April 8, 1986

Docket No.: 50-483

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

Dear Mr. Schnell:

Subject: Callaway Plant, Unit 1 - Amendment No. 15 to Facility Operating License NPF-30

The Commission has issued the enclosed Amendment No. 15 to Facility Operating License NPF-30 for the Callaway Plant, Unit 1. The amendment consists of a change to the Technical Specifications in response to your application dated November 15, 1985, as supplemented December 13, 1985, January 28, 1986, February 18, 1986, February 24, 1986, and February 28, 1986.

The amendment modifies the Technical Specifications to support a transition from a Westinghouse 17x17 low-parasitic fuel assembly fueled core to a Westinghouse 17x17 optimized fuel assembly fueled core.

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

Sincerely,

/S/

Paul W. O'Connor, Project Manager
PWR Project Directorate #4
Division of PWR Licensing-A, NRR

Enclosures:
1. Amendment No. 15 to License No. NPF-30
2. Safety Evaluation Report

cc w/enclosures:
See next page

DISTRIBUTION:
~~SEE ATTACHED~~ PAGE

* SEE PREVIOUS CONCURRENCES

PWR#4:DPWR-A	PWR#4:DPWR-A	OELD	PWR#4/DPWR-A
*MDuncan/mac	PWO *PO'Connor	*Goddard	DSH for *BJYoungblood
03/31/86	03/24/86	04/02/86	03/31/86

8604210466 860408
PDR ADOCK 05000483
P PDR

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 15, 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

/S/

B. J. Youngblood, Director
PWR Project Directorate #4
Division of PWR Licensing-A, NRR

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 8, 1986

PWOC	MD		PSH for
PWR#4:DPWR-A	PWR#4:DPWR-A	OELD	PWR#4/DPWR-A
PO'Connor/mac	MDuncan	RPerlis	BJYoungblood
03/24/86	03/31/86	04/2/86	03/31/86

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

April 8, 1986

Docket No.: 50-483

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

Dear Mr. Schnell:

Subject: Callaway Plant, Unit 1 - Amendment No.15 to Facility Operating License NPF-30

The Commission has issued the enclosed Amendment No.15 to Facility Operating License NPF-30 for the Callaway Plant, Unit 1. The amendment consists of a change to the Technical Specifications in response to your application dated November 15, 1985, as supplemented December 13, 1985, January 28, 1986, February 18, 1986, February 24, 1986, and February 28, 1986.

The amendment modifies the Technical Specifications to support a transition from a Westinghouse 17x17 low-parasitic fuel assembly fueled core to a Westinghouse 17x17 optimized fuel assembly fueled core.

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

Sincerely,

Paul W. O'Connor

Paul W. O'Connor, Project Manager
PWR Project Directorate #4
Division of PWR Licensing-A, NRR

Enclosures:
1. Amendment No. 15 to License No. NPF-30
2. Safety Evaluation Report

cc w/enclosures:
See next page

Mr. D. F. Schnell
Union Electric Company

Callaway Plant
Unit No. 1

cc:
Mr. Nicholas A. Petrick
Executive Director - SNUPPS
5 Choke Cherry Road
Rockville, Maryland 20850

Gerald Charnoff, Esq.
Thomas A. Baxter, Esq.
Shaw, Pittman, Potts & Trowbridge
1800 M Street, N. W.
Washington, D. C. 20036

Mr. J. E. Birk
Assistant to the General Counsel
Union Electric Company
Post Office Box 149
St. Louis, Missouri 63166

U. S. Nuclear Regulatory Commission
Resident Inspectors Office
RR#1
Steedman, Missouri 65077

Mr. Donald W. Capone, Manager
Nuclear Engineering
Union Electric Company
Post Office Box 149
St. Louis, Missouri 63166

Chris R. Rogers, P.E.
Manager - Electric Department
301 W. High
Post Office Box 360
Jefferson City, Missouri 65102

Regional Administrator
U. S. NRC, Region III
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Mr. Ronald A. Kucera, Deputy Director
Department of Natural Resources
P. O. Box 176
Jefferson City, Missouri 65102

Mr. Glenn L. Koester
Vice President - Nuclear
Kansas Gas and Electric Company
201 North Market Street
Post Office Box 208
Wichita, Kansas 67201

Lewis C. Green, Esq.
Green, Hennings & Henry
Attorney for Joint Intervenors
314 N. Broadway, Suite 1830
St. Louis, Missouri 65251

Ms. Marjorie Reilly
Energy Chairman of the League of
Women Voters of Univ. City, MO
7065 Pershing Avenue
University City, Missouri 63130

Mr. Donald Bollinger, Member
Missourians for Safe Energy
6267 Delmar Boulevard
University City, Missouri 63130

Mr. Dan I. Bolef, President
Kay Drey, Representative
Board of Directors Coalition
for the Environment
St. Louis Region
6267 Delmar Boulevard
University City, Missouri 63130

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

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT 1

DOCKET NO. 50-483

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 15
License No. NPF-30

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

   A. The application for amendment to the Callaway Plant, Unit 1 (the facility) Facility Operating License No. NPF-30 filed by Union Electric Company (the licensee) dated November 15, 1985, as supplemented December 13, 1985, January 28, 1986, February 18, 1986, February 24, 1986, and February 28, 1986, 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. NFP-30 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 15, 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

Darl Hood

for B. J. Youngblood, Director
PWR Project Directorate #4
Division of PWR Licensing-A, NRR

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 8, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 15

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.

Amended Page	Overleaf Page
2-10	
I	
II	
1-2	1-1
1-3	
1-4	
1-5	
1-6	
1-7	
1-8	
1-9	
2-2	2-1
2-4	2-3
2-7	
B2-1	
B2-2	
B2-3	B2-4
3/4 1-19	3/4 1-20
3/4 1-22	3/4 1-21
3/4 2-8	3/4 2-7
3/4 2-9	
3/4 2-10	3/4 10-1
3/4 2-11	
3/4 2-12	
3/4 2-13	
3/4 2-14	
3/4 10-2	
B3/4 2-1	
B3/4 2-2	
B3/4 2-3	
B3/4 2-4	
B3/4 2-5	
B3/4 4-1	B3/4 4-2
2-8	
2-9	

Delete page 3/4 2-15.
Delete page B3/4 2-6.

INDEX

DEFINITIONS

DEFINITIONS

SECTION

PAGE

DEFINITIONS (Continued)

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Technical Specification which prescribes remedial measures required under designated conditions.

ACTUATION LOGIC TEST

1.2 An ACTUATION LOGIC TEST shall be the application of various simulated input combinations in conjunction with each possible interlock logic state and verification of the required logic output. The ACTUATION LOGIC TEST shall include a continuity check, as a minimum, of output devices.

ANALOG CHANNEL OPERATIONAL TEST

1.3 An ANALOG CHANNEL OPERATIONAL TEST shall be the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY of alarm, interlock and/or trip functions. The ANALOG CHANNEL OPERATIONAL TEST shall include adjustments, as necessary, of the alarm, interlock and/or Trip Setpoints such that the Setpoints are within the required range and accuracy.

AXIAL FLUX DIFFERENCE

1.4 AXIAL FLUX DIFFERENCE shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector.

CHANNEL CALIBRATION

1.5 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel such that it responds within the required range and accuracy to known values of input. The CHANNEL CALIBRATION shall encompass the entire channel including the sensors and alarm, interlock and/or trip functions and may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.6 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:

1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or

2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.3.

b. All equipment hatches are closed and sealed,

c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,

d. The containment leakage rates are within the limits of Specification 3.6.1.2, and

e. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

CONTROLLED LEAKAGE

1.8 CONTROLLED LEAKAGE shall be that seal water flow from the reactor coolant pump seals.

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any component within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

DESIGN THERMAL POWER

1.10 DESIGN THERMAL POWER shall be a design total reactor core heat transfer rate to the reactor coolant of 3565 MWt.

DOSE EQUIVALENT I-131

1.11 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.12 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half-lives greater than 15 minutes, making up at least 95% of the total noniodine activity in the coolant.

ENGINEERED SAFETY FEATURES RESPONSE TIME

1.13 The ENGINEERED SAFETY FEATURES (ESF) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF Actuation Setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

FREQUENCY NOTATION

1.14 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of Leakage Detection Systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the Secondary Coolant System.

MASTER RELAY TEST

1.16 A MASTER RELAY TEST shall be the energization of each master relay and verification of OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.

MEMBER(S) OF THE PUBLIC

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

OFFSITE DOSE CALCULATION MANUAL

1.18 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program.

OPERABLE - OPERABILITY

1.19 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.20 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.2.

PHYSICS TESTS

1.21 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the core and related instrumentation: (1) described in Chapter 14.0 of the FSAR, or (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.22 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a nonisolable fault in a Reactor Coolant System component body, pipe wall, or vessel wall.

PROCESS CONTROL PROGRAM

1.23 The PROCESS CONTROL PROGRAM shall contain the current formula, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Part 20, 10 CFR Part 71 and Federal and State regulations, burial ground requirements, and other requirements governing the disposal of the radioactive waste.

PURGE - PURGING

1.24 PURGE or PURGING 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 required to purify the confinement.

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.

SHUTDOWN MARGIN

1.29 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

1.30 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

1.31 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

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

SOURCE CHECK

1.33 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

1.34 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

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

TRIP ACTUATING DEVICE OPERATIONAL TEST

1.36 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

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

UNRESTRICTED AREA

1.38 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

1.39 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

1.40 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

1.41 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.

TABLE 1.2

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	$\geq$ 0.99	> 5%	$\geq$ 350°F
2. STARTUP	$\geq$ 0.99	$\leq$ 5%	$\geq$ 350°F
3. HOT STANDBY	< 0.99	0	$\geq$ 350°F
4. HOT SHUTDOWN	< 0.99	0	350°F > T_{avg} > 200°F
5. COLD SHUTDOWN	< 0.99	0	$\leq$ 200°F
6. REFUELING**	$\leq$ 0.95	0	$\leq$ 140°F

*Excluding decay heat.
**Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

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.

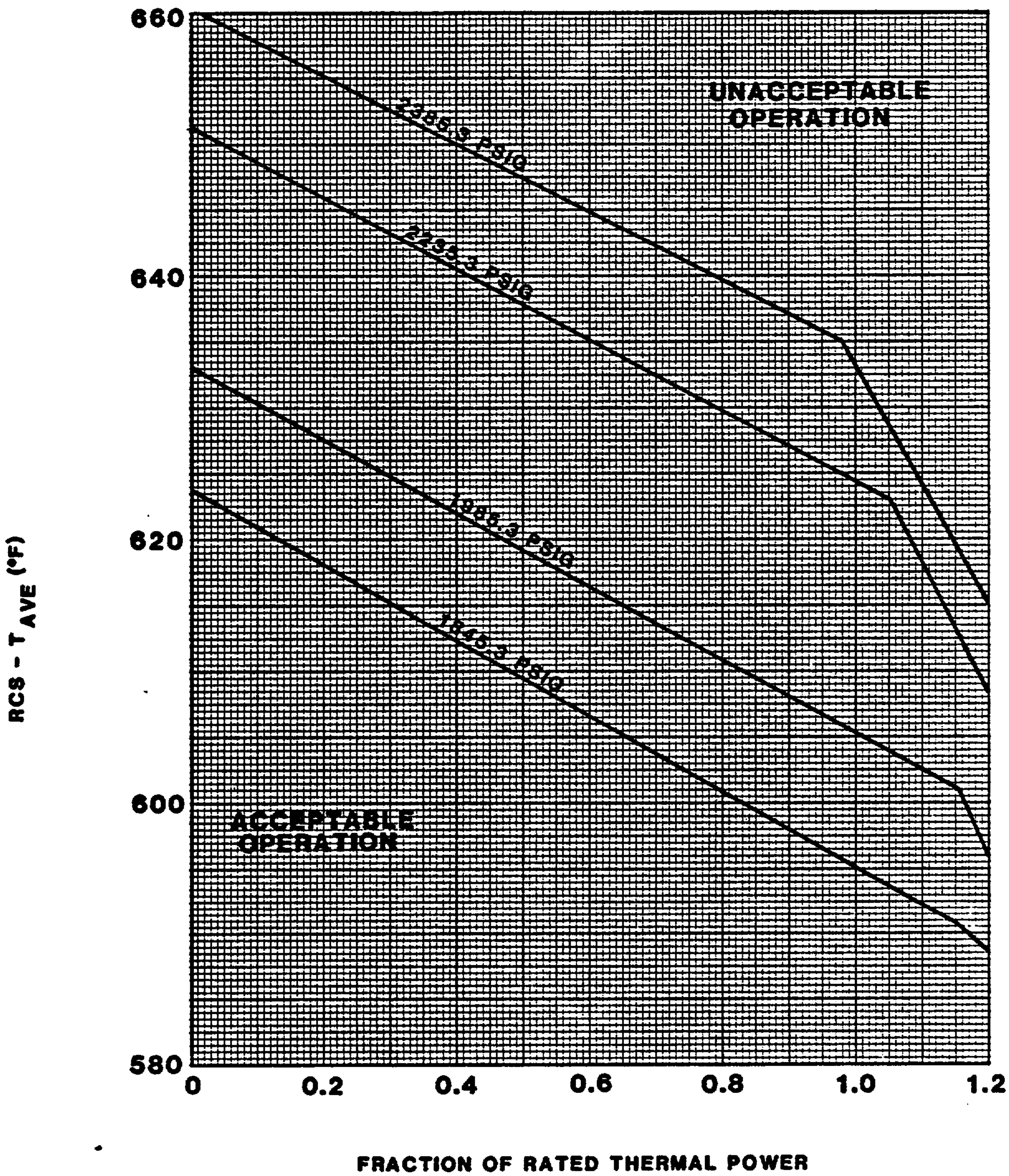

FIGURE 2.1-1

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.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	SENSOR ERROR Z	SENSOR ERROR (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	≤109% of RTP*	≤112.3% of RTP*
b. Low Setpoint	8.3	4.56	0	≤25% of RTP*	≤28.3% of RTP*
3. Power Range, Neutron Flux, High Positive Rate	2.4	0.5	0	≤4% of RTP* with a time constant ≥2 seconds	≤6.3% of RTP* with a time constant ≥2 seconds
4. Power Range, Neutron Flux, High Negative Rate	2.4	0.5	0	≤4% of RTP* with a time constant ≥2 seconds	≤6.3% of RTP* with a time constant ≥2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.41	0	≤25% of RTP*	≤35.3% of RTP*
6. Source Range, Neutron Flux	17.0	10.01	0	≤10^5 cps	≤1.6×10^5 cps
7. Overtemperature ΔT	9.3	6.77	2.06 +1.24***	See Note 1	See Note 2
8. Overpower ΔT	5.7	1.49	0.5	See Note 3	See Note 4
9. Pressurizer Pressure-Low	5.0	2.21	2.0	≥1885 psig	≥1874 psig
10. Pressurizer Pressure-High	7.5	4.96	1.0	≤2385 psig	≤2400 psig
11. Pressurizer Water Level-High	8.0	2.18	2.0	≤92% of instrument span	≤93.8% of instrument span
12. Reactor Coolant Flow-Low	2.5	1.7	0.6	≥90% of loop minimum measured flow**	≥89.2% 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 ΔT

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

Where:

ΔT	=	Measured ΔT by RTD Manifold Instrumentation;
$\frac{1+\tau_1 S}{1+\tau_2 S}$	=	Lead-lag compensator on measured ΔT;
τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, τ_1 = 8 s, τ_2 = 3 s;
$\frac{1}{1+\tau_3 S}$	=	Lag compensator on measured ΔT;
τ_3	=	Time constant utilized in the lag compensator for ΔT, τ_3 = 0 s;
ΔT_o	=	61.8°F (Referenced ΔT at DESIGN 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;
τ_4, τ_5	=	Time constants utilized in the lead-lag compensator for T_{avg}, τ_4 = 28 s, τ_5 = 4 s;
T	=	Average temperature, °F;
$\frac{1}{1+\tau_6 S}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, τ_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 ΔT 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 ΔT 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 2.0% of ΔT span.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \le \Delta T_0 \left\{K_4 - K_5 \left(\frac{\tau_7 S}{1 + \tau_7 S}\right) \left(\frac{1}{1 + \tau_6 S}\right) T - K_6 \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T''\right] - f_2(\Delta I)\right\}$$

Where:

ΔT	=	Measured ΔT by RTD Manifold Instrumentation;
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	Lead-lag compensator on measured ΔT;
τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, τ_1, = 8 s., τ_2 = 3 s;
$\frac{1}{1 + \tau_3 S}$	=	Lag compensator on measured ΔT;
τ_3	=	Time constant utilized in the lag compensator for ΔT, τ_3 = 0 s;
ΔT_0	=	61.8°F (Referenced ΔT at DESIGN 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;
τ_7	=	Time constant utilized in the rate-lag compensator for T_{avg}, τ_7 = 10 s;
$\frac{1}{1 + \tau_6 S}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, τ_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 $T \leq T''$;
T	=	Average Temperature, °F;
T''	=	Indicated T_{avg} at DESIGN THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 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 4.1% of ΔT span.

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 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 the WRB-1 Correlation).

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. In addition, margin has been maintained in the design by meeting safety analysis DNBR limits of 1.42 for thimble and 1.45 for typical cells in performing safety analyses.

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 (ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the setpoints to provide protection consistent with core safety limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

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

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

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

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the nominal values at which the Reactor trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the core and Reactor Coolant System are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. The Setpoint for a Reactor Trip System or interlock function is considered to be adjusted consistent with the nominal value when the "as measured" Setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which Setpoints can be measured and calibrated, Allowable Values for the Reactor Trip Setpoints have been specified in Table 2.2-1. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its Trip Setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the uncertainties in calibrating the instrumentation. In Equation 2.2-1, $Z + R + S \leq TA$, the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 2.2-1, in percent span, is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, between the Trip Setpoint and the value used in the analysis for Reactor trip. R or Rack Error is the "as measured" deviation, in percent span, for the affected channel from the specified Trip Setpoint. S or Sensor Error is either the "as measured" deviation of the sensor from its calibration point or the value specified in Table 2.2-1, in percent span, from the analysis assumptions. For functions which have multiple input values, due to more than one parameter providing input to the function, multiple values for S are noted which are applicable to the primary input channels. (See Westinghouse statistical setpoint study for protection systems provided for justification). Use of Equation 2.2-1 allows for a sensor drift factor, an increased rack drift factor, and provides a threshold value for REPORTABLE EVENTS.

The methodology to derive the Trip Setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensors and other instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

The various Reactor trip circuits automatically open the Reactor trip breakers whenever a condition monitored by the Reactor Trip System reaches a preset or calculated level. In addition to redundant channels and trains, the design approach provides a Reactor Trip System which monitors numerous system variables, therefore providing Trip System functional diversity. The functional capability at the specified trip setting is required for those anticipatory or diverse Reactor trips for which no direct credit was assumed in the safety analysis to enhance the overall reliability of the Reactor Trip System. The Reactor Trip System initiates a Turbine trip signal whenever Reactor trip is initiated. This prevents the reactivity insertion that would otherwise result from excessive Reactor Coolant System cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

Manual Reactor Trip

The Reactor Trip System includes manual Reactor trip capability.

Power Range, Neutron Flux

In each of the Power Range Neutron Flux channels there are two independent bistables, each with its own trip setting used for a High and Low Range trip setting. The Low Setpoint trip provides protection during subcritical and low power operations to mitigate the consequences of a power excursion beginning from low power, and the High Setpoint trip provides protection during power operations to mitigate the consequences of a reactivity excursion from all power levels.

The Low Setpoint trip may be manually blocked above P-10 (a power level of approximately 10% of RATED THERMAL POWER) and is automatically reinstated below the P-10 Setpoint.

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of a rupture of a control rod drive housing. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from mid-power.

The Power Range Negative Rate trip provides protection for control rod drop accidents. At high power a single or multiple rod drop accident could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate trip for those control rod drop accidents for which DNBRs will be greater than the limit value.

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 2.4 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.

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as shown in Figure 3.1-1.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With the control banks inserted beyond the above insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2:

a. Restore the control banks to within the limits within 2 hours, or

b. Reduce THERMAL POWER within 2 hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using the above figure, or

c. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours except during time intervals when the Rod Insertion Limit Monitor is inoperable, then verify the individual rod positions at least once per 4 hours.

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

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

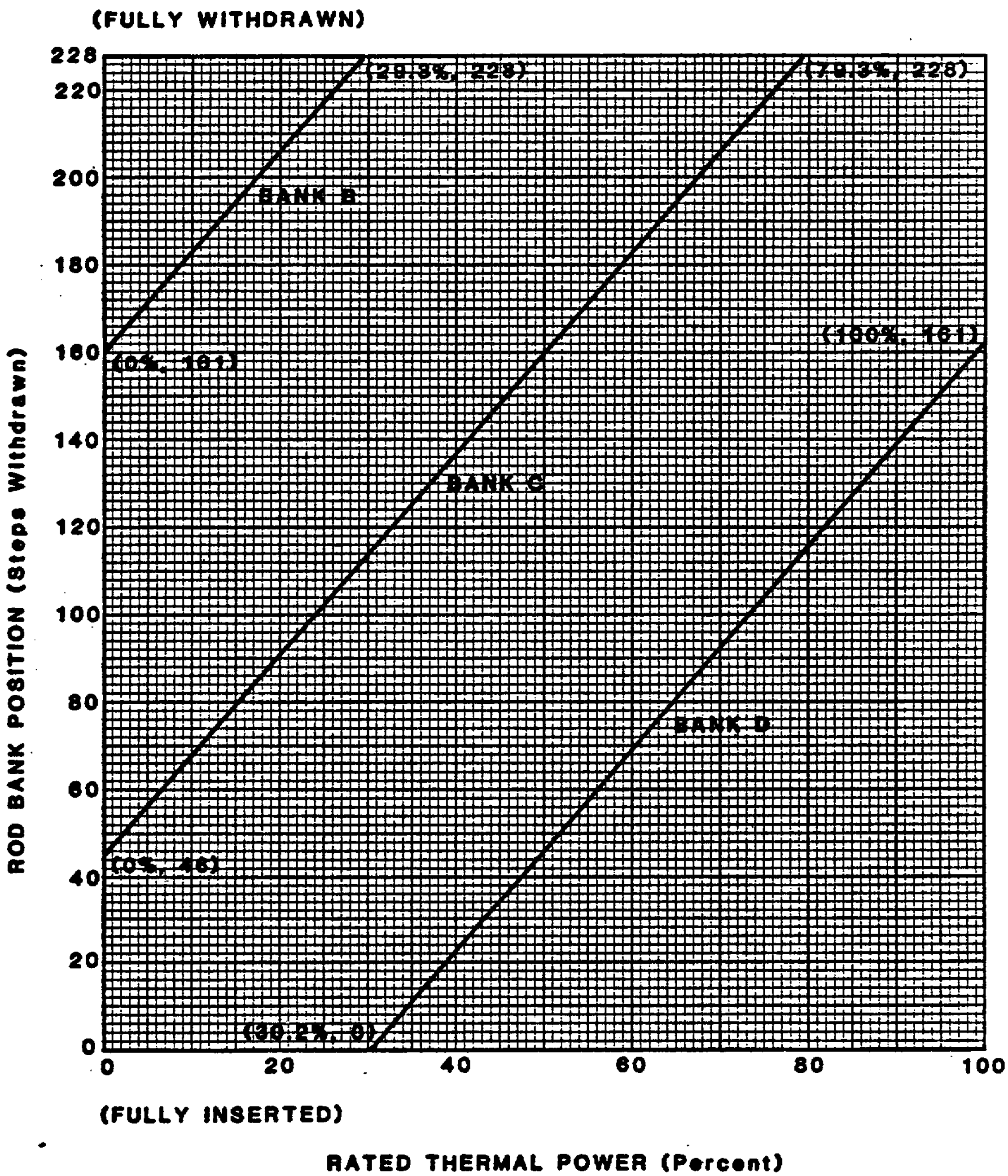

FIGURE 3.1-1

ROD BANK INSERTION LIMITS VERSUS
RATED THERMAL POWER - FOUR LOOP OPERATION

2) When the F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} at least once per 31 EFPD.

e. The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) shall be provided for all core planes containing Bank "D" control rods and all unrodded core planes in a Radial Peaking Factor Limit Report per Specification 6.9.1.9;

f. The F_{xy} limits of Specification 4.2.2.2e., above, are not applicable in the following core planes regions as measured in percent of core height from the bottom of the fuel:

1) Lower core region from 0 to 15%, inclusive,

2) Upper core region from 85 to 100%, inclusive,

3) Grid plane regions at 17.8 ± 2%, 32.1 ± 2%, 46.4 ± 2%, 60.6 ± 2% and 74.9 ± 2%, inclusive, and

4) Core plane regions within ± 2% of core height (± 2.88 inches) about the bank demand position of the Bank "D" control rods.

g. With F_{xy}^{C} exceeding F_{xy}^{L}, the effects of F_{xy} on $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limits.

4.2.2.3 When $F_Q(Z)$ is measured for other than F_{xy} determinations, 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.

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

SURVEILLANCE REQUIREMENTS

4.2.3.1 $F^N_{\Delta H}$ shall be determined to be within its limit by using the movable incore detectors to obtain a power distribution map:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

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

c. The provisions of Specification 4.0.4 are not applicable.

3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02.

APPLICABILITY: MODE 1, above 50% of RATED THERMAL POWER.*

ACTION:

a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but less than or equal to 1.09:

   1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

      a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

      b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

   2. Within 2 hours either:

      a) Reduce the QUADRANT POWER TILT RATIO to within its limit, or

      b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

   3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours, and

   4. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

*See Special Test Exception Specification 3.10.2.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

b. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to misalignment of either a shutdown or control rod:

1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

   a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

   b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

2. Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1, within 30 minutes;

3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

4. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

c. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to causes other than the misalignment of either a shutdown or control rod:

1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

   a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

   b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

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

3. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified at 95% or greater RATED THERMAL POWER.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:

a. Calculating the ratio at least once per 7 days when the alarm is OPERABLE, and

b. Calculating the ratio at least once per 12 hours during steady-state operation when the alarm is inoperable.

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75% of RATED THERMAL POWER with one Power Range channel inoperable by using the movable incore detectors to confirm that the normalized symmetric power distribution, obtained from two sets of four symmetric thimble locations or a full core flux map, is consistent with the indicated QUADRANT POWER TILT RATIO at least once per 12 hours.

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Reactor Coolant System T_{avg},

b. Pressurizer Pressure, and

c. Reactor Coolant System Total Flow Rate.

APPLICABILITY: MODE 1.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 The calculated RCS total flow rate shall be determined to be greater than or equal to 382,630* GPM.

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

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

4.2.5.3 The RCS loop flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

4.2.5.4 The RCS total flow rate shall be determined by precision heat balance measurements at least once per 18 months. Within 7 days of performing the precision heat balance, the instrumentation used for determination of steam pressure, feedwater temperature, and feedwater venturi ΔP in the calorimetric calculations shall be calibrated.

4.2.5.5 The feedwater venturi shall be inspected for fouling and cleaned as necessary at least once per 18 months.

---

*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 surveillance.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS Four Loops in Operation
Indicated Reactor Coolant System T_{avg}	$\leq$ 593.4°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.

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and SHUTDOWN MARGIN provided reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any full-length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full-length control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full-length control rod either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each full-length control rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion, and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specifications 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1 and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be less than or equal to 85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.2.2 The requirements of the below listed specifications shall be performed at least once per 12 hours during PHYSICS TESTS:

a. Specifications 4.2.2.2 and 4.2.2.3, and

b. Specification 4.2.3.1.

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 definitions 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;

$F^N_{\Delta H}$ 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; and

$F_{xy}(Z)$ Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.

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.

AXIAL FLUX DIFFERENCE (Continued)

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.

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 ± 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.

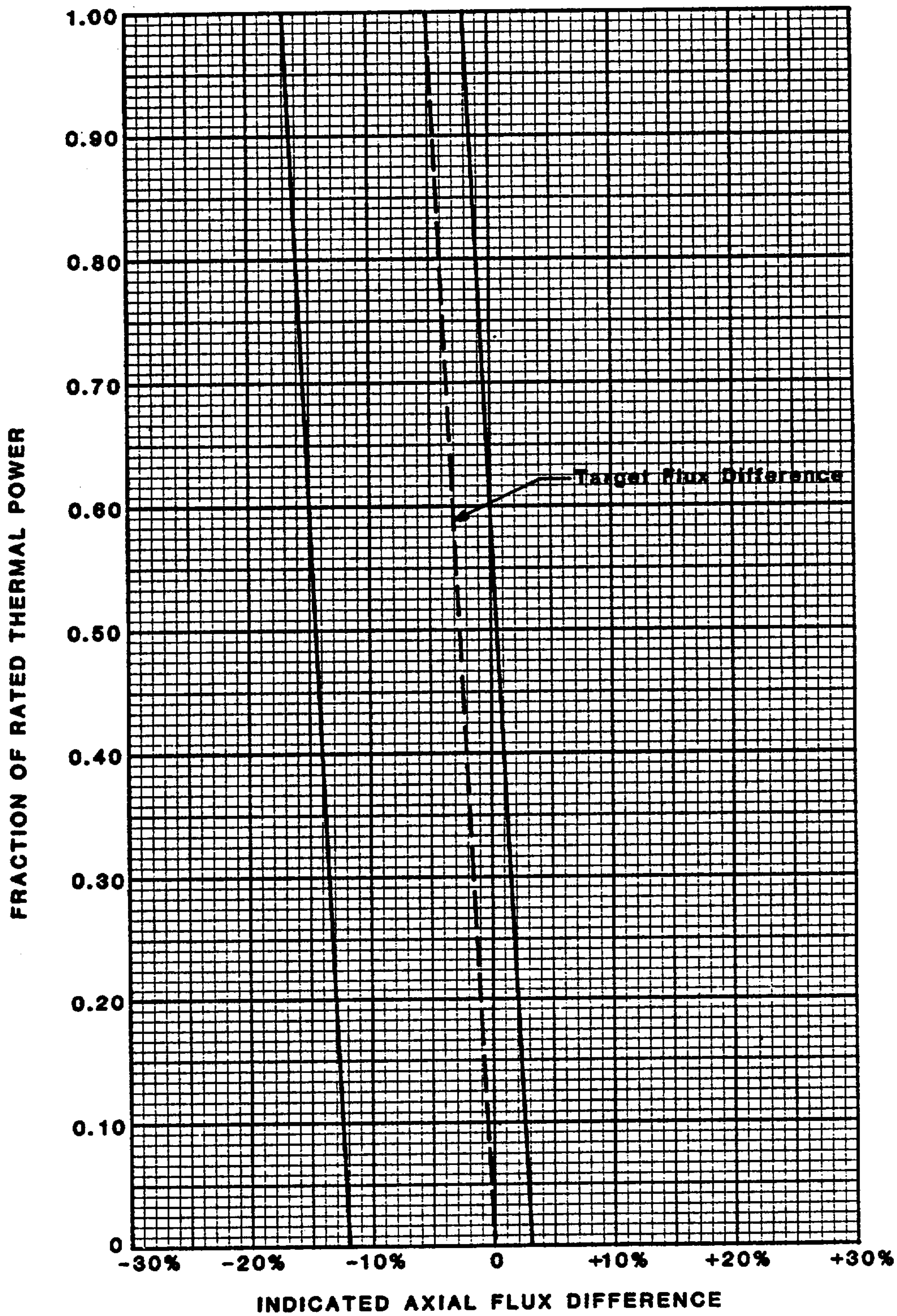

FIGURE B 3/4.2-1

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)

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 limit DNBRs (1.42 and 1.45 for thimble and typical cells, respectively) and the design limit DNBRs (1.32 and 1.34 for thimble and typical cells, respectively) is maintained. A fraction of this margin is utilized to accommodate the transition core DNBR penalty (2%) and the appropriate fuel rod bow DNBR penalty (less than 3% per WCAP-8691, Rev. 1). The 7% margin between design and safety analysis DNBR limits includes >2% margin for plant design flexibility.

The Radial Peaking Factor, F_{xy} (Z), is measured periodically to provide assurance that the Hot Channel Factor, $F_Q(Z)$, remains within its limit. The F_{xy} limit for RATED THERMAL POWER (F^{RTP}_{xy}) as provided in the Radial Peaking Factor Limit Report per Specification 6.9.1.9 was determined from expected power control maneuvers over the full range of burnup conditions in the core.

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.

QUADRANT POWER TILT RATIO (Continued)

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 593.4°F and the indicated pressurizer pressure value of 2220 psig correspond to analytical limits of 595.9°F and 2205 psig respectively, with allowance for measurement uncertainty.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

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

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above the safety analysis DNBR limits during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within 6 hours.

In MODE 3, two reactor coolant loops provide sufficient heat removal capability for removing decay heat even in the event of a bank withdrawal accident; however, single failure considerations require that three loops be OPERABLE. A single reactor coolant loop provides sufficient heat removal if a bank withdrawal accident can be prevented, i.e., by opening the Reactor Trip System breakers.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either RHR or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two RHR loops be OPERABLE.

The operation of one reactor coolant pump (RCP) or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a reactor coolant pump in MODES 4 and 5 are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer Code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperatures.

During operation, all pressurizer Code safety valves must be OPERABLE to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss-of-load assuming no Reactor trip and also assuming no operation of the power-operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.3 PRESSURIZER

The 12-hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation.

3/4.4.4 RELIEF VALVES

The power-operated relief valves (PORVs) and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs minimizes the undesirable opening of the spring-loaded pressurizer Code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 15 TO OPERATING LICENSE NO. NPF-30

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT 1

DOCKET NO. 50-483

1.0 INTRODUCTION

By letter dated November 15, 1985, 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 2. In support of the application the licensee provided a report entitled, "Safety Evaluation for the Callaway Plant Transition to Westinghouse 17x17 Optimized Fuel Assemblies". 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 EVALUATION

As part of the core reload for Cycle 2 the licensee has elected to initiate a transition from standard Westinghouse LOPAR fuel to Optimized Fuel Assembly (OFA) fuel. In addition the analyses are being performed under the assumption of a core power level of 3565 thermal megawatts (MWt) in preparation for a future power up-rating. The plant will continue to operate at 3411 MWT during Cycle 2. Also Wet Annular Burnable Absorber (WABA) fuel is being introduced in Cycle 2.

Analyses have been performed for cores having partial OFA loadings (the Cycle 2 core will consist of approximately 40% OFA fuel) and for a core consisting entirely of that fuel. The operating limits and protection system settings have been based on the most limiting of the core loadings.

2. FUEL EVALUATION

The use of 17x17 OFA fuel has been approved in other Westinghouse reactors (e.g., McGuire) and its use in Callaway is acceptable. This fuel has been designed to be compatible with Westinghouse standard (LOPAR) fuel in order to facilitate the transition from one fuel to the other. The mechanical behavior of the two fuels has been examined for the Callaway plant and it is concluded that all applicable criteria are met. We conclude that the fuel mechanical evaluation is acceptable.

Thermal evaluation of the fuel was performed with the PAD code. PAD is now the standard Westinghouse code for this purpose and its use by Callaway is acceptable.

3. NUCLEAR EVALUATION

The transition from LOPAR to OFA fuel has a minimal effect on the neutronic parameters of the core. No changes will be required in the current nuclear design bases. The analyses of the transition cores and of the all-OFA core were performed with the Reload Safety Evaluation Methodology which has been used and approved in other reactors. We conclude that the nuclear evaluation for the transition to OFA fuel is acceptable.

As part of the transition (but not required by it) the multiplier in the algorithm for obtaining the permitted value of $F^N_{\Delta H}$ as a function of power has been changed from 0.2 to 0.3. This has the effect of permitting higher values of $F^N_{\Delta H}$ at low power levels than was previously the case. Account has been taken of this change in the safety analysis. The change has been previously approved for other plants and we find it acceptable for Callaway.

4. THERMAL-HYDRAULIC EVALUATION

The Callaway plant has been operating with a 17x17 low-parasitic (LOPAR) fueled core. It is planned to eventually operate with a full core of 17x17 Optimized Fuel Assembly (OFA) fuel. The presence of transitional mixed cores containing both the standard LOPAR and OFA fuel requires that particular attention be paid to the thermal-hydraulic analysis of the core. Cycle 2 is the first Callaway cycle utilizing OFA fuel and will contain 109 LOPAR fuel assemblies and eighty-four 17x17 OFAs (approximately 43% OFA fuel). A number of the OFAs will employ the Wet Angular Burnable Absorber (WABA) rods. The core safety analyses have been performed at a core Design Thermal Power of 3565 MWt and a slightly reduced reactor coolant flow to account for up to 10% steam generator tube plugging. However, the Callaway Cycle 2 core will be operated at the currently licensed Rated Thermal Power of 3411 MWt.

The licensee has presented a safety evaluation for the Callaway plant for transition to Westinghouse 17x17 OFA fuel in Attachment B of Reference 1. In response to questions, the licensee supplied information (Ref. 2) on the thermal-hydraulic design comparison. This includes Table 1, which presents values for cores using both 17x17 LOPAR fuel and 17x17 OFA fuel and presents Cycle 2 operating parameters and design parameters. From Table 1 it is seen that the following values are constant for LOPAR and OFA fuel: reactor heat input, core pressure, total flow rate, nominal inlet temperature, average temperature rise, average linear power (kW/ft) and peak linear power (kW/ft). The active heat transfer surface area for the OFA fuel is smaller than for the LOPAR fuel (fuel rods have smaller O.D). Also, the average velocity along the fuel rods is less for OFA fuel. However, the average heat flux for the OFA fuel is larger than for the LOPAR fuel. The core pressure drop for a OFA fueled core and a LOPAR fueled core are 26.4 ± 2 psia and 26.5 ± 2.6 psia respectively and are therefore approximately the same.

The OFA and LOPAR fuel assemblies have been tested for hydraulic characteristics (Ref. 3) and they have been shown to be hydraulically compatible. Since the core pressure drops in an all LOPAR core and an all OFA core are approximately the same, the core flow remains the same also. The actual measured flow rate for the Callaway plant in the last cycle (Cycle 1) was approximately 411,000 gpm (Ref. 11), which is well over the Technical Specification minimum measured flow of 382,630 gpm as shown in Table 1. The OFA fuel assemblies should resist liftoff as the current holddown spring design remains the same as for the LOPAR fuel and the pressure drop and reactor system flow also remain approximately the same as before.

The thermal-hydraulic analysis of this mixed core was performed using the approved "Improved Thermal Design Procedure" (ITDP) (Ref. 4) together with the WRB-1 DNB correlation (Ref. 5). Use of this correlation for OFA fuel has been demonstrated and documented in WCAP-9401-A for 17x17 fuel which has been approved.

In the Improved Thermal Design Procedure the safety analyses are performed using nominal values of the plant operating, nuclear, thermal, and fuel fabrication parameters. Uncertainties in the DNBR value due to variations in these parameters are combined statistically and added to the correlation DNBR limit (1.17) to obtain the design DNBR limit. The values obtained for this quantity for Callaway are 1.32 for thimble cells (three fuel rods and a thimble tube) and 1.34 for typical cells (four fuel rods). The licensee has provided information concerning the plant specific uncertainties for Callaway which support these values. Transition core and rod bow effects are not included in the design DNBR limit. In order to account for these effects additional margin is provided to arrive at analysis values which are 1.42 and 1.45 for thimble and typical cells, respectively.

In response to a question, the licensee supplied information (Ref. 6) which provided responses to the eleven items listed in the NRC cover letter for Safety Evaluation of WCAP-9500-A (Ref. 7) for plants using ITDP. This included information on plant specific margins used to offset reduction in DNBR due to rod bowing and for the transition core penalty. The OFA fuel assemblies have sufficient margin (approximately 7%) between the safety analysis minimum DNBR and the design limit DNBR, as shown below, to accommodate the rod bow penalty and transition core penalty.

	17x17 OFA	
	THIMBLE	TYPICAL
Correlation	WRB-1	WRB-1
Correlation Limit	1.17	1.17
Design Limit	1.32	1.34
Safety Analysis Minimum DNBR	1.42	1.45

Because of the rod bow phenomena as described in Reference 8, rod bow DNBR penalities for full-flow and low-flow are required. These have been identified as being less than 3% using the information in Reference 9 which has been approved. This penalty is accommodated by the 7% margin available between the safety analysis minimum DNBR and the design DNBR limit.

The approved method of calculating the transition core DNB is given in Reference 7 from which a 2% DNBR transition core penalty is applied to the Callaway plant. Using this penalty, the transition core is analyzed as if it were a full core of OFAs. The 7% margin available between the safety analysis minimum DNBR and the design DNBR limit accommodates the 2% transition core DNBR penalty as well as the 3% rod bow DNBR penalty.

For the Callaway Cycle 2 core, WABA rods will be used instead of the glass absorbers of the Cycle 1 core. Since the WABAs provide an additional by-pass flow path (in the annulus of the absorber) they will slightly increase the total thimble tube by-pass flow. However, the number of WABA rods is well within the limit of acceptability as specified in Reference 10 which has been approved.

An RCS flow measurement uncertainty analysis, which is needed for the ITDP, was presented in Reference 6. This included a description of a generic calculational method (Appendix A of Reference 6) and a plant specific calculation (Appendix B of Reference 6). The plant specific calculation for Callaway supports a value of RCS flow measurement uncertainty of ±2.2%, which is the value used in the Callaway Technical Specifications. This value includes 0.1% to account for feedwater venturi fouling. The 2.2% value is based on using normalized elbow tap instrumentation readings after flow calorimetric measurements. We find the flow measurement analyses for the 2.2% flow measurement uncertainty to be acceptable.

5. TRANSIENTS AND ACCIDENTS

Each of the transients and accidents which were evaluated in the FSAR have been examined to determine whether a reanalysis is required to account for the effects of the transition from LOPAR to OFA fuel. The effects of the change in the $F^N_{\Delta H}$ multiplier and of the increase in design thermal power are also treated.

The change to OFA fuel affects the thermal-hydraulic performance of the fuel (see Section 4) in a negative way. In order to regain calculated margin the WRB-1 DNB correlation and the Improved Thermal Design procedure are used. Another effect of the use of OFA fuel is the increase in control rod scram time due to slightly reduced diameter of the guide tubes. This effect is accounted for in the analysis.

The increase in the $F^N_{\Delta H}$ multiplier is accounted for in establishing the core safety limits. The analyses are performed at the Design Thermal Power of 3565 MWt instead of the Current Rated Thermal Power of 3411 MWt. Each of the accidents reanalyzed is discussed below.

5.1 Increase in Heat Removal by the Secondary System
Events in this Category include:

1. Feedwater system malfunctions that result in a decrease in feedwater temperature.
2. Feedwater system malfunctions that cause an increase in feedwater flow.
3. Excessive increase in secondary steam flow.
4. Inadvertent opening of a steam generator relief or safety valve.
5. Steam supply piping failure.

The first four of these events are classified as Condition II events (anticipated transients) while the fifth is classified as a Condition IV event (design basis accident). Of the first three events the third (a 10% step increase in steam demand) is the limiting event. For that event analyses were performed with approved methods and procedures for both manual and automatic control at both minimum and maximum reactivity feedback. In no case was the DNBR safety limit violated. We conclude that the analysis of these events is acceptable.

For the fourth event (opening of a steam relief, safety or dump valve) a conservative calculation is performed with a bounding value of steam flow and hot standby conditions at end of cycle. The analysis shows that DNB does not occur for this event. Since the analysis was performed with acceptable methods and procedures and conservative input conditions were assumed we conclude that the analysis of this event is acceptable.

The fifth of these events is the steam line break accident. For this event the W-3 DNBR correlation was used for the W OFA fuel rather than the WRB-1 correlation as the minimum pressure falls below the range of the WRB-1 correlation ($1440 \leq P \leq 2490$ psia). The minimum pressure also falls below the pressure range given in most references (1000 psia) for the W-3 correlation. However, the licensee justified the use of the W-3 correlation for lower pressure based on data (Ref. 2) that showed no abnormality exists for pressure (the pressure does not show trends in predicted and measured DNB heat fluxes as a function of pressure), which reinforces its acceptablity. As part of the generic review for Westinghouse plants these data have been used to arrive at a new DNBR value slightly larger than 1.3 for the W-3 correlation for lower pressure. Also, the results of the analysis performed by the licensee (Ref. 11) show that the minimum DNBR value (over 1.8) during the SLB accident is well above the limit of 1.3. On the basis of the data presented and the substantial DNBR margin available, we find the W-3 correlation acceptable for the SLB analysis presented for Callaway.

Rupture of a steam pipe is assumed to include any accident which involves inadvertent steam release from a steam generator. Under no load conditions, a negative temperature coefficient, and the most reactive rod stuck out of the core, the cooldown would result in reduction of the shutdown margin.

Return to power would be a potential problem to the extent that there is a large increase in the hot channel factor when the highest reactivity rod is fully withdrawn. A number of protection systems will be activated in case of steam pipe rupture such as: safety injection, overpower trips, isolation of the feedwater lines and trip of the steam line isolation valves. The transient analysis is accomplished using the LOFTRAN code to compute the reactor and coolant system status and the THINC IV code to compute whether the DNB ratio falls below the minimum value. Analyses were performed using a .013 reactivity shutdown margin, a negative temperature coefficient corresponding to the EOC with all but the most reactive rod inserted, assumption of a single failure in the ECCS, power peaking factors corresponding to one rod stuck out, and different sizes of the steam line break. The results indicate that following a steamline break the DNBR will remain higher than the design DNBR limit. Therefore, the assumed reactivity shutdown margin is adequate and the results are acceptable.

5.2 Decrease in Heat Removal by Secondary System Events in this category include the following:

1. Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow
2. Loss of External Electrical Load
3. Turbine Trip
4. Inadvertent Closure of Main Steam Isolation Valves
5. Loss of Condenser Vacuum and Other Events Resulting in Turbine Trip
6. Loss of non-emergency AC Power to the Station Auxiliaries
7. Loss of Normal Feedwater Flow
8. Feedwater System Pipe Break

The above items are considered to be ANS Condition II events, with the exception of a Feedwater System Pipe Break, which is considered to be an ANS Condition IV event.

The first event is not applicable to PWRs. The Loss of External Electrical Load event is less limiting than the Turbine Trip event. Events 4 and 5 are also bounded by the Turbine Trip event.

5.2.1 Turbine Trip

The Turbine Trip event is more severe than the loss of load event because of a more rapid loss of steam flow due to the more rapid closure of the turbine stop valve than is the case for the turbine control valve. The analysis is performed with the approved LOFTRAN code and the following assumptions are made:

1. Both minimum and maximum reactivity feedback calculations are performed.
2. Cases taking credit for pressurizer spray and power operated relief valves to reduce coolant pressure are analyzed as well as those for which such credit is not taken. Safety valves are operable.

3. Credit is taken only for the safety valves in limiting secondary pressure.
4. No credit is taken for auxiliary feedwater flow during the event.
5. No credit is taken for the direct reactor trip on turbine trip.

The results of the analyses show that in each case the DNBR remains well above the design DNBR limit and the coolant system and steam generators are protected against over pressure by their respective safety valves. We find the analysis of the Turbine Trip event to be acceptable.

5.2 Loss of Non-Emergency AC Power to Plant Auxiliaries

Loss of non-emergency power may result in loss of power to plant auxiliaries, i.e., reactor coolant pumps, condensate pumps, etc. The transient is more severe than the turbine trip event because the decrease in heat removal by the secondary system is accompanied by a coolant flow coastdown which further reduces the capacity of the primary coolant to remove heat from the core.

The approved LOFTRAN code is used to perform the analysis. Conservative input assumptions are used including operation at 102 percent power, low value of average coolant temperature, conservative residual heat, no credit for reactor trip on loss of power, operation of pressurizer spray and power operated relief valves and secondary steam relief through the safety valves.

The results of the analysis show that DNBR remains above the design DNBR limit, that auxiliary feedwater capacity is sufficient to prevent water relief through the pressurizer relief and safety valves, and that natural circulation flow is sufficient to remove the residual heat from the fuel. We conclude that this analysis is acceptable.

5.2.3 Loss of Normal Feedwater Flow

A loss of normal feedwater may occur due to a pump failure, valve failures or loss of offsite AC power. The limiting event is that of total loss of normal feedwater. An analysis of this event is performed to show that fuel thermal design limits are met and that the auxiliary feedwater system is capable of removing the stored and residual heat and thus of returning the plant to a safe condition.

The approved LOFTRAN code is used as well as conservative input assumptions including prior operation at 102 percent design power, conservative decay heat, late reactor trip and initiation of auxiliary feedwater flow, worst single failure in auxiliary feedwater system, operability of pressurizer sprays and PORVs, and failure of the steam generator PORVs and relief valves.

Results of the analyses show that DNB is not approached during the transient and that the auxiliary feedwater system is capable of removing the stored and decay heat from the fuel. We conclude that the evaluation of the loss of normal feedwater event is acceptable.

5.2.4 Feedwater System Pipe Break

The Feedwater System Pipe Break is treated as a design basis (Condition IV) event. Analyses are performed to demonstrate that the reactor coolant pressure will remain below 110% of design value, that the core remains coolable and that resultant doses remain below acceptable limits.

The analyses were done with the approved LOFTRAN code for cases both with and without loss of offsite power. Conservative assumptions are made with respect to plant operating power, decay heat, initial values of reactor coolant temperature and pressure, pressurizer water level, operation of the protection system and ECCS equipment, and break size and location. The results of the analysis show that the core remains covered, that the hot leg temperature does not reach saturation, and that the Auxiliary Feedwater System provides sufficient cooling to remove decay heat. The radioactivity doses are bounded by those of the steamline break. We conclude that the analysis for this event is acceptable.

5.3 Decrease in Reactor Coolant System Flow Rate

Events in this category include the following:

1. Partial Loss of Forced Reactor Coolant Flow
2. Complete Loss of Forced Reactor Coolant Flow
3. Reactor Coolant Pump Shaft Seizure (Locked Rotor)
4. Reactor Coolant Pump Shaft Break

The first of these events is an anticipated transient (Condition II), the second is an unanticipated occurrence (Condition III), and the final two are design basis (Condition IV) events.

5.3.1 Partial Loss of Forced Reactor Coolant

The loss of two pumps with four loops in operation is analyzed for this event. Three codes which have been previously used by the licensee and accepted by the staff are used in the analysis - LOFTRAN, FACTRAN and THINC. LOFTRAN is used to obtain power and flow conditions during the transient, FACTRAN is used to obtain the heat flux as function of time and THINC is used to obtain DNBR as a function of time. The Improved Thermal Design Procedure is used and conservative reactivity coefficients are supplied as input.

The results of the analysis show that DNBR does not decrease below the design DNBR limit at any time during the transient. The applicable criterion for this event is thus met and we conclude that the analysis is acceptable.

5.3.2 Complete Loss of Forced Reactor Coolant Flow

This event is analyzed in the same manner as that described in Section 5.3.1 above except that loss of all pumps is assumed and trip occurs on loss of pump power instead of low core flow.

The results show that DNBR does not fall below the safety analysis value. Thus the criterion for a Condition II event is met which is acceptable for this event.

5.3.3 Locked Rotor

The coolant pump shaft seizure (Locked Rotor) is treated as a design basis event (Condition IV). Analyses are performed to show that the core remains in a coolable condition and that appropriate limits on offsite radiation doses are met. The analysis is performed with two codes - LOFTRAN, with which the power and flow transients are calculated and FACTRAN, with which the thermal behavior of the fuel is calculated.

Conservative assumptions made in the calculations include operation at 102 percent of Thermal Design Power, maximum coolant pressure and temperature, failure of pressurizer spray and power operated relief valves, and onset of DNB at initiation of the event. The effect of the zirconium-steam reaction is also included.

The results of the analysis show a maximum pressure of less than 110 percent of the design value, a maximum clad temperature at the hot spot of less than 2000 degrees Fahrenheit and zirconium water reaction at the core hot spot of 0.3 weight percent. We thus conclude that the core will remain in a coolable condition following the event. The offsite radiation dose is discussed in Section 5.7.3 below and assumes that cladding failure occurs for all fuel rods with DNB less than the safety limit.

5.4 Reactivity and Power Distribution Anomalies

This category includes the following events:

1. Uncontrolled Rod Cluster Assembly Bank Withdrawal from a Subcritical or Low Power Startup Condition
2. Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power
3. Rod Cluster Control Assembly Misoperation
4. Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature
5. A Malfunction or Failure of the Flow Controller in a BWR Recirculation Loop that Results in an Increased Reactor Coolant Flow Rate (not applicable to Callaway).
6. Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant.
7. Spectrum of Rod Cluster Control Assembly Ejection Accidents.

Of these the seventh is a design basis (Condition IV) event, and the third contains both Condition II and Condition III events. The rest are Condition II events.

5.4.1 Uncontrolled Rod Bank Withdrawal

Uncontrolled Rod Withdrawal events are analyzed for both startup conditions and operation at power. For these events the amount of reactivity which may be inserted and the rate of insertion is limited by the permitted rod insertions as a function of power level.

The startup event is analyzed by the TWINKLE, FACTRAN and THINC codes. TWINKLE, a spatial neutron kinetics code, is used to obtain the core power as a function of time, FACTRAN provides the fuel rod temperature transient and THINC is used for the transient DNBR calculation. Conservative input assumptions, including maximum reactivity insertion rate, minimum reactivity feedback and bounding values of axial and radial power shapes were used. The results of the calculations show that the safety analysis value of DNBR is not violated. We conclude that the analysis of the rod bank withdrawal event at startup conditions is acceptable.

The analysis of the event at power operating condition was performed with the LOFTRAN code. The Improved Thermal Design Procedure was used. Analyses were done as a function of both power level and reactivity insertion rate. Protection is provided by the combination of the high neutron flux trip and the overtemperature - delta T trip. Conservative values of trip setpoints are assumed and both maximum and minimum reactivity feedback cases are analyzed. The results of the calculations show that in no case does the DNB fall below the design DNBR limit. We conclude that the analyses of the rod bank withdrawal events are acceptable.

5.4.2 Rod Misoperation Events

These events include misalignment of a rod or rods in a bank, the dropped rod, dropped rod bank, and the accidental withdrawal of a single rod (as opposed to a bank withdrawal). The last of these is a Condition III event while the others are Condition II.

The limiting static misalignment events - a single rod at bottom with the rest of the bank withdrawn and the reverse situation have been analyzed with the standard Westinghouse nuclear design codes TURTLE and LEOPARD. In neither case is the DNBR criterion violated when the core is at full power. We conclude that this analysis is acceptable.

For a dropped rod bank the reactor is tripped by the negative flux rate trip and DNBR rises from its initial value. For a dropped single rod when operating in manual mode the core power reaches a stable value below full power and the reduction in power offsets the increase in radial peaking. Thus the core DNBR value is not decreased. In automatic mode the controller withdraws the other rods to increase core power and a power overshoot may result. The limiting case has been analyzed and the results show that DNBR does not fall below the design DNBR limit. We conclude that the analyses of rod bank and single rod drop events are acceptable.

The withdrawal of a single rod from the core requires multiple equipment failures or multiple operator errors. Thus this event is classified as Condition III. This classification has been previously approved for this event and is acceptable. The Condition III classification permits a limited amount of fuel failure. The power distributions in the core are calculated by the standard Westinghouse core parameter computer codes. The THINC code is then used to obtain the resultant DNBR values.

The calculation was performed with minimum reactivity feedback and resulted in the conclusion that the bounding value of failed fuel is 5 percent of the rods in the core. We find the analysis of this event to be acceptable.

5.4.3 Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature

The inadvertent startup of an idle reactor coolant pump can, under certain conditions, result in the injection of water colder than the reactor coolant into the core. This would cause an increase in power and a reduction in DNBR. This event has been analyzed with the LOFTRAN-FACTRAN-THINC code combination previously described. Conservative input assumptions included adverse reactivity feedback and conservative trip setpoints in the protection system. The results of the analysis show that DNBR remains above the safety analysis value during the transient. We conclude that the analysis of this event is acceptable.

5.4.4 Boron Dilution Events

A decrease in the boron concentration in the core may occur if an operator error or equipment malfunction results in pumping unborated water into the core. Such events are classified as Condition II events. Analyses were performed for dilution during refueling, cold shutdown, hot shutdown, hot standby, start-up, and power operation.

During refueling the introduction of non-borated water into the core is precluded by locking the relevant valves in the closed position. The only available sources of water contain borated water. In the cold shutdown mode the increase in the source range monitor response is detected by the nuclear instrumentation and an alarm is sounded. The valves through which the clean water is being inserted are automatically closed and valves which initiate boration are opened. This stops the dilution before criticality is reached. In hot shutdown and hot standby the same instrumentation stops the dilution.

In the startup and power operation modes the shutdown and regulating rods are withdrawn. In the event of an inadvertent dilution the power will rise to the trip setpoint and the reactor will be shut down. The operator then has adequate time (20 to 40 minutes) to take action to prevent return to criticality.

Thus in all modes of operation the reactor is protected against damage due to the inadvertent dilution of the boron concentration in the core. We conclude that the analysis of this event is acceptable.

5.4.5 Rod Ejection Accident

This is a design basis (Condition IV) event and is hypothesized to occur in order to investigate the effects of the very rapid insertion of a significant amount of reactivity. The mechanical failure of the control rod mechanism pressure housing is assumed, resulting in the complete ejection of the control rod from the core in approximately 0.1 seconds. The analysis of this event was performed by the same methods and techniques which were found to be acceptable in the FSAR. Analyses were performed at zero power and at full power at both beginning and end of cycle. Conservative assumptions on reactivity feedback and power distributions were made. The results show that in no case did the peak fuel enthalpy exceed our acceptance criterion of 280 calories per gram. The pressure surge from the event was mild and did not exceed our criterion for this event. Less than 10 percent of the fuel in the hot pellet was melted as a result of the event. We conclude that the analysis of this event is acceptable.

5.5 Increase in Reactor Coolant Inventory

There are two events in this category:

1. Inadvertent Operation of the ECCS During Power Operation
2. Chemical and Volume Control System (CVCS) Malfunction that increases reactor coolant inventory.

The events are considered to be Condition II events.

5.5.1 Inadvertent Operation of the ECCS

Inadvertent operation of the Emergency Core Cooling System may occur through operator error or through equipment failure. The effect is to inject borated water having a boron concentration of 2000 ppm into the core. This has the effect of reducing the reactor power and creating a mismatch between the core and turbine. As a result the coolant decreases in temperature and shrinks. The reactor may trip on the spurious safety injection signal or on low pressurizer pressure. A turbine trip will follow and the coolant temperature will rise due to decay heat. The DNBR value increases during the transient and at no time does the pressurizer empty. We conclude that the analysis of this event is acceptable.

5.5.2 CVCS Malfunction

Increases in coolant inventory caused by the CVCS malfunction may occur due to operator error or equipment failure. In this case the injected water has the same temperature and boron concentration as that in the core and no power change or change in DNBR occurs. The effect of the malfunction is simply to initiate filling of the pressurizer. If the failure of the level trip in the pressurizer is postulated reactor trip will not occur. Alarms will however alert the operator to the situation.

Analyses have been performed for four cases: minimum and maximum reactivity feedback, each with and without automatic pressurizer spray. In each case the operator has more than 30 minutes from the receipt of the first alarm until the pressurizer fills. We conclude that this is sufficient time to permit diagnosis and correction of the error and thus the analysis is acceptable.

5.6 Decrease in Reactor Coolant Inventory

The events in this category include:

1. Inadvertent opening of a pressurizer safety or relief valve.
2. Break in instrument line or other lines
3. Steam generator tube rupture (SGTR)
4. Loss of coolant accident (LOCA)

The last two of these events is a design basis (Condition IV) accident. The rest are Condition II events.

5.6.1 Inadvertent Opening of a Pressurizer Safety or Relief Valve

The most severe event in this category is the accidental opening of a pressurizer safety valve since it has approximately twice the steam flow rate of a relief valve. The result is a rapidly decreasing reactor pressure leading to a reduction in reactor power due to the positive moderator density coefficient of reactivity. The event is terminated by the over temperature delta T or a pressurizer low pressure trip.

The event is analyzed with the LOFTRAN code and the Improved Thermal Design Procedure is used. Conservatism in the analysis includes use of most conservative reactivity feedback coefficients, neglect of void effects, and operation of the automatic rod control system. The results show that DNBR remains above the safety analysis value throughout the transient. This satisfies the criterion for the event and is acceptable.

5.6.2 Break in Instrument Line

The FSAR analysis is still valid for this event. Radiological consequences are discussed in Section 5.7 below.

5.6.3 Steam Generator Tube Rupture

The licensee provided an analysis of the Steam Generator Tube Rupture (SGTR) accident in their submittal of December 1985. Although the SGTR issue is not yet fully resolved, we have determined that there is sufficient assurance that the Callaway plant can operate safely for the next fuel cycle for the following reasons: (1) all components necessary for mitigation of the design basis SGTR are safety related; (2) the Callaway plant steam lines and supports are designed for the resulting loads if the steam lines are filled with water; and (3) there is a low probability of a SGTR approaching the severity of the design basis event during the next cycle of operation.

5.6.4 LOCA

The licensee evaluated the consequences of both large and small break loss of coolant accidents. These analyses were performed at the stretch power level which is approximately 4.5% greater than the licensed power level of 3411 MWT.

The large break LOCA calculations for Cycle 2 utilize the approved 1981 Westinghouse model which was modified to include the BART computer code for calculation of core heat transfer during reflood. Use of the BART code has been approved by the NRC staff. In the FSAR the highest cladding temperature was calculated for a double ended cold leg rupture and was determined to be 2174.2°F which is less than the acceptance criterion of 2200°F. The FSAR calculation was performed utilizing the 1978 Westinghouse model which had been superseded.

License condition 14 requires that following the first refueling outage the licensee shall submit the worst large break LOCA using the 1981 Westinghouse model. The option of using the BART code for core reflood heat transfer evaluation was included. Using the 1981 evaluation model with BART the peak cladding temperature was calculated to be 2153°F for a double ended cold leg break. The reduced cladding temperature in the revised calculation results from more realistic heat transfer modeling. BART utilizes local fluid conditions to calculate the hot channel heat transfer coefficients. The previous model utilized empirical correlations using inlet conditions and data from the FLECHT reflooding experiments.

The licensee recalculated the consequences of a spectrum of small break LOCAs using the NOTRUMP computer code which has been approved by the staff. The NOTRUMP code was developed in response to staff requirements described in Section II.K.3.30 of the TMI Action Plan (NUREG-0737).

Small break LOCA analysis methods were required to be developed which would be in compliance with Appendix K to 10 CFR 50 and which would conservatively predict trends in data from recent test loop experiments. Licensees were required to submit small break LOCA analyses using the new model under Item II.K.3.31 of the action plan.

Union Electric Company submitted small break LOCA analyses for a spectrum of postulated small break LOCA events for the stretch power level. The highest peak cladding temperature was determined for a 3 inch equivalent diameter break in a cold leg (1299°F). Larger breaks resulted in lower calculated temperatures and smaller breaks were determined not to result in core uncovery. The limiting small break LOCA analysis currently in the FSAR was performed for the licensed power level of 3411 MWT and resulted in a peak cladding temperature of 1790°F. The WFLASH code was used for this analysis. Even though the initial power level was increased, a lower cladding temperature was calculated using NOTRUMP. This is the result of models in NOTRUMP allowing draining of the hot legs into the core and improved modeling of the cold leg loop seals which reduce the extent and duration of core uncovery.

The staff concludes that license condition 14 requiring reanalysis of the worst large break LOCA is met. In addition, Callaway conforms to the requirements of TMI Action Item II.K.3.31 for a plant specific analysis of a small break LOCA. The analyses were performed at a power level 4.5% greater than that required for the current licensed power level. The results therefore indicate adequate margin for meeting the criteria of 10 CFR 50.46.

5.7 Radiological Consequences

The use of OFA fuel has a negligible impact on the source term presented in the FSAR. The use of the Improved Thermal Design Procedure results in a reduction, in some cases, of the amount of failed fuel. For most of the accidents evaluated the conclusions in the FSAR are not changed. The exceptions are discussed below.

5.7.1 Steam Line Break

The analysis of this event shows a slight reduction in steam releases compared to the FSAR values. This results in a slight (1-3 percent) reduction in doses and is acceptable.

5.7.2 Loss of AC Power to Plant Auxiliaries

As a result of the reanalysis the steam releases during the first two hours are reduced but those during the next six hours are slightly increased. Corresponding changes occur in the thyroid dose rates but the results remain well within 10 CFR 100 limits and are acceptable.

5.7.3 Locked Rotor

As a result of the use of the Improved Thermal Design Procedure, the amount of fuel which suffers DNB is reduced from that which was calculated in the FSAR. This results in a reduction (by about 25 percent) in the resultant doses. This is acceptable.

6. TECHNICAL SPECIFICATIONS

Changes in the Technical Specifications are required in order to account for the introduction of the OFA fuel, the use of the Improved Thermal Design Procedure (ITDP), the change in the $F^N_{\Delta H}$ multiplier, and the introduction of the concept of the Design Thermal Power. Each of the changes is discussed below.

<u>Definition 1.10 DESIGN THERMAL POWER</u>

This definition was added in order to permit reference to this quantity in the Technical Specification. This is acceptable.

<u>Definitions 1.11 to 1.41</u>

These definitions were renumbered to account for the insertion of Definition 1.10. This is an editorial change and is acceptable.

<u>Figure 2.1-1 REACTOR CORE SAFETY LIMITS</u>

This figure was revised to reflect the use of the ITDP and the Design Thermal Power. These are consistent with the values used in the safety analyses and are acceptable.

Table 2.2-1 REACTOR TRIP SYSTEM SETPOINTS

Changes in this table include use of minimum measured flow instead of design flow and revisions to the Overpower delta T and Overtemperature delta T trip setpoints. These changes are required to account for the use of the ITDP and the WRB-1 DNB correlation. The setpoints were derived using standard Westinghouse methods and are acceptable.

Bases 2.1.1

The bases are altered to be consistent with the altered Technical Specifications and are acceptable.

Specification 3.1.3.3 ROD DROP TIME

The rod drop time has been increased to 2.4 seconds to account for the presence of the OFA fuel. This is consistent with the standard value used for OFA fuel and is acceptable.

Figure 3.1-1 ROD BANK INSERTION LIMITS

The change from "Relative Thermal Power" to "Rated Thermal Power" is for clarification and is acceptable.

Specification 3.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The multiplier in the $F^N_{\Delta H}$ algorithm has been changed from 0.2 to 0.3. Since this change has been accounted for in the safety analyses, including the safety system setpoints, we find the change acceptable.

The curve of flow as a function of R has been deleted and the core flow requirements have been transferred to Specification 3.2.5. This is done to simplify the Technical Specification and to account for the revised handling of instrument uncertainties in the ITDP. The content of the Specification has not been changed. We find this change acceptable.

Specification 3.2.5

See discussion under Specification 3.2.3 above.

Table 3.2.1

The maximum indicated reactor coolant system average temperature was increased to account for the use of Design Thermal Power and the indicated flow value was added to the table as described above. These changes are acceptable.

Specification 4.10.2.2

The reference to Specification 4.2.3.2 was changed to Specification 4.3.2.1 to account for a numbering change. This is acceptable.

Bases

The bases of the various specifications have been altered to make them consistent with the revised Specifications. This is acceptable.

7. CONCLUSIONS

We conclude that the licensee may reload and operate the Callaway Plant, Unit 1 for Cycle 2, at the rated power of 3411 thermal megawatts, without undue hazard to the health and safety of the public. This conclusion is based on the following considerations.

1. The use of 17x17 OFA fuel, Wet Annular Burnable Poison Rods, the Improved Thermal Design and the WRB-1 DNB correlation have been generically approved for use in Westinghouse reactors. The licensee has provided the required plant specific information to support their use.

2. The methods used for the safety analyses are the same as those which were used and approved for the FSAR analyses or have been subsequently approved for use.

3. Conservative input assumptions have been used in the safety analyses.

4. The results meet the applicable acceptance criteria.

With respect to the operation at higher than current rated power, the staff may require additional information in some areas to justify operation at higher power.

8.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant change in the types or significant increase in the amounts 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.

9.0 CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (51 FR 6831) on February 16, 1986, and consulted with the state of Missouri. No public comments were received, and the state of Missouri did not have any comments.

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

10. REFERENCES

1. Letter from Donald F. Schnell, Union Electric Company, to H. R. Denton, NRC, dated November 15, 1985 (has six attachments).

2. Letter from Robert J. Schukai to H. R. Denton, NRC dated February 18, 1986.

3. Davidson, S. L.; Iorii, J. A., (Eds.), "Verification Testing and Analyses of the 17x17 Optimized Fuel Assembly," WCAP-9401-P-A and WCAP-9402-A, August 1981.

4. Chelemer, H.; Boman, L. H.; Sharp, D. R., "Improved Thermal Design Procedures," WCAP-8567 (Proprietary) and WCAP-8586 (Non-Proprietary), July 1975.

5. Motley, F. E.; Hill, K. W., et al., "New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids," WCAP-8762-P-A and WCAP-8763-A, July 1984.

6. Letter from Donald F. Schnell, Union Electric Company, to H. R. Denton, NRC dated December 13, 1985.

7. Davidson, S. L.; Iorii, J. A., "Reference Core Report - 17x17 Optimized Fuel Assembly," WCAP-9500-A, May 1982.

8. Skaritka, J., (Ed.), "Fuel Rod Bow Evaluation," WCAP-8691, Revision 1 (Proprietary), July 1979.

9. "Partial Response to Request Number-1 for Additional Information on WCAP-8691, Revision 1" Letter, E. P. Rahe, Jr., (Westinghouse) to J. R. Miller (NRC), NS-EPR-2515, dated October 9, 1981; "Remaining Response to Request Number 1 for Additional Information on WCAP-8691, Revision 1," Letter, E. P. Rahe, Jr. (Westinghouse) to J. R. Miller (NRC) NS-EPR-2572, dated March 16, 1982.

10. J. Skaritka, "Westinghouse Wet Annular Burnable Absorber Evaluation Report," WCAP-10021, February 1982.

11. Letter from Donald F. Schnell, Union Electric Company, to H. R. Denton, NRC, dated February 24, 1986.

Principal Contributors: P. O'Connor, PWR#4
W. Brooks, PARS
W. Jensen, PARS
H. Balukjian, PARS

Dated: April 8, 1986

April 8, 1986

DISTRIBUTION
Docket File
NRC PDR
Local PDR
NSIC
PWR#4 R/F
HThompson
MDuncan
TAlexion
PO'Connor
OELD
LHarmon
EJordan
BGrimes
JPartlow
TBarnhart (4)
WJones
ACRS (10)
OPA
LFMB
VBenaroya