accident analyses. These differences in vessel $\Delta T$ can arise due to several factors, the most prevalent being measured RCS loop flows greater than Minimum Measured Flow, and slightly asymmetric power distributions between quadrants. While RCS loop flows are not expected to change with cycle life, radial power redistribution between quadrants may occur, resulting in small changes in loop specific vessel $\Delta T$ values. Accurate determination of the loop specific vessel $\Delta T$ value should be made when performing the Incore/Excore quarterly recalibration and under steady state conditions (i.e., power distributions not affected by Xe or other transient conditions).

## Overpower $\Delta T$

The Overpower $\Delta T$ trip provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions, limits the required range for Overtemperature $\Delta T$ trip, and provides

Overpower $\Delta T$ (Continued)

a backup to the High Neutron Flux trip. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water, and (2) rate of change of temperature for dynamic compensation for piping delays from the core to the loop temperature detectors, to ensure that the allowable heat generation rate (kW/ft) is not exceeded. The Overpower $\Delta T$ trip provides protection to mitigate the consequences of various size steam breaks as reported in WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Releases."

Delta-$T_0$, as used in the Overtemperature and Overpower $\Delta T$ trips, represents the 100% RTP value as measured by the plant for each loop. This normalizes each loop's $\Delta T$ trips to the actual operating conditions existing at the time of measurement, thus forcing the trip to reflect the equivalent full power conditions as assumed in the accident analyses. These differences in vessel $\Delta T$ can arise due to several factors, the most prevalent being measured RCS loop flows greater than Minimum Measured Flow, and slightly asymmetric power distributions between quadrants. While RCS loop flows are not expected to change with cycle life, radial power redistribution between quadrants may occur, resulting in small changes in loop specific vessel $\Delta T$ values. Accurate determination of the loop specific vessel $\Delta T$ value should be made when performing the Incore/Excore quarterly recalibration and under steady state conditions (i.e., power distributions not affected by Xe or other transient conditions).

Pressurizer Pressure

In each of the pressurizer pressure channels, there are two independent bistables, each with its own Trip Setting to provide for a High and Low Pressure trip thus limiting the pressure range in which reactor operation is permitted. The Low Setpoint trip protects against low pressure which could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

On decreasing power the Low Setpoint trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

The High Setpoint trip functions in conjunction with the pressurizer relief and safety valves to protect the Reactor Coolant System against system overpressure.

Pressurizer Water Level

The Pressurizer High Water Level trip is provided to prevent water relief through the pressurizer safety valves. On decreasing power the Pressurizer High Water Level trip is automatically blocked by P-7 (a power level of

Pressurizer Water Level (Continued)

approximately 10% of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

Reactor Coolant Flow

The Low Reactor Coolant Flow trips provide core protection to prevent DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

On increasing power above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent) an automatic Reactor trip will occur if the flow in more than one loop drops below 90% of nominal full loop flow. Above P-8 (a power level of approximately 48% of RATED THERMAL POWER) an automatic Reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow. Conversely, on decreasing power between P-8 and P-7 an automatic Reactor trip will occur on low reactor coolant flow in more than one loop and below P-7 the trip function is automatically blocked.

## 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 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 551°F, and

b. All Reactor Coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the rod 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.

b. With the rod drop times within limits but determined with three reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to less than or equal to 66% of RATED THERMAL POWER.

### SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of full-length rods shall be demonstrated through measurement prior to reactor criticality:

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.

SHUTDOWN ROD INSERTION LIMIT

## LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be fully withdrawn.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With a maximum of one shutdown rod not fully withdrawn, except for surveillance testing pursuant to Specification 4.1.3.1.2, within 1 hour either:

a. Fully withdraw the rod, or

b. Declare the rod to be inoperable and apply Specification 3.1.3.1.

## SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown rod shall be determined to be fully withdrawn:

a. Within 15 minutes prior to withdrawal of any rods in Control Bank A, B, C, or D during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

#With $K_{eff}$ greater than or equal to 1.

## 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. ±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 the equal to the

$$\underset{\text{over } Z}{\text{maximum}} \left[ \frac{2.32 \ * \ K(Z)}{F_Q^M(Z) \ * \ W(Z)_{NO}} \ * \ 100 \right]$$

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 0% at the end of the cycle life. The provisions of Specification 4.0.4 are not applicable.

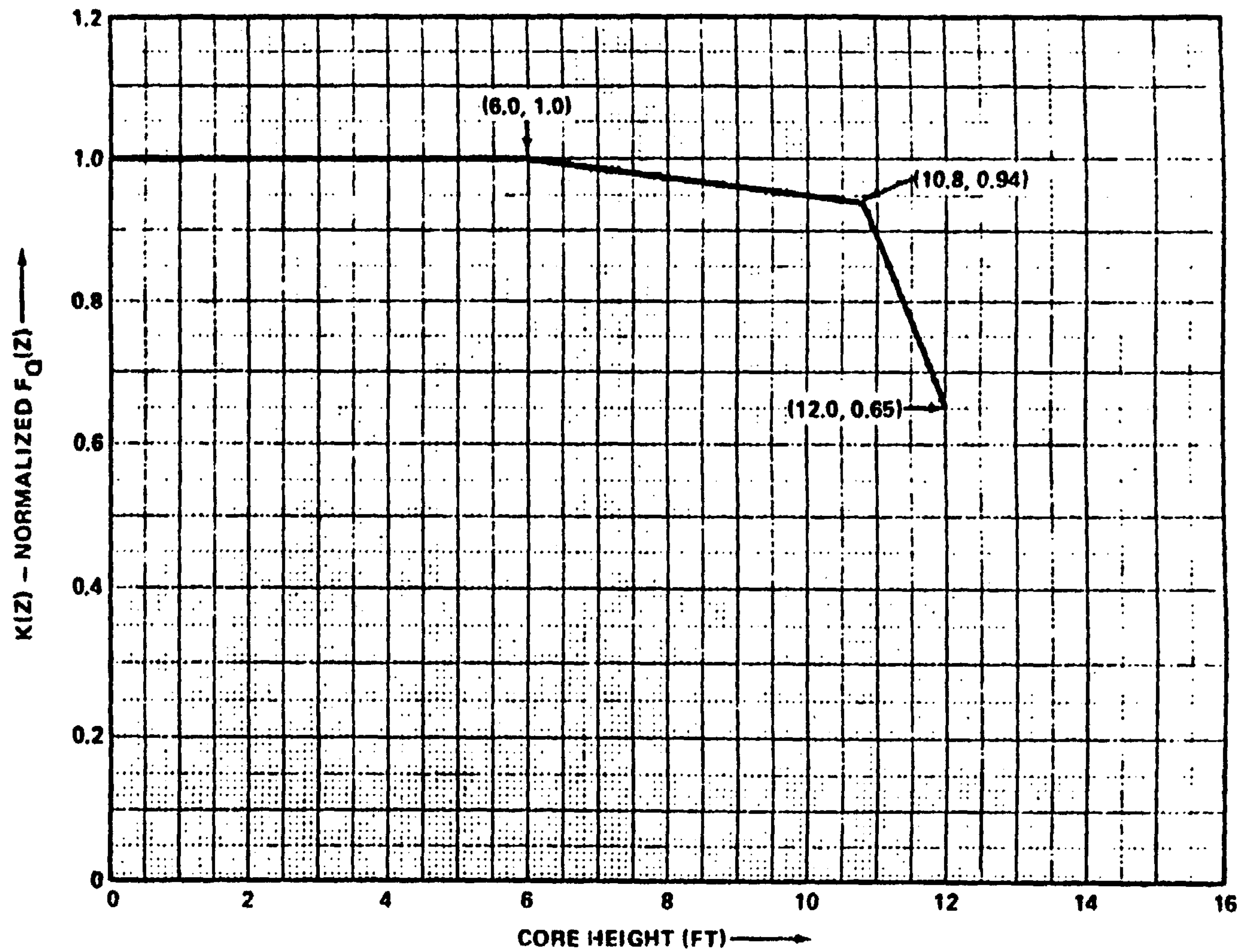


FIGURE 3.2-2.
K(Z) – NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT

## 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:

$$F_Q^M(z) \leq \frac{2.32 \times K(z)}{P \times W(z)_{NO}} \text{ for } P > 0.5$$

$$F_Q^M(z) \leq \frac{2.32 \times K(z)}{W(z)_{NO} \times 0.5} \text{ for } P \leq 0.5$$

where $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, 2.32 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:

$$\left[ \left( \text{max. over } z \text{ of} \left( \frac{F_Q^M(z) \;\; x \;\; W(z)\ NO}{\frac{2.32}{P} \;\; x \;\; K(z)} \right) \right) -1 \right] x\ 100 \quad \text{for} \quad P \geq 0.5$$

$$\left[ \left( \text{max. over } z \text{ of} \left( \frac{F_Q^M(z) \;\; x \;\; W(z) NO}{\frac{2.32}{0.5} \;\; x \;\; K(z)} \right) \right) -1 \right] x\ 100 \quad \text{for} \quad P < 0.5$$

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.

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:

$$APL^{RAFDO} = \underset{\text{over } z}{\text{minimum}} \left[\frac{2.32 \times K(z)}{F_Q^M(z) \times W(z)}\right]_{RAFDO} \times 100\%$$

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 2.32. $K(z)$ 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:

$$F_Q^M(z) \leq \frac{2.32 \times K(z)}{P \times W(z)_{RAFDO}} \quad \text{for } P > APL^{ND}$$

where: $F_Q^M(z)$ is the measured $F_Q(z)$. The $F_Q$ limit is 2.32. $K(z)$ 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:

$$\left[ \left( \text{max. over } z \text{ of} \left( \frac{F_Q^M(z) \times W(z)_{RAFDO}}{\frac{2.32}{P} \times K(z)} \right) \right) -1 \right] \times 100 \text{ for } P \geq APL^{ND}$$

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.

## 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:

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

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

$F^N_{\Delta H}$ = 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 $\leq$ 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.

TABLE 3.2-1

DNB PARAMETERS

| PARAMETER | LIMITS Four loops in Operation |
|---|---|
| Indicated Reactor Coolant System $T_{avg}$ | $\leq$ 592.6°F |
| Indicated Pressurizer Pressure | $\geq$ 2220 psig* |
| Calculated Reactor Coolant System Total Flow Rate | $\geq$ 382,630**GPM |

* 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 RATED THERMAL POWER.

** The calculated value of RCS total flow rate shall be used since uncertainties of 2.2% for flow (including 0.1% for feedwater venturi fouling) measurement have been included in the above operating limit.

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

| FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | ANALOG CHANNEL OPERATIONAL TEST | TRIP ACTUATING DEVICE OPERATIONAL TEST | ACTUATION LOGIC TEST | MODES FOR WHICH SURVEILLANCE IS REQUIRED |
|---|---|---|---|---|---|---|
| 18. Reactor Trip System Interlocks (Continued) | | | | | | |
| d. Power Range Neutron Flux, P-10 | N.A. | R(4) | R | N.A. | N.A. | 1, 2 |
| e. Turbine Impulse Chamber Pressure, P-13 | N.A. | R | R | N.A. | N.A. | 1 |
| 19. Reactor Trip Breaker | N.A. | N.A. | N.A. | M (7, 11) | N.A. | 1, 2, 3*, 4*, 5* |
| 20. Automatic Trip and Interlock Logic | N.A. | N.A. | N.A. | N.A. | M(7) | 1, 2, 3*, 4*, 5* |

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

*Only if the Reactor Trip System breakers happen to be closed and the Control Rod Drive System is capable of rod withdrawal.

#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.

##Below P-6 (Intermediate Range Neutron Flux interlock) Setpoint.

###Below P-10 (Low Setpoint Power Range Neutron Flux interlock) Setpoint.

(1) If not performed in previous 31 days.

(2) Comparison of calorimetric to excore power indication above 15% of RATED THERMAL POWER. Adjust excore channel gains consistent with calorimetric power if absolute difference is greater than 2%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(3) Single point comparison of incore to excore AXIAL FLUX DIFFERENCE above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(4) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) Detector plateau curves shall be obtained, evaluated and compared to manufacturer's data. For the Intermediate Range and Power Range Neutron Flux channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(6) Incore - Excore Calibration, above 75% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1. Determination of the loop specific vessel $\Delta T$ value should be made when performing the Incore/Excore quarterly recalibration, under steady state conditions.

(7) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS. The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the Undervoltage and Shunt Trip Attachments of the Reactors Trip Breakers.

(8) Deleted

(9) Quarterly surveillance in MODES 3*, 4*, and 5* shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window. Quarterly surveillance shall include verification of the Boron Dilution Alarm Setpoint of less than or equal to an increase of twice the count rate within a 10-minute period.

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

(10) Setpoint verification is not required.

(11) Following maintenance or adjustment of the Reactor trip breakers, the TRIP ACTUATING DEVICE OPERATIONAL TEST shall include independent verification of the Undervoltage and Shunt trips.

(12) At least once per 18 months during shutdown, verify that on a simulated Boron Dilution Doubling test signal the normal CVCS discharge valves will close and the centrifugal charging pumps suction valves from the RWST will open within 30 seconds.

(13) CHANNEL CALIBRATION shall include the RTD bypass loops flow rate.

(14) Each channel shall be tested at least every 92 days on a STAGGERED TEST BASIS.

(15) The surveillance frequency and/or MODES specified for these channels in Table 4.3-2 are more restrictive and, therefore, applicable.

(16) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the Undervoltage and Shunt Trip circuits for the Manual Reactor Trip function. The test shall also verify the OPERABILITY of the Bypass Breaker trip circuit.

## 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 1900 and 2100 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 in HOT SHUTDOWN 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 in HOT SHUTDOWN 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.

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*Pressurizer pressure above 1000 psig.

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.

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 envelope of 2.32 times the normalized axial peaking factor is 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.

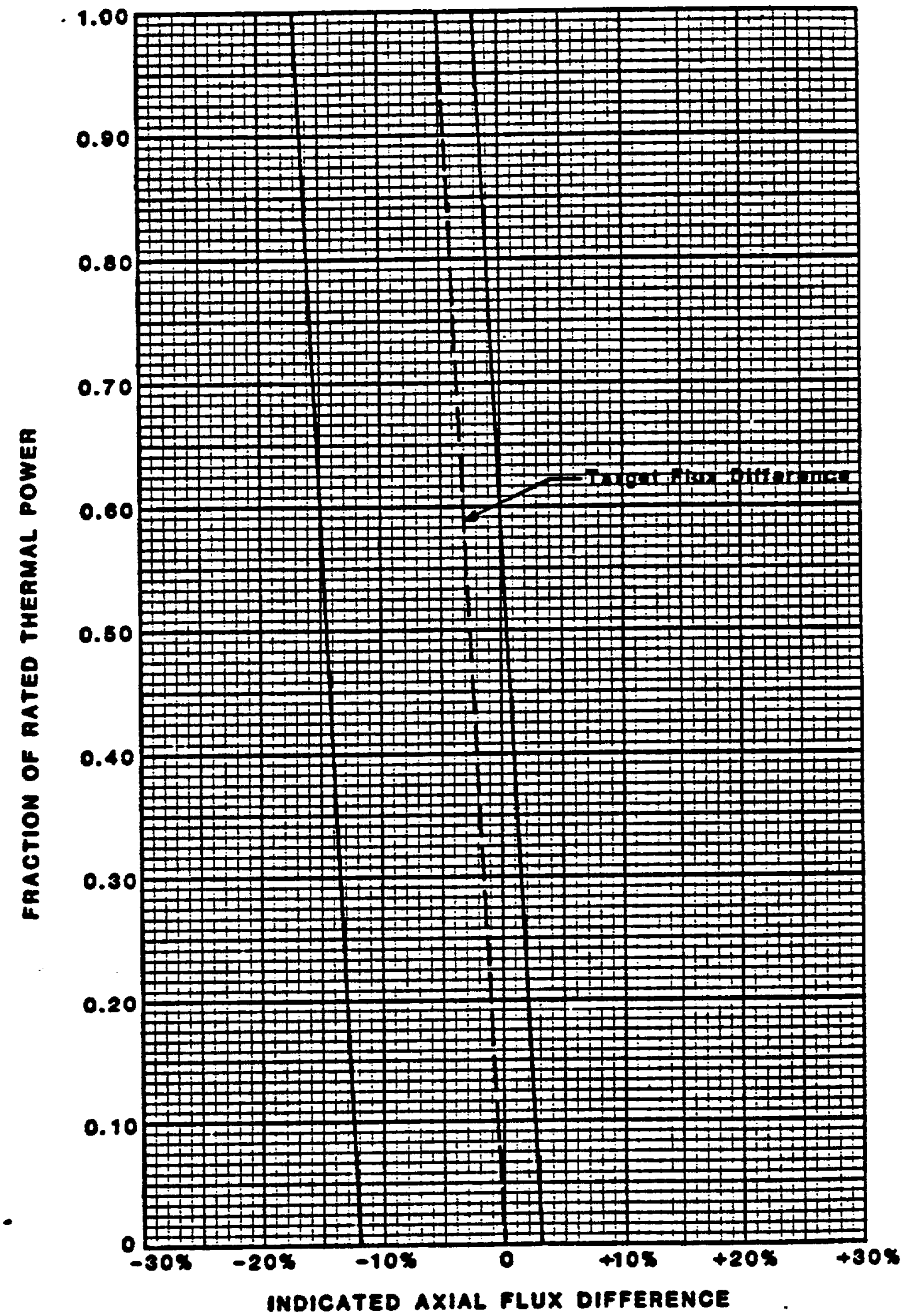


**FIGURE B 3/4.2-1**

**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.

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 (1.32 and 1.34 for the Optimized fuel thimble and typical cells and 1.32 and 1.33 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 (2% for Optimized fuel, 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 7% for Optimized fuel and 18% for VANTAGE 5 fuel includes greater than 3% margin for Optimized fuel and 4% margin for VANTAGE 5 fuel for plant design flexibility.

The hot channel factor $F^M_Q(z)$ 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

POWER DISTRIBUTION LIMITS

BASES

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

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 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 nonconservative manner. Therefore, an inspection is performed on the feedwater venturi each refueling outage.

PEAKING FACTOR LIMIT REPORT

6.9.1.9 The W(z) functions for Normal and RESTRICTED AFD OPERATION and the value for $APL^{ND}$ (as required) shall be established for at least each reload core and shall be maintained available in the Control Room. The limits shall be established and implemented on a time scale consistent with normal procedural changes.

The analytical methods used to generate the W(z) functions and $APL^{ND}$ shall be those previously reviewed and approved by the NRC*. If changes to these methods are deemed necessary, they will be evaluated in accordance with 10 CFR 50.59 and submitted to the NRC for review and approval prior to their use if the change is determined to involve an unreviewed safety question or if such a change would require amendment of previously submitted documentation.

A report containing the W(z) functions, as a function of core height (and burnup, if applicable) and $APL^{ND}$ shall be provided to the NRC Document Control Desk with copies to the Regional Administrator and the Resident Inspector within 30 days of their implementation.

SPECIAL REPORTS

6.9.2 Special Reports shall be submitted to the Regional Administrator of the NRC Regional Office within the time period specified for each report.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level;

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety;

c. All REPORTABLE EVENTS;

d. Records of surveillance activities; inspections and calibrations required by these Technical Specifications;

*WCAP-8385, "Power Distribution Control and Load Following Procedures," WCAP-9272-A, "Westinghouse Reload Safety Evaluation Methodology," and WCAP-10216-P-A, "Relaxation of Constant Axial Offset Control / $F_Q$ Surveillance Technical Specification."

RECORD RETENTION (Continued)

e. Records of changes made to the procedures required by Specification 6.8.1;

f. Records of radioactive shipments;

g. Records of sealed source and fission detector leak tests and results; and

h. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report;

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories;

c. Records of radiation exposure for all individuals entering radiation control areas;

d. Records of gaseous and liquid radioactive material released to the environs;

e. Records of transient or operational cycles for those unit components identified in Table 5.7-1;

f. Records of reactor tests and experiments;

g. Records of training and qualification for current members of the unit staff;

h. Records of in-service inspections performed pursuant to these Technical Specifications;

i. Records of quality assurance activities required by the QA Program;

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59;

k. Records of meetings of the ORC and the NSRB;

l. Records of the service lives of all hydraulic and mechanical snubbers required by Specification 3.7.8 including the date at which the service life commences and associated installation and maintenance records;

m. Records of secondary water sampling and water quality; and

n. Records of analysis required by the Radiological Environmental Monitoring Program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 28 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 March 31, 1987, Union Electric Company (the licensee) made application to amend the license of the Callaway Plant, Unit 1, in order to reload and operate the unit for Cycle 3. In support of the application, the licensee provided a report entitled, "Safety Evaluation for the Callaway Plant Transition to Westinghouse 17 x 17 Vantage 5 Fuel." Further information was provided in response to NRC requests. Also provided were proposed Technical Specification changes to assure the safe operation of the plant.

## 2.0 DISCUSSION

### VANTAGE 5 FUEL

The Cycle 3 core loading for Callaway will consist of 28 17x17 Westinghouse VANTAGE 5 (V-5) fuel assemblies in Region 5A (3.6 w.o U-235), 32 V-5 assemblies in Region 5B (3.8 w/o U-235), and 36 V-5 assemblies in Region 5C (4.2 w/o U-235) in addition to the 13 Low Parasitic (LOPAR) and 84 Optimized Fuel Assemblies (OFA's) remaining in the core. This is the first application of V-5 fuel. Eventually, an all V-5 fueled core is anticipated in Cycle 5. A number of the V-5 assemblies will employ integral fuel burnable absorbers (IFBA's). Other V-5 design features include intermediate flow mixer (IFM) grids, reconstitutable top nozzles, extended burnup, and axial blankets. All but the latter will be employed in Callaway Cycle 3.

The V-5 fuel design is a modification of the current 17x17 LOPAR and OFA approved fuel designs. The V-5 fuel assembly design was approved generically via the staff review of Westinghouse Topical Report WCAP-10444-P-A. The staff approval of the V-5 fuel design was subject to 13 conditions. The conditions and how they are addressed in the Callaway submittal are as follows:

(1) The statistical convolution method described in WCAP-10125 for the evaluation of initial fuel rod to nozzle growth gap has not been approved. This method should not be used in VANTAGE 5.

Callaway: In Section 3.0 of the safety evaluation provided as Attachment 1 of the V-5 licensing submittal, the fuel rod performance discussion indicates that the worst case fabrication tolerances were used to determine the initial fuel rod to nozzle growth gap for fuel rod irradiation growth. This is in compliance with the above condition.

(2) For each plant application, it must be demonstrated that the loss-of-coolant accident (LOCA)/seismic loads considered in WCAP-9401 bound the plant in question; otherwise additional analysis will be required to demonstrate the fuel assembly structural integrity.

Callaway: As addressed on page 11 of Attachment 1 of the V-5 licensing submittal, the LOCA/seismic loads considered in WCAP-9401 bound the Callaway Plant.

(3) An irradiation demonstration program should be performed to provide early confirmation performance data for the VANTAGE 5 design.

Callaway: A summary of V-5 demonstration programs is provided on pages 2 and 3 of Attachment 1 of the V-5 licensing submittal. We consider that the demonstration described there adequately fulfills the requirements of the condition.

(4) For those plants using the improved thermal design procedure (ITDP), the restrictions enumerated in Section 4.1 of this report must be addressed and information regarding measurement uncertainties must be provided.

Callaway: As stated on page 4 of Section 2.0 of Attachment 1 of the V-5 licensing submittal, Westinghouse has addressed the restrictions enumerated in SER Section 4.1. This and the information regarding measurement uncertainties for the Callaway Plant were provided to the NRC in ULRNC-1227, dated December 13, 1985.

(5) The WRB-2 correlation with a departure from nucleate boiling ratio (DNBR) limit of 1.17 is acceptable for application to 17x17 VANTAGE 5 fuel. Additional data and analysis are required when applied to 14x14 or 15x15 fuel with an appropriate DNBR limit. The applicability range of WRB-2 is specified in Section 4.2.

Callaway: Callaway will utilize 17x17 V-5 fuel.

(6) For 14x14 and 15x15 VANTAGE 5 fuel designs, separate analyses will be required to determine a transitional mixed-core penalty. The mixed-core penalty and plant-specific safety margin to compensate for the penalty should be addressed in the plant Technical Specifications Bases.

Callaway: This condition does not apply to the 17x17 V-5 fuel which will be utilized in Callaway.

(7) Plant-specific analyses should be performed to show that the DNBR limit will not be violated with the higher value of $F_{\Delta H}$.

Callaway: The reload analysis was performed in conformance with this condition and is discussed in the section on thermal-hydraulic design, below.

(8) The plant-specific safety analysis for the steam system piping failure event should be performed with the assumption of loss of offsite power if that is the most conservative case.

Callaway: This event was evaluated for Callaway Plant and there was no change in the results from the OFA analysis. That is, the analysis in the OFA licensing submittal is bounding.

(9) With regard to the reactor-coolant-system (RCS) pump shaft seizure accident, the fuel failure criterion should be the 95/95 DNBR limit. The mechanistic method mentioned in WCAP-10444 is not acceptable.

Callaway: The 95/95 DNBR fuel failure criterion was used for this accident as discussed in Section 15.3.3 of Attachment 5 of the V-5 licensing submittal.

(10) If a positive moderator-temperature coefficient (MTC) is intended for VANTAGE 5, the same positive MTC consistent with the plant technical specifications should be used in the plant-specific safety analysis.

Callaway: The plant technical specifications do not allow a positive MTC for Cycle 3. A positive MTC analysis was used for the safety analysis. This will be discussed below in the section on accident analysis.

(11) The LOCA analysis performed for the reference plant with higher $F_Q$ of 2.55 has shown that the peak-clad-temperature (PCT) limit of 2200°F is violated during transitional mixed core configuration. Plant-specific LOCA analysis must be done to show that with the appropriate value of $F_Q$, the 2200°F criterion can be met during use of transitional mixed core.

Callaway: The plant-specific LOCA analysis performed with an $F_Q$ of 2.50 will be discussed below in the section on accident analysis.

(12) Our SER on Westinghouse's extended burnup topical report WCAP-10125 is not yet complete; the approval of the VANTAGE 5 design for operation to extended burnup levels is contingent on NRC approval of WCAP-10125. However, VANTAGE 5 fuel may be used to those burnups to which Westinghouse fuel is presently operating. Our review of the Westinghouse extended burnup topical report has not identified any safety issues with operation to the burnup value given in the extended burnup report.

WCAP-10125 has been approved. The extended burnup methodology contained in this report was applied to and is discussed with satisfactory results in Section 3.0 of Attachment 1 of the V-5 licensing submittal.

(13) Recently, a vibration problem has been reported in a French reactor having 14-foot fuel assemblies; vibration below the fuel assemblies in the lower portion of the reactor vessel is damaging the movable incore instrumentation probe thimbles. The staff is currently evaluating the implications of this problem to other cores having 14-foot long fuel bundle assemblies. Any limitations to the 14-foot core design resulting from the staff evaluation must be addressed in plant-specific evaluations.

Callaway: Callaway plant has 12-foot long fuel assembly bundles.

## 3.0 EVALUATION

The staff finds that Callaway Plant satisfies the conditions of the generic approval of the V-5 fuel design, as discussed above. Therefore, the use of V-5 fuel in Callaway is found to be acceptable. Additional discussion of this acceptability is contained in the following sections of this evaluation.

### FUEL DESIGN

In Section 3.0 of Attachment 1 of the V-5 licensing submittal, the licensee presents the results of its evaluation showing the fuel rod designs satisfy the requirements of the Standard Review Plan (SRP), NUREG-0800. Considered are fuel rod performance, grid assemblies, the reconstitutable top nozzle and bottom nozzle design features, axial blankets, and mechanical compatibility of the V-5/OFA/LOPAR assemblies. The staff concurs with the licensee's evaluation that indicates satisfactory performance in each of these areas.

In addition, the licensee's evaluation indicates that fuel rod bow for V-5 fuel is predicted to be no greater than that for OFA rods, since both fuel designs have the same fuel rod diameter, zircaloy grid spacings, and grid designs. In addition, no difference in rod bow magnitude or frequency is expected in fuel rods containing IFBA's. No indications of abnormal rod bow have been observed in the inspections performed on test IFBA rods to date. Rod growth measurements were also within predicted bounds.

Also addressed is fuel rod wear. Due to the OFA, LOPAR and V-5 fuel assembly designs employing different grids, there is unequal axial pressure distribution between the assemblies. Results of wear inspection and analysis referred to in the licensee's evaluation revealed that the V-5 fuel assembly wear characteristic was similar to the 17x17 OFA when the data were normalized to the test duration time. It is concluded that the V-5 fuel rod wear would be less than the maximum wear depth established for the 17x17 OFA at end-of-life (EOL).

An evaluation of the V-5 fuel assembly structural integrity considering the lateral effects of a LOCA and seismic accident was also presented. The results indicate that the V-5 fuel has more margin in withstanding the faulted condition transient load than the OFA design.

As a result of the analyses presented, the staff concludes that the mechanical aspects of the V-5 fuel design and mixed core loading are acceptable.

### NUCLEAR DESIGN

The evaluation of transition and equilibrium cycle V-5 cores presented in WCAP-10444-P-A as well as the transition and equilibrium core evaluations for the Callaway Plant demonstrate that the impact of implementing V-5 causes no significant change to the physics characteristics of the Callaway core beyond the normal range of variations seen from cycle to cycle. This is expected because the V-5 fuel assemblies are very similar to the previously approved OFA and LOPAR assemblies, particularly the former.

Generation of core characteristics for the accident analysis was performed using the approved methods contained in WCAP-9272-P-A. The analyses were performed at a core thermal power of 3565 MWt, 15% steam generator tube plugging, full power enthalpy rise hot channel factor ($F_{\Delta}H$) of 1.65 for the V-5 assemblies, a maximum heat flux hot channel factor ($F_Q$) of 2.5 and a positive moderator temperature coefficient of +5 pcm/°F from 0-70% power and decreasing linearly to 0 pcm/°F at 100% power. The reload submittal does not request any licensing changes for the above parameters, so the physics characteristics generated for the accident analyses will be conservative for Cycle 3. The $F_Q$ (and to a lesser degree, $F_{\Delta H}$) assumption leads to a reevaluation of the LOCA analyses. The positive moderator assumption is a greater perturbation relative to the need for reevaluation for most of the other accidents than the variations caused by normal core loading variations or the transition to V-5 fuel.

Because the values of the reactor parameter chosen for the nuclear analysis are conservative for Cycle 3 and because the calculations were performed with approved methods, the staff finds the predictions of the core characteristics acceptable for the accident analyses.

THERMAL-HYDRAULIC DESIGN

The analyses of the V-5 fuel include use of the approved WRB-2 departure-from-nucleate-boiling (DNB) correlation and the ITDP. The existing LOPAR and OFA fuel use the ITDP and the WRB-1 DNB correlation. The WRB-2 DNB correlation takes credit for the improvement in the accuracy of initial heat flux prediction over previous DNB correlations and the V-5 fuel assembly mixing vane design. A DNBR limit of 1.17 is applicable to both the WRB-1 and WRB-2 correlations. The ITDP statistically combines uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters such that there is at least a 95 percent probability with a 95 percent confidence level that the minimum DNBR will be greater than or equal to 1.17 for the limiting power rod. Plant parameter uncertainties are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the DNBR limit, establishes a DNBR value which must be met in plant safety analyses. Since the parameter uncertainties are considered in determining the design DNBR value, the plant safety analyses are performed using values of input parameters without uncertainties. For Cycle 3, the minimum required design DNBR values are 1.32 for thimble cold wall cells (three fuel rods and a thimble tube) and 1.34 for typical cells (four fuel rods) for LOPAR/OFA fuel and 1.32 for thimble and 1.33 for typical cells for the V-5 fuel.

In addition to the above, a plant-specific DNBR margin has been allowed for in the analyses. These consist of safety analysis DNBR limits of 1.42 for the thimble and 1.45 for the typical cells for LOPAR/OFA fuel, and 1.61 and 1.64 for thimble and typical cells, respectively, for V-5 fuel. This extra margin is utilized as follows: 2% for OFA and 12.5% for V-5 is allotted to accommodate the transition core DNBR penalty, an appropriate fuel rod bow DNBR penalty which is less than 1.5%, and the remaining 3% DNBR margin in the OFA fuel and 4% DNBR margin in the V-5 fuel is reserved for flexibility in the design.

The transition core DNBR allowance is acceptable. In view of this, and the other considerations discussed above, the staff finds the thermal-hydraulic design of Callaway Cycle 3 acceptable.

ACCIDENT ANALYSIS (NUCLEAR FUEL)

The $F_{\Delta H}$ for the OFA fuel during the transition cycles remains 1.55 as in the current licensing basis. The non-LOCA transients reanalyzed for the V-5 fuel assumed a full power $F_{\Delta H}$ of 1.65. Safety analyses which incorporate explicit modeling of this peaking factor were reanalyzed. They are:

- locked rotor (rods-in-DNB calculation)
- partial loss of forced reactor coolant flow
- complete loss of forced reactor coolant flow
- startup of an inactive reactor coolant loop at an incorrect temperature

The positive moderator coefficient described previously led to the reanalysis of the following transients:

- control rod assembly withdrawal from subcritical
- control rod assembly withdrawal at power
- loss of reactor coolant flow (partial and complete)
- locked rotor
- loss of external load/turbine trip
- RCS depressurization
- loss of normal feedwater
- feedline break
- control rod ejection
- loss of non-emergency AC power

The analyses were based on a +5 pcm/°F moderator temperature coefficient, which is assumed to remain constant for variations in temperature. Exceptions are rod ejection and rod withdrawal from subcritical which are based on an MTC of +5 pcm/°F at zero power nominal average temperature and which, due to moderator temperature feedback modeled in the TWINKLE diffusion-theory code, becomes less positive for higher temperatures.

The non-LOCA safety analyses not listed above include those resulting in excessive heat removal from the reactor coolant system for which a large negative moderator temperature coefficient is more limiting, and those for which heatup effects following reactor trip are not sensitive to the moderator temperature coefficient. These transients were evaluated to remain applicable for positive moderator temperature coefficient. The boron dilution transient was reanalyzed to incorporate the increases in RCS boron concentration expected with a positive moderator temperature coefficient. Because the positive moderator temperature coefficient (PMTC) has not been incorporated into the Callaway Cycle 3 fuel design, changes to the refueling water storage tank (RWST) and accumulator boron concentrations are not required and the existing related technical specifications remain applicable.

The Callaway OFA transition core report provided safety analysis justification to support up to 10% plant total steam generator tube plugging - not to exceed 10% in any single steam generator. All non-LOCA safety analyses reanalyzed for this report have incorporated any necessary changes to model 15% plant total steam generator tube plugging. No conclusions are made regarding the impact of this assumption on those transients not requiring explicit reanalysis for this report.

The analytical procedures and computer codes used for the non-LOCA transient analyses were established in previous analyses for both OFA and LOPAR fuel cores.

For each of the accidents reanalyzed, it was found that the appropriate safety criteria are met. The staff concurs with the licensee's determination of accidents which did not require reanalysis, and the conclusion that the existing analyses remain applicable for the proposed changes to the plant. Because of this, and the acceptable results of the accidents reanalyzed, the staff finds the non-LOCA accident analysis for Callaway Cycle 3 acceptable. It will support the value of the positive moderator coefficient used in future licensing actions.

The large break LOCA accident analysis for the Callaway Plant, applicable to a full core of V-5 assemblies, was performed to develop the Callaway Plant-specific peaking factor limits. The analysis was performed with the approved BASH model. The analysis assumed a full core of V-5 fuel. That this conservatively applies to transition cores was demonstrated in WCAP-10444-P-A.

Fuel assembly design specific analyses were performed with a version of the BART computer code, which accurately models mixed core cases during reflood. Westinghouse transition core designs, including a specific 17x17 LOPAR to VANTAGE 5 transition core case, were analyzed. For this case, BART modeled both fuel assembly types and predicted the reduction in axial flow at the appropriate elevations. As expected, the increase in hydraulic resistance for the VANTAGE 5 assembly was shown to produce a reduction in reflood steam flow rate for the VANTAGE 5 fuel at mixing vane grid elevations during the transition core period. This reduction in steam flow rate is partially offset by the fuel grid heat transfer enhancement predicted by the BART model during reflood. The various fuel assembly specific transition core analyses performed resulted in peak clad 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 during transition cores is 50°F (licensee's V-5 submittal). Once a full core of the VANTAGE 5 fuel is achieved, the large break LOCA analysis will apply without the crossflow penalty.

For breaks up to and including the double-ended severance of a reactor coolant pipe, the emergency core cooling system (ECCS) will meet the acceptance criteria as presented in 10 CFR 50.46. That is:

1. The calculated peak fuel element clad temperature is below the requirement of 2200°F.

2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1 percent of the total amount of Zircaloy in the reactor.

3. The localized cladding oxidation limit of 17 percent is not exceeded during or after quenching.

4. The core remains amenable to cooling during and after the break.

5. The core temperature is reduced and decay heat is removed for an extended period of time. This is required to remove the heat from the long-lived radioactivity remaining in the core.

In order to demonstrate that criterion 5 is met, a specific evaluation of the core subcriticality is made for each reload cycle. For Cycle 3 this evaluation demonstrates that there is sufficient boron available in the ECCS fluid to maintain the core subcritical for long-term core cooling.

The large break VANTAGE 5 LOCA analysis for the Callaway Plant, utilizing the BASH model, resulted in a peak clad temperature of 2004°F for the limiting break case at a peaking factor of 2.50. The maximum local metal-water reaction is 5.17 percent, and the total metal-water reaction is less than 0.3 percent for all cases analyzed. The clad temperature transients turn around at a time when the core geometry is still amenable to cooling.

The impact of crossflow for transition core cycles is conservatively evaluated at most a 50°F effect, which is easily accommodated in the margin to 10 CFR 50.46 limits.

In a letter dated April 15, 1987, the licensee addressed reliability enhancements which were implemented to alleviate a problem encountered during a LOCA analysis using BASH for a plant similar in design to Callaway. The limiting large break LOCA was reanalyzed with the result of a 10°F increase in the peak clad temperature from 2004°F to 2014°F. Since this is well below the limiting value of 2200°F, the staff concludes that the results of the large break LOCA acceptably meet the requirements of 10 CFR 50.46.

The small break LOCA was reanalyzed using the approved NOTRUMP model. The analysis was performed for a spectrum of cold leg breaks. The most limiting break size, 4 inches, resulted in a peak clad temperature of 1528°F. This is well below the required limits of 10 CFR 50.46. The staff, therefore, finds the results of the small break LOCA analysis acceptable.

SPENT FUEL POOL ANALYSIS

The licensee stated that the thermal-hydraulic analysis of the spent fuel pool was reanalyzed to assure that depleted V-5 assemblies can be stored in the spent fuel pool (SFP) with adequate cooling and without adverse impact on fuel building HVAC performance. A reeanalysis of the SFP cooling system was performed to assure that discharged spent fuel assemblies would be adequately cooled and pool boiling would not occur for both normal and off-normal conditions. This reeanalysis was performed to assess the impact on the SFP cooling system from the adoption of the V-5 fuel design, corresponding increased burnup levels, and plant operation at higher (3565 MWt) core thermal power. A reanalysis was also performed to assure that the plant operation at increased burnup levels associated with V-5 fuel would not adversely impact the proper operation of the fuel building HVAC system.

The licensee determined that the SFP cooling system will provide adequate cooling of discharged spent fuel assemblies to limit the SFP bulk temperatures for the maximum normal condition to below existing FSAR limits (140°F) and to assure that the spent fuel pool is not subject to bulk boiling for the abnormal condition (full core offload). The component cooling water system (CCWS) will adequately cool the SFP water based on the "maximum" normal heat duty capability of the CCWS heat exchanger without exceeding its design limit.

The staff performed independent analyses (calculations) considering the "maximum" heat loads to the SFP for the normal condition and for the abnormal (full core offload) condition, in accordance with the guidelines of the SRP, Branch Technical Position ASB 9-2, "Residual Decay Energy for Light Water Reactors for Fuel Pool Cooling and Cleanup System." The results of these analyses verify the licensee's findings that, under the maximum normal condition, the SFP bulk temperature would be kept below 140°F, assuming a single failure in the SFP cooling system which is an SRP Section 9.1.3 acceptance criterion to meet the requirements of General Design Criterion (GDC) 44. Further, the temperature of the SFP would be kept below boiling for the abnormal condition (full core offload), which is also an SRP Section 9.1.3 acceptance criterion. It was assumed in the staff analyses that adequate secondary side cooling is provided by the CCWS. The staff reviewed the licensee's reanalysis concerning the CCWS heat exchanger capability, under the "maximum" normal heat load condition and found that adequate secondary side cooling for the SFP cooling system was provided.

The staff also determined that the licensee's reanalysis demonstrates that the fuel building HVAC system has sufficient margin to maintain the building air temperature below 104°F and 50% relative humidity when the SFP temperature is kept below 140°F (maximum normal condition). This ensures that the fuel building's HVAC system is adequate to maintain an environment consistent with personnel comfort and safety. The ability for the fuel building HVAC to limit the accidental release of radioisotopes in the event of a fuel-handling accident to below acceptable limits is not impacted by the change to V-5 fuel and extended burnup as the amount of radioiodine gas from such an event is essentially the same.

The staff, therefore, finds that proper fuel building HVAC operation under normal and accident conditions is provided and, therefore, the criteria of SRP Section 9.4.2 and requirements of GDC 60 and 61 are met.

In summary, the staff concludes that the proposed Callaway licensing amendment concerning the Cycle 3 reload using Westinghouse V-5 fuel assemblies at the 3565 MWt reactor thermal power level meets the requirements of GDC 44, 60 and 61 with regard to SFP cooling capability and fuel-building HVAC performance, and is, therefore, acceptable.

ACCIDENT ANALYSIS (RADIOLOGICAL)

The licensee's accident analyses are reported in Appendices C and D of Attachment 5 of its March 31, 1987 submittal. The submittal was reviewed and found to have no significant differences from the previously accepted analysis presented for Cycle 2, with the following exceptions. First, the higher burnup fuel can result in increases in the quantities of some of the radionuclides in the core (e.g., cesium-134 and cesium-137) and/or in the primary coolant. Second, the higher burnup fuel can affect the internal pressure in the fuel rods and change the amount of volatile fission products (e.g., noble gases and radioiodines) that can be released from the fuel pellets into the gaps between the pellets and the cladding. To assess the effect of higher burnup, the doses from a postulated fuel-handling accident and a rod ejection accident were reevaluated by the staff. These accidents are the two principal design basis accidents whose doses could be increased by the use of the higher burnup fuel because they involve the release of gap activity. These accidents were previously evaluated in the Safety Evaluation Report (SER), dated October 1981, but for a lower burnup.

The methodology used by the staff to evaluate the fuel-handling accident was described in the SER and is based on positions of Regulatory Guide 1.25 and SRP Section 15.7.4. The staff assumed that a single fuel assembly was dropped in the fuel pool during refueling operations and that all of the fuel rods in the assembly were damaged, releasing radioactive materials from the gaps in the rods in one fuel assembly into the fuel pool. In the case of a fuel-handling accident outside of the containment, the radioactive materials that escaped from the pool were assumed to be released over a 2-hour period with most of the iodine activity reduced by engineered safety feature grade filtration. For the accident occurring inside the containment, a 20 percent mixing volume was used with a containment isolation time of 25 seconds. Although there is filtration in the reactor building, no credit was given for filtration in the dose calculation since the filter is classified as a non-engineered safety grade feature.

The evaluation of the fuel handling accident was performed in accordance with the methodology of Regulatory Guide 1.25, even though the conditions at the end of Cycle 3 will be beyond the bases stated in the guide. For assemblies with burnup up to 38,000 MWD/MTU batch average at discharge, Regulatory Guide 1.25 stipulates that 10 percent of the iodines and noble gases (with the exception of 30 percent for Kr-85) in the fuel assembly will be released from the gap and plenum volume once the cladding is perforated. Since the average burnup of fuel in the highest region at Callaway is anticipated to be 48,000 MWD/MTU, a larger release fraction was used for radioiodines. The release fraction for radioiodines was taken from a draft report entitled "Radioactive Gas Release from LWR Fuel" (NUREG/CR-2715). In determining the release fraction, the staff used a maximum allowable linear heat generation rate of 14.23 kilowatts per foot (KW/ft), and a burnup of 23,800 MWD/MTU for the highest power assembly in Cycle 3. The assumptions and parameters used in the analysis are listed in Table 1. The staff also notes that the methodology used for estimating doses is conservative (i.e., it is more likely that the estimated doses are higher than the actual doses that might be received).

The offsite doses for the postulated fuel-handling accident are listed in Table 2. Due to the larger release fraction used for the radioiodines, the doses to the thyroid are about 20% greater than the values in the SER. Since the potential doses for the fuel handling accidents remain less than 25% of the guideline values given in 10 CFR Part 100, the staff concludes that the fuel storage and handling systems will continue to meet the requirements of Item 3 of General Design Criterion 61 of 10 CFR Part 50, Appendix A.

The methodology used by the staff to evaluate the rod ejection accident is described in Chapter 15.4.2 of the SER. A release fraction of 0.2 was used for the radioiodines, and the offsite doses for the postulated rod ejection accident are listed in Table 2. The potential doses for the rod ejection accident remain less than 25% of the guideline values given in 10 CFR Part 100. The staff concludes that the Callaway design and the technical specification limits on primary-to-secondary coolant leakage provide reasonable assurance that the potential doses can be maintained well within the 10 CFR Part 100 exposure guidelines and, therefore, remain acceptable.

TABLE 1

Assumptions for Analyzing Fuel Handling Accident

| | |
|---|---|
| Power level | 3565 MWt |
| Release Fractions | |
| - iodines | 0.2* |
| - noble gases (except Kr-85) | 0.1 |
| - Kr-85 | 0.3 |
| Decay Time | 100 hr |
| Number of fuel assemblies affected | 1 |
| Volume of reactor building | $2.5 \times 10^6$ ft$^3$ |
| Number of fuel assemblies in core | 193 |
| Mixing volume (reactor building) | 20 percent |
| HVAC exhaust rate | |
| Reactor building | 20,000 cfm |
| Fuel building | 9,000 cfm |
| Filter efficiency (fuel building) | 90 percent |
| Reactor building isolation time | 25 sec |
| Activity release period | |
| Reactor building | 25 sec |
| Fuel building | 2 hr |

* This is an upper bound estimate.

TABLE 2

Radiological Consequences of Design Basis Accidents

| | Exclusion Area[1] 2-Hr Dose (Rems) | | Low Population Zone[2] 8-Hr Dose (Rems) | |
|---|---|---|---|---|
| Postulated Accident | Thyroid | Whole Body | Thyroid | Whole Body |
| Fuel Handling Accident | | | | |
| Inside fuel building | 5.2 | 0.2 | 0.7 | <0.1 |
| Inside containment | 1.0 | <0.1 | 0.1 | <0.1 |

| Postulated Accident | Exclusion Area[1] 2-Hr Dose (Rems) | | Low Population Zone[2] 8-Hr Dose (Rems) | |
|---|---|---|---|---|
| | Thyroid | Whole Body | Thyroid | Whole Body |
| Rod Ejection Accident | | | | |
| Secondary side leakage | 5.2 | 0.1 | 7.4 | <0.1 |
| Containment leakage | 23. | 0.1 | 3.4 | <0.1 |

[1]Exclusion area distance = 1200m.
[2]Low population zone distance = 4023 m.

In summary, the staff has reviewed the licensee's accident analyses and has reevaluated the doses from the two design basis accidents whose doses could be increased by the use of the higher burnup fuel. The potential doses for the fuel handling accident and the rod ejection accident remain less than 25% of the guideline values given in 10 CFR Part 100. The licensee's submittal meets the applicable regulatory guidance and requirements and is, therefore, acceptable.

## TECHNICAL SPECIFICATION CHANGES

The Technical Specification changes proposed for the Callaway Cycle 3 reload and their acceptability are as follows:

1. Figure 2.1-1. Revised Reactor Core Safety Limits. The new limits account for an increase in RCS core bypass flow due to V-5 fuel using the approved methodology described in WCAP-10444-P-A and WCAP-9272-P-A and are acceptable.

2. Table 2.2-1. Revised value for Delta To. The licensee revised the change in the original submittal to use the indicated ΔTo at full power. The staff finds this proposal technically sound and, therefore, acceptable. The licensee also proposed to add a surveillance to Table 4.3-1 which is acceptable.

3. Table 2.2-1. Revised values for the OP and OT setpoint parameters. These are calculated for the V-5 fuel design using approved methodology and are acceptable.

4. TS 2.1.1. Basis added the WRB-2 correlation to the DNBR limits for V-5 fuel. The staff concludes the basis changes adequately describe the design changes.

5. TS 3.2.1, 4.2.2.2, and 4.2.2.3. The changes to these specifications implement a method for monitoring the TS peaking factor. There are two parts to the changes. The first substitutes $F_Q$ surveillance for $F_{xy}$ surveillance. The methodology for $F_Q$ surveillance is described in

WCAP-10216-P-A. This methodology has been approved and is therefore acceptable for Callaway. $F_Q$ surveillance involves W(z) functions which are cycle-specific. The functions will be provided in a peaking factor limit report within 30 days of their implementation as specified in Section 6.9.1.9. The second part of the changes involves providing for a mode of restricted axial flux difference (AFD) operation which allows the reactor to operate with a very restrictive (AFD) band (± 3%) (RAFDO) to gain power generation capability if the normal AFD band of +3 to -120% does not support full power operation. Operation in the RAFDO mode is essentially steady-state operation with no maneuvering capability. This type of specification has been approved for the McGuire and Turkey Point power plants and has been approved via an Appendix to Section 4.3 to the SRP, NUREG-0800, and is, therefore, acceptable for Callaway.

As submitted, the technical specification for Callaway had three areas requiring modification. These were restricting applicability of the second footnote to specification 3.2.1 to restrict direct entry into and from RAFDO, and removing redundancy between Specifications 4.2.2.2.f.2) a and b and between 4.2.2.4.f.1 and 2. The licensee has provided satisfactory resolution to these concerns in a submittal dated July 16, 1987. With these modifications, the staff finds that the proposed technical specifications adequately implement $F_Q$ surveillance and RAFDO operation, and are, therefore, acceptable.

6. Table 3.2-1. Revised indicated DNB parameter for $F_{xy}$. This change reflects use of the ITDP discussed in the thermal-hydraulic design section above and is, therefore, acceptable.

7. T.S. 3.5.1. Revised the range of required volumes for the RCS accumulators by a small amount. This change facilitates plant operation, has negligible effect on the safety analysis, and is, therefore, acceptable.

8. Basis 3/4.2 and 3/4.2.1. Deleted definition of $F_{xy}(Z)$ and added discussion on normal and restricted AFD modes of operation. These changes properly support the changes made in Item 5 above and are, therefore, acceptable.

9. Basis 3/4.2.2 and 3/4.2.3. Revised discussion on design and safety analysis DNBR limits to include those for the V-5 fuel design, including the transition core penalty as discussed in the licensee's letter dated October 6, 1987. Replaced discussion on the $F_{xy}$ surveillance with discussion on the $F_Q$ surveillance. These changes appropriately clarify the DNBR limits for the fuel and discuss $F_Q$ surveillance and are, therefore, acceptable.

10. Basis 3/4.2.5. Revised indicated and analytical limits in the basis. These changes correctly explain the ITDP uncertainty values found acceptable in Item 6 above, and are, therefore, also acceptable.

11. T.S. 6.9.1.9. Revised the requirements on the peaking factor limit report. This specification reflects the revised guidelines for the peaking factor limit report developed by the NRC staff and first implemented in the Vogtle Technical Specifications. The specification requires the peaking factor limit report to be submitted within 30 days following its implementation. The quantities reported are calculated by approved methods referenced in the specification. The staff concludes that this specification adequately reflects the requirements set forth for its use, and is, therefore, acceptable.

EVALUATION SUMMARY

As indicated in the sections above, the various design areas, use of V-5 fuel, accident analyses, spent fuel pool analyses, and proposed technical specification changes for Callaway Cycle 3 have been found acceptable. The staff, therefore, approves operation of the cycle as described, and concludes such operation will not create any increase in hazards to the health and safety of the public.

## 4.0 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 changes to the surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that 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.

Adoption of the V-5 fuel design will result in corresponding increased burnup levels and plant operation at higher core thermal power. The NRC staff prepared an environmental assessment of the higher burnup fuel (52 FR 37681) and, pursuant to 10 CFR 51.32, the Commission has determined that the issuance of the amendment will have no significant impact on the environment.

## 5.0 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.

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