Docket

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 27, 1994

Docket Nos. 50-498
and 50-499

Mr. William T. Cottle
Group Vice-President, Nuclear
Houston Lighting & Power Company
South Texas Project Electric
Generating Station
P. O. Box 289
Wadsworth, Texas 77483

Dear Mr. Cottle:

SUBJECT: SOUTH TEXAS PROJECT, UNITS 1 AND 2 - AMENDMENT NOS. 61 AND 50 TO FACILITY OPERATING LICENSE NOS. NPF-76 AND NPF-80 (TAC NOS. M86688 AND M86689)

The Commission has issued the enclosed Amendment Nos. 61 and 50 to Facility Operating License Nos. NPF-76 and NPF-80 for the South Texas Project, Units 1 and 2 (STP). The amendments consist of changes to the Technical Specifications (TSs) and the Updated Final Safety Analysis Report (UFSAR) in response to your application dated May 27, 1993, as supplemented by letter dated April 18, 1994.

The amendments revise Technical Specifications Tables 2.2-1 and 3.3-4; Figures 2.1-1, 3.1-1, 3.1-2, 3.1-2a and 5.6-7; and Technical Specifications 3/4.2.5, 3/4.6.1.1, 3/4.6.1.2, 3/4.6.1.3, 3/4.6.1.5, 3/4.7.1.2, 5.2.1, 5.3.1, 5.6.1, and 5.2.2 and associated Bases to accommodate an upgrade of the fuel used in the STP reactors to Westinghouse VANTAGE 5 Hybrid (V5H) design. Several safety analysis and operational margin improvements to the STP UFSAR result from these amendments.

Your submittal proposed that the STP design basis limiting reactor coolant system peak pressure criterion for locked rotor events be changed in the UFSAR from the current basis of 110 percent of design pressure (2750 psi) to faulted stress limits (about 2900 psi). The staff considers the acceptance criteria for accident analyses to be generic positions. Consequently, the staff does not accept this plant specific proposal, and continues to evaluate STP locked rotor event analyses by its current RCS pressure criterion.

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A copy of our related Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

Original Signed By

Lawrence E. Kokajko, Senior Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 61 to NPF-76
2. Amendment No. 50 to NPF-80
3. Safety Evaluation

cc w/enclosures:
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

HOUSTON LIGHTING & POWER COMPANY

CITY PUBLIC SERVICE BOARD OF SAN ANTONIO

CENTRAL POWER AND LIGHT COMPANY

CITY OF AUSTIN, TEXAS

DOCKET NO. 50-498

SOUTH TEXAS PROJECT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 61
License No. NPF-76

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

   A. The application for amendment by Houston Lighting & Power Company* (HL&P) acting on behalf of itself and for the City Public Service Board of San Antonio (CPS), Central Power and Light Company (CPL), and City of Austin, Texas (COA) (the licensees) dated May 27, 1993, as supplemented by letter dated April 18, 1994, 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, as amended, 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 license 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.

*Houston Lighting & Power Company is authorized to act for the City Public Service Board of San Antonio, Central Power and Light Company and City of Austin, Texas and has exclusive responsibility and control over the physical construction, operation and maintenance of the facility.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility Operating License No. NPF-76 is hereby amended to read as follows:

   2. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 61, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and is to be implemented prior to the completion of the Unit 1 RE05 outage.

FOR THE NUCLEAR REGULATORY COMMISSION

James E. Lyons

James E. Lyons, Acting Director
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 27, 1994

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

HOUSTON LIGHTING & POWER COMPANY

CITY PUBLIC SERVICE BOARD OF SAN ANTONIO

CENTRAL POWER AND LIGHT COMPANY

CITY OF AUSTIN, TEXAS

DOCKET NO. 50-499

SOUTH TEXAS PROJECT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 50
License No. NPF-80

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

A. The application for amendment by Houston Lighting & Power Company* (HL&P) acting on behalf of itself and for the City Public Service Board of San Antonio (CPS), Central Power and Light Company (CPL), and City of Austin, Texas (COA) (the licensees) dated May 27, 1993, as supplemented by letter dated April 18, 1994, 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, as amended, 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 license 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.

*Houston Lighting & Power Company is authorized to act for the City Public Service Board of San Antonio, Central Power and Light Company and City of Austin, Texas and has exclusive responsibility and control over the physical construction, operation and maintenance of the facility.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility Operating License No. NPF-80 is hereby amended to read as follows:

   2. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 50, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and is to be implemented prior to the completion of the Unit 1 RE05 outage.

FOR THE NUCLEAR REGULATORY COMMISSION

James E. Lyons

James E. Lyons, Acting Director
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 27, 1994

ATTACHMENT TO LICENSE AMENDMENT NOS. 61 AND 50

FACILITY OPERATING LICENSE NOS. NPF-76 AND NPF-80

DOCKET NOS. 50-498 AND 50-499

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain marginal lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
viii	viii
xvii	xvii
2-2	2-2
2-4	2-4
2-5	2-5
2-6	2-6
2-7	2-7
2-8	2-8
2-10	2-10
B 2-1	B 2-1
3/4 1-3	3/4 1-3
3/4 1-5	3/4 1-5
3/4 1-7a	3/4 1-7a
3/4 2-11	3/4 2-11
3/4 3-29	3/4 3-29
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3/4 6-5	3/4 6-5
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3/4 6-8	3/4 6-8
3/4 7-5	3/4 7-5
B 3/4 1-1	B 3/4 1-1
B 3/4 1-2	B 3/4 1-2
B 3/4 2-4	B 3/4 2-4
B 3/4 2-6	B 3/4 2-6
B 3/4 6-1	B 3/4 6-1
B 3/4 6-2	B 3/4 6-2
B 3/4 7-2	B 3/4 7-2

REMOVE	INSERT
5-1	5-1
5-6	5-6
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5-9	5-9
---	5-9a
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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

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ADMINISTRATIVE CONTROLS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

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

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.

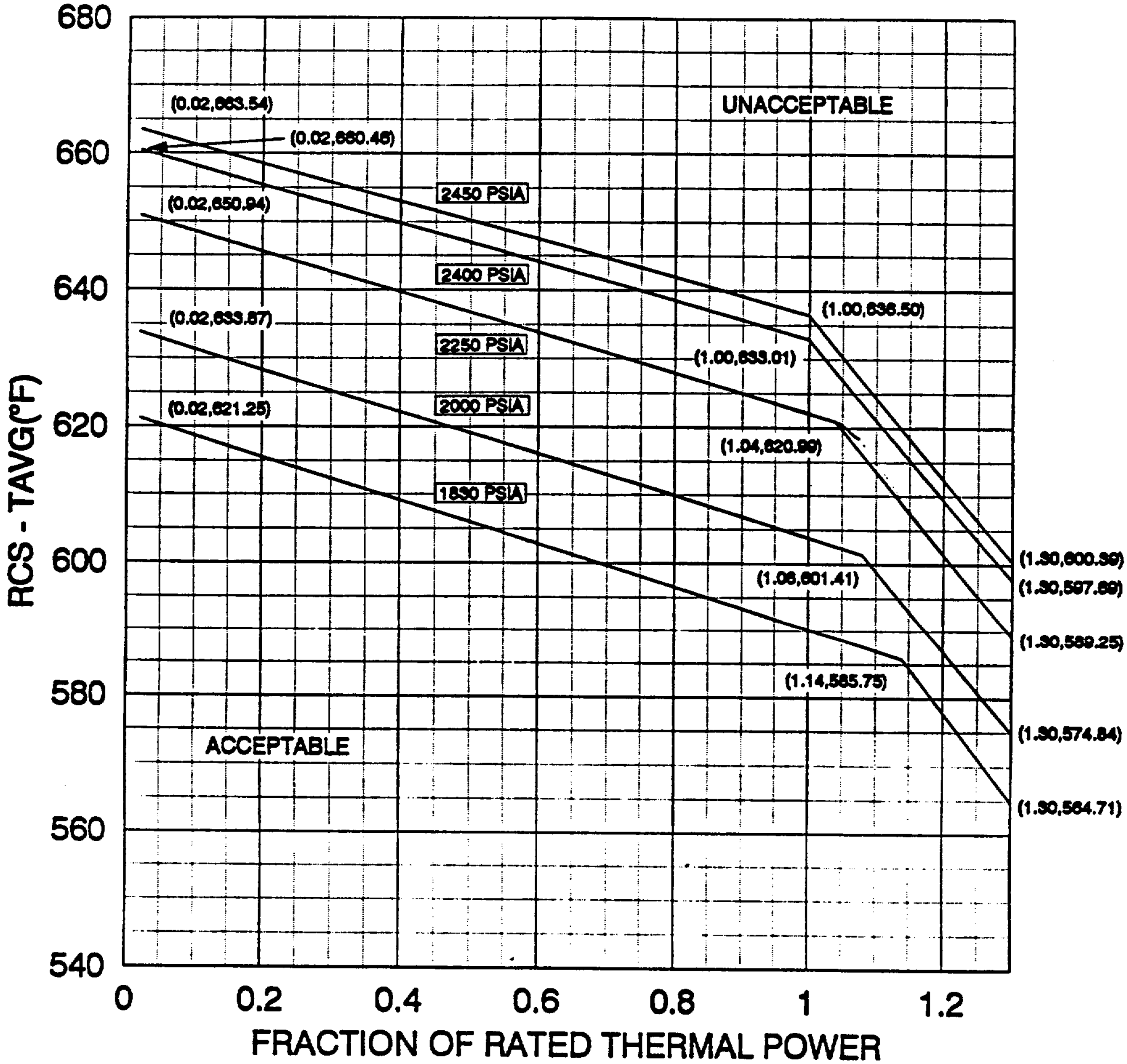

FIGURE 2.1-1

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 Interlock 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 Value 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 Columh 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)	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	6.1	0	≤109% of RTP**	≤110.7% of RTP**
b. Low Setpoint	8.3	6.1	0	≤25% of RTP**	≤27.7% of RTP**
3. Power Range, Neutron Flux, High Positive Rate	2.1	0.5	0	≤5% of RTP** with a time constant ≥2 seconds	≤6.7% of RTP** with a time constant ≥2 seconds
4. Deleted					
5. Intermediate Range, Neutron Flux	16.7	8.4	0	≤25% of RTP**	≤31.1% of RTP**
6. Source Range, Neutron Flux	17.0	10.0	0	≤10^5 cps	≤1.4 x 10^5 cps
7. Overtemperature ΔT	10.7	8.7	1.5 + 1.5#	See Note 1	See Note 2
8. Overpower ΔT	4.7	2.1	1.5	See Note 3	See Note 4
9. Pressurizer Pressure-Low	5.0	2.3	2.0	≥1870 psig	≥1860 psig
10. Pressurizer Pressure-High	5.0	2.3	2.0	≤2380 psig	≤2390 psig
11. Pressurizer Water Level-High	7.1	4.3	2.0	≤92% of instrument span	≤94.1% of instrument span
12. Reactor Coolant Flow-Low	4.0	2.1	0.6	≥91.8% of loop design flow*	≥90.5% of loop design flow*

*Loop design flow = 95,400 gpm
**RTP = RATED THERMAL POWER
#1.5% span for ΔT; 1.5% span for Pressurizer Pressure

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
13. Steam Generator Water Level Low-Low	20.0	15.3	2.0 + 0.2##	≥33% of narrow range instrument span	≥30.7% of narrow range instrument span
14. Undervoltage - Reactor Coolant Pumps	11.9	0.3	0	≥10,014 volts	≥9339 volts
15. Underfrequency - Reactor Coolant Pumps	3.4	0.0	0	≥57.2 Hz	≥57.1 Hz
16. Turbine Trip					
a. Low Emergency Trip Fluid Pressure	232.1	100.8	131.3	≥1245.8 psig	≥1114.5 psig
b. Turbine Stop Valve Closure	N.A.	N.A.	N.A.	<Fully closed	Fully closed
17. Safety Injection Input from ESFAS	N.A.	N.A.	N.A.	N.A.	N.A.

##2.0% span for Steam Generator Level; 0.2% span for Reference Leg RTDs

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
18. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	N.A	N.A.	N.A.	$\geq 1 \times 10^{-10}$ amp	$\geq 6 \times 10^{-11}$ amp
b. Low Power Reactor Trips Block, P-7					
1) P-10 input	N.A.	N.A.	N.A.	≤10% of RTP**	≤11.7% of RTP**
2) P-13 input	N.A.	N.A.	N.A.	≤10% RTP** Turbine Impulse Pressure Equivalent	≤11.7% RTP** Turbine Impulse Pressure Equivalent
c. Power Range Neutron Flux, P-8	N.A.	N.A.	N.A.	≤40% of RTP**	≤41.7% of RTP**
d. Power Range Neutron Flux, P-9	N.A.	N.A.	N.A.	≤50% of RTP**	≤51.7% of RTP**
e. Power Range Neutron Flux, P-10	N.A.	N.A.	N.A	≥10% of RTP**	≥8.3% of RTP**
f. Turbine Impulse Chamber Pressure, P-13	N.A.	N.A.	N.A.	≤10% RTP** Turbine Impulse Pressure Equivalent	≤11.7% RTP** Turbine Impulse Pressure Equivalent
19. Reactor Trip Breakers	N.A.	N.A.	N.A	N.A.	N.A.
20. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	N.A.

**RTP = RATED THERMAL POWER

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 RCS Instrumentation;
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	Lead-lag compensator on measured ΔT;
τ_1, τ_2	=	Time constant utilized in lead-lag compensator for ΔT, τ_1 = 8 sec, τ_2 = 3 sec;
$\frac{1}{1 + \tau_3 S}$	=	Lag compensator on measured ΔT;
τ_3	=	Time constant utilized in the lag compensator for ΔT, τ_3 = 0 sec;
ΔT_o	=	Indicated ΔT at RATED THERMAL POWER;
K_1	=	1.14;
K_2	=	0.028 /°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 sec, τ_5 = 4 sec;
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 sec;

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

T'	$\leq$	593.0°F (Nominal T_{avg} at RATED THERMAL POWER);
K_3	=	0.00143 /psig;
P	=	Pressurizer pressure, psig;
P'	=	2235 psig (Nominal RCS operating pressure);
S	=	Laplace transform operator, sec^{-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:

(1) For $q_t - q_b$ between -70% and + 8 %, $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(2) For each percent that the magnitude of $q_t - q_b$ exceeds -70%, the ΔT Trip Setpoint shall be automatically reduced by 0.0% of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t - q_b$ exceeds +8 %, the ΔT Trip Setpoint shall be automatically reduced by 2.65% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.6% ΔT span.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

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

Where: ΔT = As defined in Note 1,

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = As defined in Note 1,

τ_1, τ_2 = As defined in Note 1,

$\frac{1}{1 + \tau_3 S}$ = As defined in Note 1,

τ_3 = As defined in Note 1,

ΔT_o = As defined in Note 1,

K_4 = 1.08,

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 sec,

$\frac{1}{1 + \tau_6 S}$ = As defined in Note 1,

τ_6 = As defined in Note 1,

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6	=	0.002 /°F for $T > T''$ and $K_6 = 0$ for $T \leq T''$,
T	=	As defined in Note 1,
T''	=	Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 593.0°F),
S	=	As defined in Note 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 1.9% 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 through the WRB-1 correlation. The WRB-1 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux and is indicative of the margin to DNB.

The DNB design basis is as follows: uncertainties in the WRB-1 correlation, plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, and computer codes are considered statistically such that there is at least a 95 percent probability with a 95 percent confidence level that DNBR will not occur on the most limiting fuel rod during Condition I and II events. This establishes a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties. In addition, margin has been maintained in the design by meeting safety analysis DNBR limits 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.

These curves are based on a nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$, and a reference cosine with a peak of 1.61 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = F^{RTP}_{\Delta H} \, [1 + PF_{\Delta H} \, (1-P)]$$

where: $F^{RTP}_{\Delta H}$ is the limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMITS REPORT (COLR);

$PF_{\Delta H}$ is the Power Factor Multiplier for $F^N_{\Delta H}$ specified in the COLR; and,

P is the fraction of RTP.

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

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

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

The reactor vessel, pressurizer, and the RCS piping, valves, and fittings 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 125% (3110 psig) of design pressure, to demonstrate integrity prior to initial operation.

REQUIRED SHUTDOWN MARGIN FOR MODES 1 AND 2
1.30% DELTA RHO

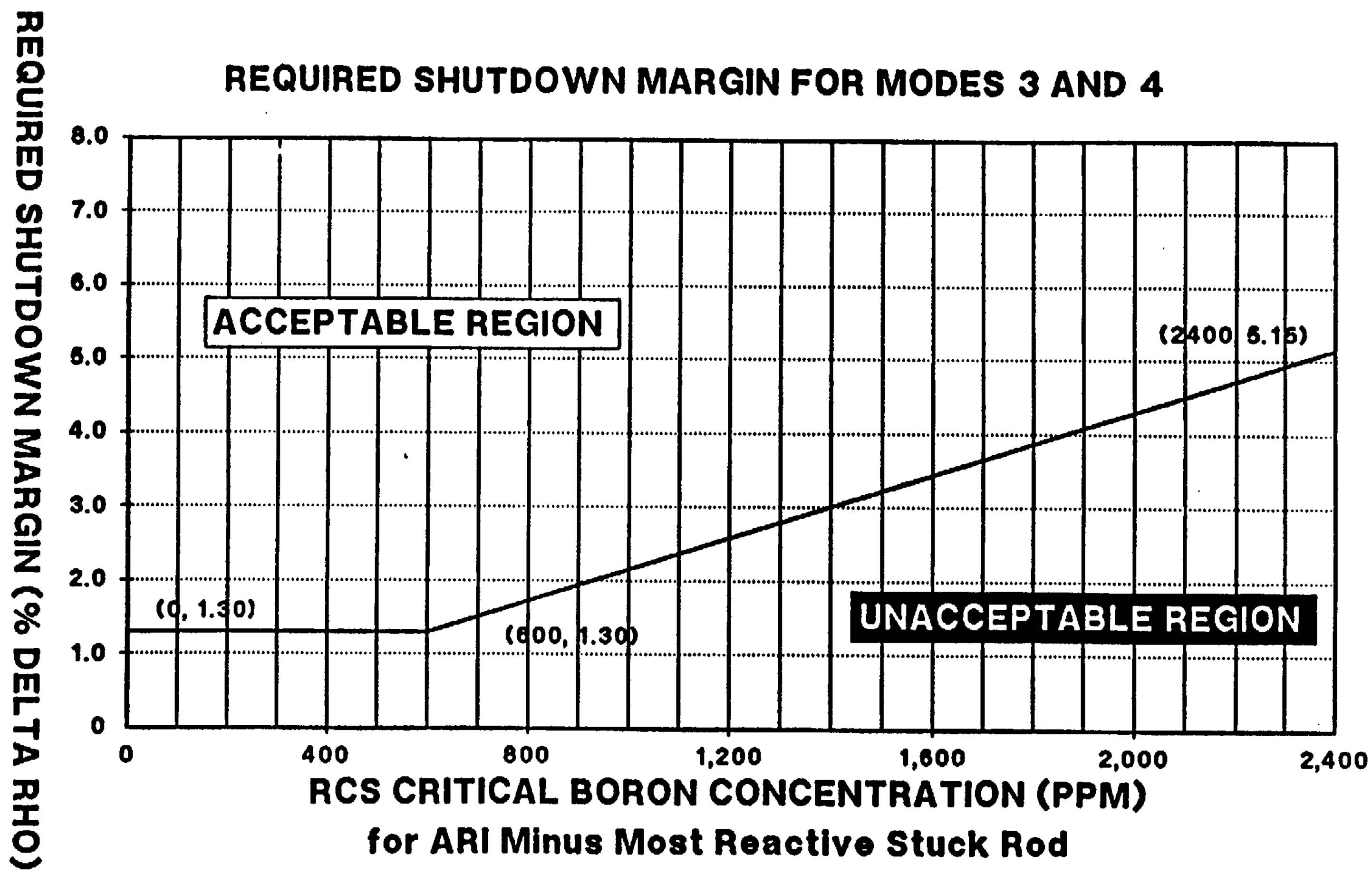

FIGURE 3.1-1

REQUIRED SHUTDOWN MARGIN VERSUS RCS CRITICAL BORON CONCENTRATION

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

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to the limit as shown in Figure 3.1-2.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than the limit as shown in Figure 3.1-2, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to the limit as shown in Figure 3.1-2:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s); and

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

   1) Reactor Coolant System boron concentration,

   2) Control rod position,

   3) Reactor Coolant System average temperature,

   4) Fuel burnup based on gross thermal energy generation,

   5) Xenon concentration, and

   6) Samarium concentration.

REQUIRED SHUTDOWN MARGIN FOR MODE 5

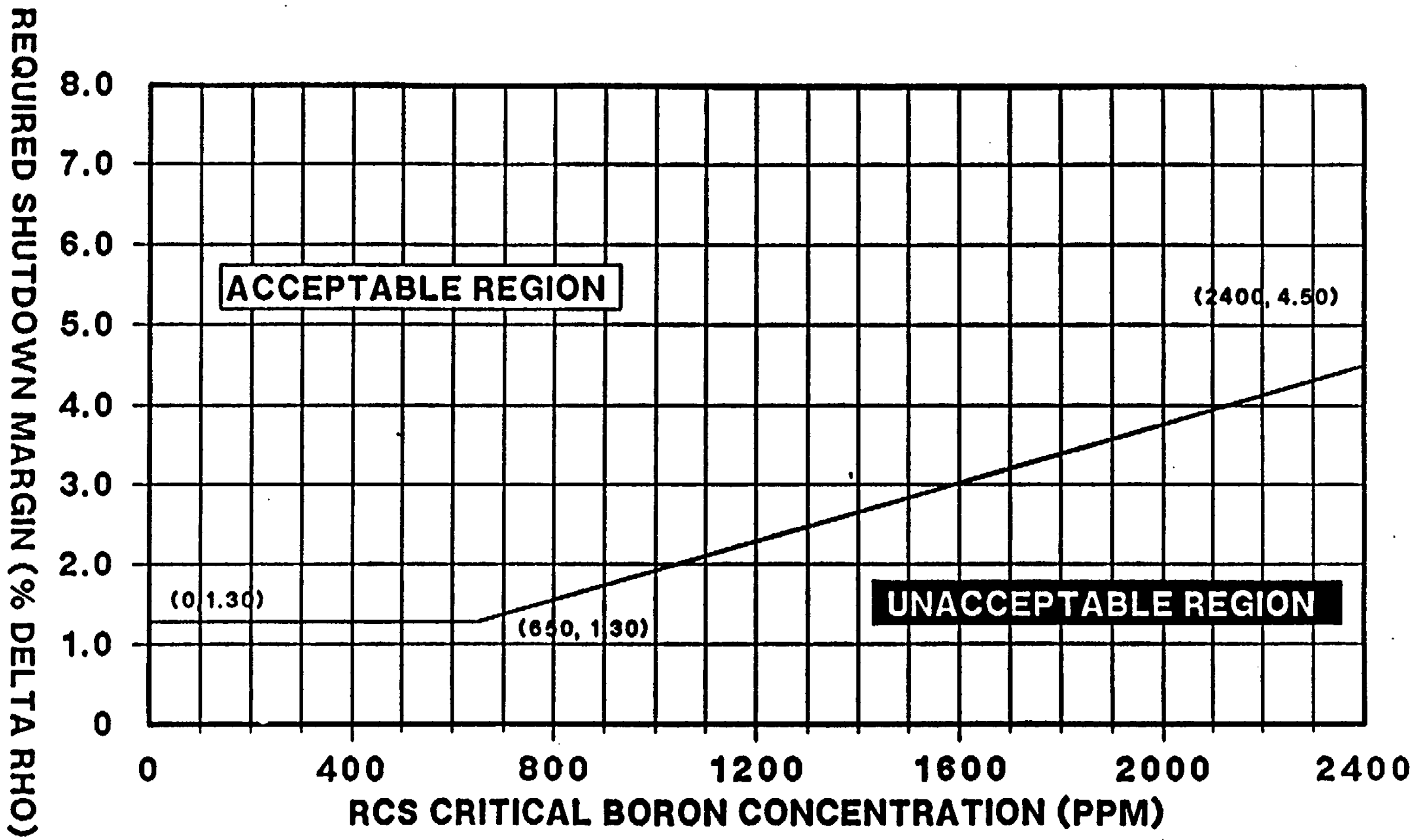

FIGURE 3.1-2

REQUIRED SHUTDOWN MARGIN VERSUS RCS CRITICAL BORON CONCENTRATION

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be within the limits specified in the Core Operating Limits Report (COLR). The maximum upper limit shall be less than or equal to that shown in Figure 3.1-2a.

APPLICABILITY: Beginning of Life (BOL) limit - MODES 1 and 2* only**.
End of Life (EOL) limit - MODES 1, 2, and 3 only**.

ACTION:

a. With the MTC more positive than the BOL limit specified in the COLR, operation in MODES 1 and 2 may proceed provided:

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

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

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

b. With the MTC more negative than the EOL limit specified in the COLR, be in HOT SHUTDOWN within 12 hours.

---

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

**See Special Test Exceptions Specification 3.10.3.

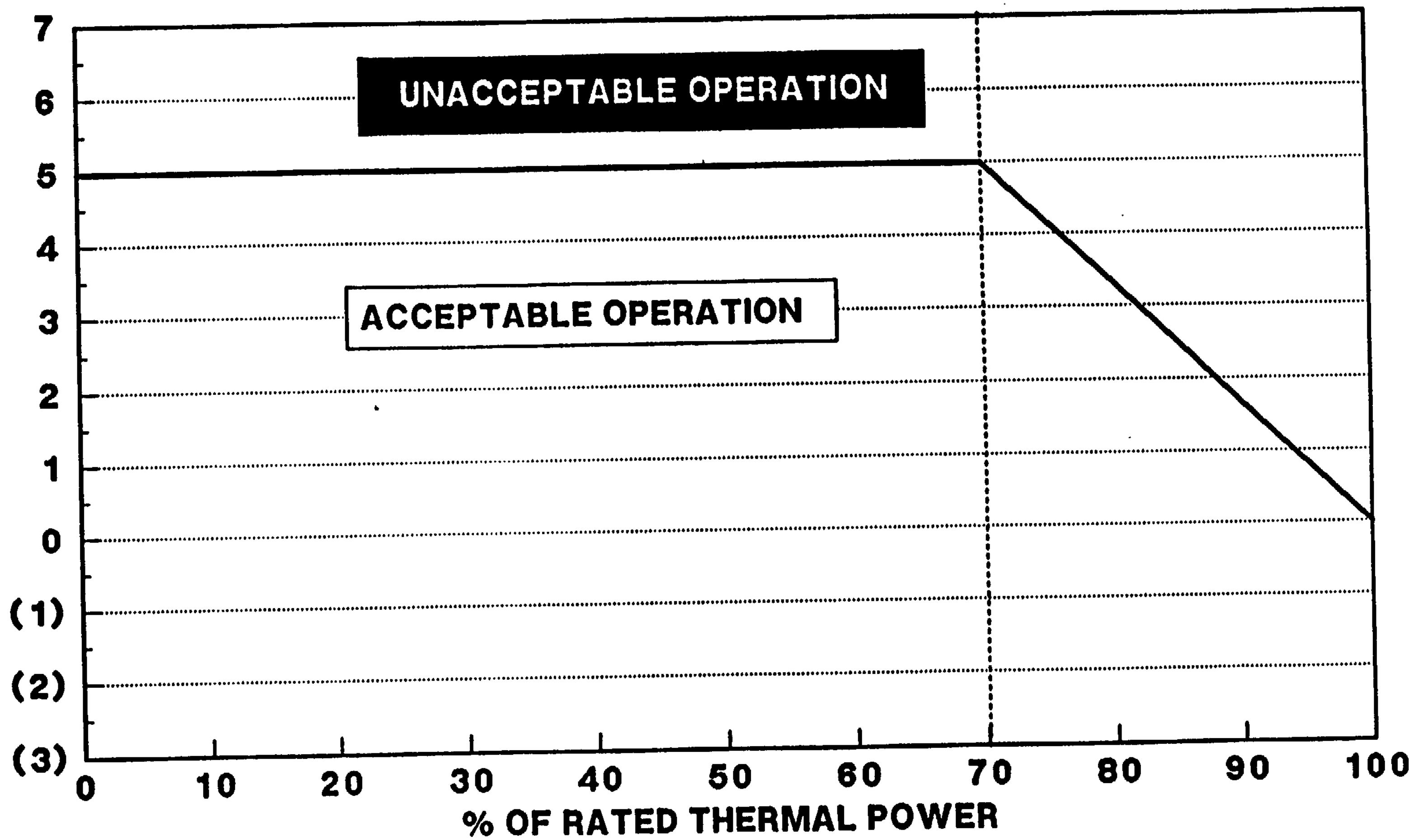

Figure 3.1-2a

MTC versus Power Level

NOTE: Cycle specific MTC limits are displayed in the COLR.

REACTIVITY CONTROL SYSTEMS

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.4 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be greater than or equal to 561°F.

APPLICABILITY: MODES 1 and 2* **.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) less than 561°F, restore T_{avg} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.4 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 561°F:

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

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{avg} is less than 571°F with the T_{avg}-T_{ref} Deviation Alarm not reset.

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

**See Special Test Exceptions Specification 3.10.3.

POWER DISTRIBUTION LIMITS

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

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

a. Reactor Coolant System T_{avg}, $\leq$ 598°F

b. Pressurizer Pressure, > 2189 psig*

c. Reactor Coolant System Flow, $\geq$ 392,300 gpm**

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 shown above shall be verified to be within its limits at least once per 12 hours. The provisions of Specification 4.0.4 are not applicable for verification that RCS flow is within its limit.

4.2.5.2 The RCS flow rate indicators shall be subjected to a channel calibration at least once per 18 months.

4.2.5.3 The RCS total flow rate shall be determined by precision heat balance measurements at least once per 18 months. The provisions of Specification 4.0.4 are not applicable.

* Limit not applicable during either a Thermal Power ramp in excess of 5% of RTP per minute or a Thermal Power step in excess of 10% RTP.

**Includes a 2.8% flow measurement uncertainty.

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Emergency Ventilation, Start Standby Diesel Generators, Reactor Containment Fan Coolers, and Essential Cooling Water)					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	N.A.
c. Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
d. Containment Pressure--High 1	3.6	0.7	2.0	≤ 3.0 psig	≤ 4.0 psig
e. Pressurizer Pressure--Low	19.6	17.4	2.0	≥ 1857 psig	≥ 1851 psig
f. Compensated Steam Line Pressure-Low	16.4	12.8	2.0	≥ 735 psig	≥ 709 psig*
2. Containment Spray					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	N.A.
c. Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
d. Containment Pressure--High-3	3.6	0.7	2.0	≤ 9.5 psig	≤ 10.5 psig

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
2) Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	N.A.
3) Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
4) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
b. Containment Ventilation Isolation					
1) Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	N.A.
2) Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
3) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
4) RCB Purge Radioactivity-High	3.1×10^{-4} µCi/cc	1.8×10^{-4} µCi/cc	1.3×10^{-4} µCi/cc	$\leq 5 \times 10^{-4}$ µCi/cc ###	$\leq 6.4 \times 10^{-4}$ µCi/cc
5) Containment Spray - Manual Initiation	See Item 2. above for Containment Spray manual initiation Trip Setpoints and Allowable Values.				
6) Phase "A" Isolation - Manual Initiation	See Item 3.a. above for Phase "A" Isolation manual initiation Trip Setpoints and Allowable Values.				
c. Phase "B" Isolation					
1) Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	N.A.
2) Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
3) Containment Pressure--High-3	3.6	0.7	2.0	$\leq$ 9.5 psig	$\leq$ 10.5 psig
4) Containment Spray-Manual Initiation	See Item 2. above for Containment Spray manual initiation Trip Setpoints and Allowable Values.				

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
d. RCP Seal Injection Isolation					
1) Automatic Acutation Logic and Activation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
2) Charging Header Pressure - Low	4.6	1.0	2.0	≥ 560.0 psig	≥ 495.4 psig
Coincident with Phase "A" Isolation	See Item 3.a. above for Phase "A" Isolation Setpoints and Allowable Values				
4. Steam Line Isolation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Steam Line Pressure - Negative Rate--High	2.6	0.5	0	≤ 100 psi	≤ 126 psi**
d. Containment Pressure - High-2	3.6	0.7	2.0	≤ 3.0 psig	≤ 4.0 psig
e. Compensated Steam Line Pressure - Low	16.4	12.8	2.0	≥ 735 psig	≥ 709 psig*
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. Steam Generator Water Level--High-High (P-14)	10.8	6.5	2.0+0.2#	≤ 87.5% of narrow range instrument span.	≤ 89.8% of narrow range instrument span.
c. Deleted					

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
5. Turbine Trip and Feedwater Isolation (Continued)					
d. Deleted					
e. Safety Injection	See Item 1 above for all Safety Injection Trip Setpoints and Allowable Values.				
f. T_{avg}-Low Coincident with Reactor Trip (P-4) (Feedwater Isolation Only)	4.5	1.1	0.8	$\geq$ 574°F	$\geq$ 571.7 °F
6. Auxiliary Feedwater					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	N.A.
c. Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
d. Steam Generator Water Level--Low-Low	20.0	15.3	2.0+0.2#	$\geq$ 33.0% of narrow range instrument span.	$\geq$ 30.7% of narrow range instrument span.
e. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
6. Auxiliary Feedwater (Continued)					
f. Loss of Power (Motor Driven Pumps Only)	See Item 8. below for all Loss of Power Trip Setpoints and Allowable Values.				
7. Automatic Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. RWST Level--Low-Low	5.0	1.21	2.0	$\geq$ 11%	$\geq$ 9.1%
Coincident With: Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
8. Loss of Power					
a. 4.16 kV ESF Bus Undervoltage (Loss of Voltage)	N.A.	N.A.	N.A.	$\geq$ 3107 volts with a $\leq$ 1.75 second time delay.	$\geq$ 2979 volts with a $\leq$ 1.93 second time delay.
b. 4.16 kV ESF Bus Undervoltage (Tolerable Degraded Voltage Coincident with SI)	N.A.	N.A.	N.A.	$\geq$ 3835 volts with a $\leq$ 35 second time delay.	$\geq$ 3786 volts with a $\leq$ 39 second time delay.
c. 4.16 kV ESF Bus Undervoltage (Sustained Degraded Voltage)	N.A.	N.A.	N.A.	$\geq$ 3835 volts with a $\leq$ 50 second time delay.	$\geq$ 3786 volts with a $\leq$ 55 second time delay.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
9. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	N.A.	N.A.	N.A.	$\leq$ 1985 psig	$\leq$ 1995 psig
b. Low-Low T_{avg}, P-12	N.A.	N.A.	N.A.	$\geq$ 563°F	$\geq$ 560.7 °F
c. Reactor Trip, P-4	N.A.	N.A.	N.A.	N.A.	N.A.
10. Control Room Ventilation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
c. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
d. Control Room Intake Air Radioactivity - High	3.7×10^{-5} μCi/cc	2.2×10^{-5} μCi/cc	1.6×10^{-5} μCi/cc	$\leq 6.1\times10^{-5}$ μCi/cc	$\leq 7.8\times10^{-5}$ μCi/cc
e. Loss of Power	See Item 8. above for all Loss of Power Trip Setpoints and Allowable Values.				
11. FHB HVAC					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
11. FHB HVAC (Continued)					
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
d. Spent Fuel Pool Exhuast Radioactivity - High	3.1×10^{-4} µCi/cc	1.8×10^{-4} µCi/cc	1.3×10^{-4} µCi/cc	$\leq 5.0 \times 10^{-4}$ µCi/cc	$\leq 6.4 \times 10^{-4}$ µCi/cc

TABLE 3.3-4 (Continued)

TABLE NOTATIONS

* Time constants utilized in the lead-lag controller for Steam Line Pressure-Low are τ, $\geq$ 50 seconds and τ, $\leq$ 5 seconds. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

** The time constant utilized in the rate-lag controller for Steam Line Pressure-Negative Rate-High is greater than or equal to 50 seconds. CHANNEL CALIBRATION shall ensure that this time constant is adjusted to this value.

\# 2.0% span for Steam Generator Level; 0.2% span for Reference Leg RTDs.

\#\# Deleted

\#\#\#This setpoint value may be increased up to the equivalent limits of ODCM Control 3.11.2.1 in accordance with the methodology and parameters of the ODCM during containment purge or vent for pressure control, ALARA and respirable air quality considerations for personnel entry.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

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

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Specification 3.6.3;

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3; and

c. After each closing of each penetration subject to Type B testing, except the containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at a pressure not less than P_a, 41.2 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2d. for all other Type B and C penetrations, the combined leakage rate is less than 0.60 L_a.

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*Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

CONTAINMENT SYSTEMS

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of less than or equal to L_a, 0.30% by weight of the containment air per 24 hours at P_a, 41.2 psig.

b. A combined leakage rate of less than 0.60 L_a for all penetrations and valves subject to Type B and C tests, when pressurized to P_a.

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

ACTION:

With either the measured overall integrated containment leakage rate exceeding 0.75 L_a or the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 L_a, restore the overall integrated leakage rate to less than 0.75 L_a and the combined leakage rate for all penetrations subject to Type B and C tests to less than 0.60 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria, methods and provisions specified or endorsed in Appendix J or 10 CFR Part 50:

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during shutdown at a pressure not less than P_a, 41.2 psig, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection;

SURVEILLANCE REQUIREMENTS (Continued)

b. If any periodic Type A test fails to meet 0.75 L_a, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet 0.75 L_a, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75 L_a at which time the above test schedule may be resumed;

c. The accuracy of each Type A test shall be verified by a supplemental test which:

   1) Confirms the accuracy of the test by verifying that the supplemental test result, L_c, is in accordance with the following equation:

$$|L_c-(L_{am} + L_o)| \leq 0.25\ L_a$$

   where L_{am} is the measured Type A test leakage and L_o is the superimposed leak;

   2) Has a duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test; and

   3) Requires that the rate at which gas is injected into the containment or bled from the containment during the supplemental test is between 0.75 L_a and 1.25 L_a.

d. Type B and C tests shall be conducted with gas at a pressure not less than P_a, 41.2 psig, at intervals no greater than 24 months except for tests involving:

   1) Air locks,

   2) Purge supply and exhaust isolation valves with resilient material seals, and

   3) Penetrations using continuous Leakage Monitoring Systems.

e. Air locks shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.3;

f. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.7.2 or 4.6.1.7.3, as applicable;

SURVEILLANCE REQUIREMENTS (Continued)

g. Leakage from isolation valves that are sealed with fluid from a Seal System may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the Seal System and valves are pressurized to at least 1.10 P_a and the seal system capacity is adequate to maintain system pressure for at least 30 days;

h. Type B tests for penetrations employing a continuous Leakage Monitoring System shall be conducted at P_a, 41.2 psig, at intervals no greater than once per 3 years; and

i. The provisions of Specification 4.0.2 are not applicable.

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a, 41.2 psig.

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

ACTION:

a. With one containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed;

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days;

3. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and

4. The provisions of Specification 3.0.4 are not applicable.

b. With the containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying seal leakage is less than 0.01 L_a as determined by precision flow measurements when measured for at least 30 seconds with the volume between the seals at a constant pressure not less than P_a;

b. By conducting overall air lock leakage tests at not less than P_a, 41.2 psig, and verifying the overall air lock leakage rate is within its limit:

   1) At least once per 6 months,* and

   2) Prior to establishing CONTAINMENT INTEGRITY when maintenance has been performed on the air lock that could affect the air lock sealing capability.**

c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

d. By verifying at least once per 7 days that the instrument air pressure in the header to the personnel airlock seals is ≥ 90 psig.

e. By verifying the door seal pneumatic system OPERABLE at least once per 18 months by conducting a seal pneumatic system leak test and verifying that system pressure does not decay more than 1.5 psi from 90 psig minimum within 24 hours.

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* The provisions of Specification 4.0.2 are not applicable.

**This represents an exemption to Appendix J. paragraph III.D.2 of 10 CFR Part 50.

CONTAINMENT SYSTEMS

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between -0.1 and +0.3 psig.

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

ACTION:

With the containment internal pressure outside of the limits above, restore the internal pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The primary containment internal pressure shall be determined to be within the limits at least once per 12 hours.

CONTAINMENT SYSTEMS

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall not exceed 110°F.

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

ACTION:

With the containment average air temperature greater than 110°F, reduce the average air temperature to within the limit within 8 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5 The primary containment average air temperature shall be the arithmetical average of a minimum of four RCFC inlet temperatures and shall be determined at least once per 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1) Verifying that each motor-driven pump develops a discharge pressure of greater than or equal to 1454 psig at a flow of greater than or equal to 500 gpm;

2) Verifying that the steam turbine-driven pump develops a discharge pressure of greater than or equal to 1454 psig at a flow of greater than or equal to 500 gpm when the secondary steam supply pressure is greater than 1000 psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3;

3) Verifying that each non-automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position; and

4) Verifying that each automatic valve in the flow path is in the correct position whenever the Auxiliary Feedwater System is placed in automatic control or when above 10% RATED THERMAL POWER.

b. At least once per 18 months during shutdown by:

1) Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of an Auxiliary Feedwater Actuation test signal, and

2) Verifying that each auxiliary feedwater pump starts as designed automatically upon receipt of an Auxiliary Feedwater Actuation test signal.

3) Verifying that each auxiliary feedwater flow regulating valve limits the flow to each steam generator between 550 gpm and 675 gpm.

4.7.1.2.2 An auxiliary feedwater flow path to each steam generator shall be demonstrated OPERABLE following each COLD SHUTDOWN of greater than 30 days prior to entering MODE 2 by verifying normal flow to each steam generator.

AUXILIARY FEEDWATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The auxiliary feedwater storage tank (AFST) shall be OPERABLE with a contained water volume of at least 485,000 gallons of water.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With the AFST inoperable, within 4 hours restore the AFST to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3 The AFST shall be demonstrated OPERABLE at least once per 12 hours by verifying the contained water volume is within its limits.

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

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

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. In MODES 1 and 2, the most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN OF 1.3% $\Delta k/k$ is required to control the reactivity transient. The 1.3% $\Delta k/k$ SHUTDOWN MARGIN is the design basis minimum for the 14-foot fuel using silver-indium-cadmium and/or Hafnium control rods (Ref. FSAR Table 4.3-3). Accordingly, the SHUTDOWN MARGIN requirement for MODES 1 and 2 is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. In MODES 3, 4, and 5, the most restrictive condition occurs at BOL, when the boron concentration is the greatest. In these modes, the required SHUTDOWN MARGIN is composed of a constant requirement and a variable requirement, which is a function of the RCS boron concentration. The constant SHUTDOWN MARGIN requirement of 1.3% $\Delta k/k$ is based on an uncontrolled RCS cooldown from a steamline break accident. The variable SHUTDOWN MARGIN requirement is based on the results of a boron dilution accident analysis, where the SHUTDOWN MARGIN is varied as a function of RCS boron concentration, to guarantee a minimum of 15 minutes for operator action after a boron dilution alarm, prior to a loss of all SHUTDOWN MARGIN.

The boron dilution analysis assumed a common RCS volume, and maximum dilution flow rate for MODES 3 and 4, and a different volume and flow rate for MODE 5. The MODE 5 conditions assumed limited mixing in the RCS and cooling with the RHR system only. The MODE 5 SHUTDOWN MARGIN curve (Figure 3.1-2) can be used to provide the required C_B in the rapid refueling condition (MODE 5 with ARO). The cycle-specific reload safety analysis verifies this curve to be bounding in this condition.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

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

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

MODERATOR TEMPERATURE COEFFICIENT (Continued)

The most negative MTC value, equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analysis to nominal operating conditions. These corrections involved: (1) a conversion of the MDC used in the FSAR analysis to its equivalent MTC, based on the rate of change of moderator density with temperature at RATED THERMAL POWER conditions, and (2) subtracting from this value the largest differences in MTC observed at EOL, all rods withdrawn, RATED THERMAL POWER conditions, and those most adverse conditions of moderator temperature and pressure, rod insertion, axial power skewing, and xenon concentration that can occur in nominal operation and lead to a significantly more negative EOL MTC at RATED THERMAL POWER. These corrections transformed the MDC values used in the FSAR analysis into the limiting EOL MTC value specified in the CORE OPERATING LIMITS REPORT (COLR). The 300 ppm surveillance MTC value specified in the COLR represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration, and is obtained by making these corrections to the limiting MTC value.

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

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 561°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the trip instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) the reactor vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

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

With the RCS average temperature above 350°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1.3% $\Delta k/k$ after xenon decay and cooldown to 200°F. The maximum expected boration capability requires 27,000 gallons of 7000 ppm borated water from the boric acid storage system or 458,000 gallons of 2800 ppm borated water from the refueling water storage tank (RWST). The RWST volume is an ECCS requirement and is more than adequate for the required boration capability.

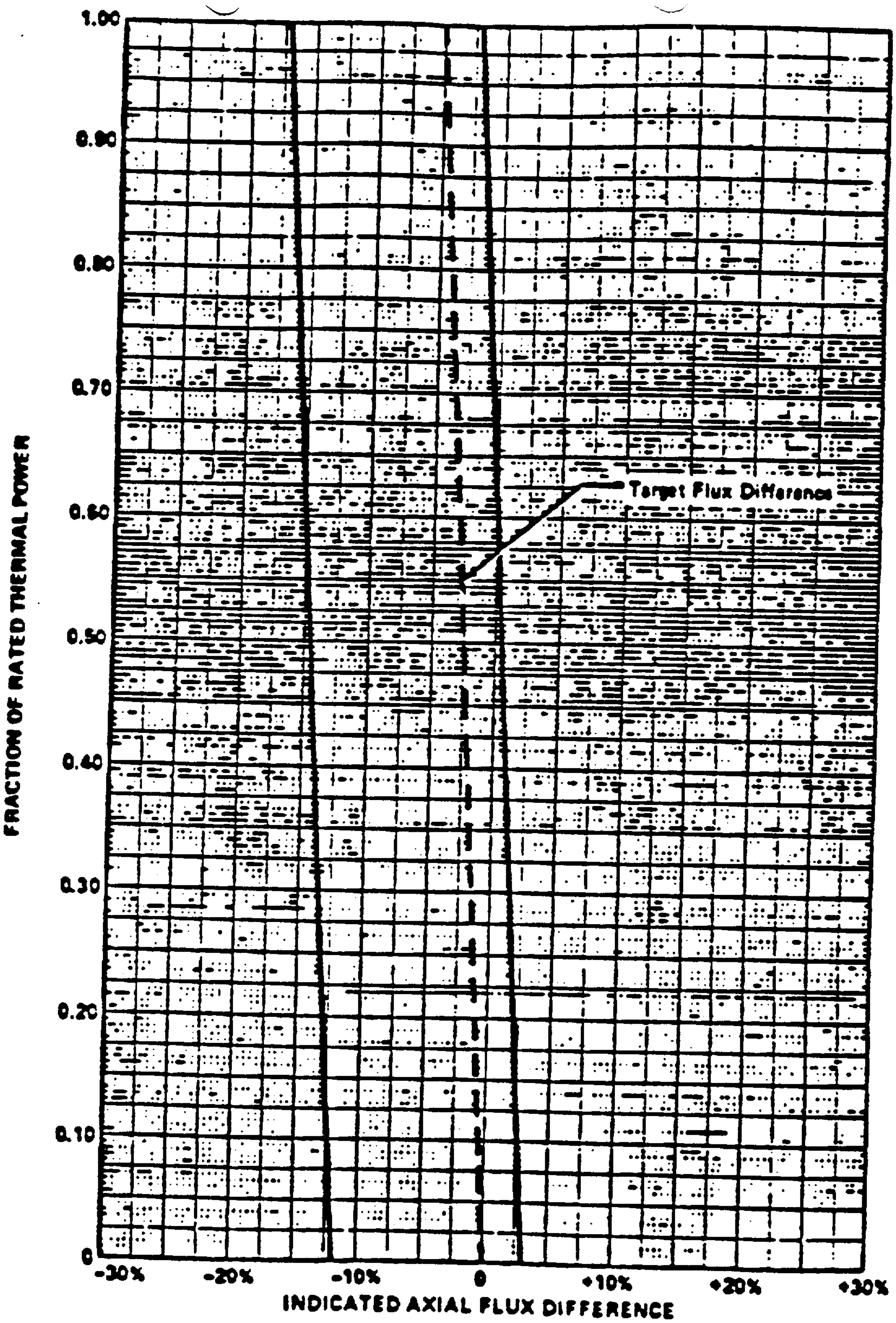


FIGURE B 3/4.2-1

TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

HEAT FLUX HOT CHANNEL FACTOR and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

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

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 combination of the RCS flow requirement (392,300 gpm) and the requirement on $F^N_{\Delta H}$ guarantees that the DNBR used in the safety analysis will be met. 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 $F^N_{\Delta H}$ is measured, no additional allowances are necessary prior to comparison with the limit. A measurement error of 4% for $F^N_{\Delta H}$ has been allowed for in the determination of the design DNBR value.

Fuel rod bowing reduces the value of DNB ratio. Margin has been maintained between the DNBR value used in the safety analyses and the design limit to offset the rod bow penalty and other penalties which may apply.

HEAT FLUX HOT CHANNEL FACTOR and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

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.

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_{xy}^{RTP}) as provided in the Core Operating Limits Report (COLR) per Specification 6.9.1.6 was determined from expected power control manuevers 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.

The 2-hour time allowance for operation with a tilt condition greater than 1.02 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 moveable 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 are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the

3/4.2.5 DNB PARAMETERS (Continued)

initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of greater than or equal to the design limit throughout each analyzed transient. The T_{avg} value of 598°F and the pressurizer pressure value of 2198 psig are analytical values. The readings from four channels will be averaged and then adjusted to account for measurement uncertainties before comparing with the required limit. The flow requirement (392,300 gpm) includes a measurement uncertainty of 2.8%.

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.

3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the dose guideline values of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, P_a (41.2 psig). As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J of 10 CFR Part 50.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that: (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 3.5 psig, and (2) the containment peak pressure does not exceed the design pressure of 56.5 psig during LOCA or steam line break conditions.

The maximum peak pressure expected to be obtained from a LOCA or steam line break event is 41.2 psig (P_a). The limit of 0.3 psig for initial positive containment pressure will limit the total pressure to 41.2 psig, which is less than design pressure and is consistent with the safety analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the safety analysis for a LOCA or steam line break accident. Measurements shall be made by fixed instruments, prior to determining the average air temperature.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 41.2 psig (P_a) in the event of a LOCA or steam line break accident. The measurement of containment tendon lift-off force, the tensile tests of the tendon wires, the visual examination of tendons, anchorages and exposed interior and exterior surfaces of the containment, and the Type A leakage test are sufficient to demonstrate this capability.

The Surveillance Requirements for demonstrating the containment's structural integrity are in compliance with the recommendations of Regulatory Guide 1.35, "Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures," and proposed Regulatory Guide 1.35.1, "Determining Prestressing Forces for Inspection of Prestressed Concrete Containments," April 1979.

The required Special Reports from any engineering evaluation of containment abnormalities shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, the results of the engineering evaluation, and the corrective actions taken.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The 48-inch containment purge supply and exhaust isolation valves are required to be sealed closed during plant operations since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves sealed closed during plant operation ensures that excessive quantities of radioactive materials will not be released via the Containment Purge System. To provide assurance that these containment valves cannot be inadvertently opened, the valves are sealed closed in accordance with Standard Review Plan 6.2.4 which includes mechanical devices to seal or lock the valve closed, or prevents power from being supplied to the valve operator.

The use of the containment purge lines is restricted to the 18-inch purge supply and exhaust isolation valves since, unlike the 48-inch valves, the 18-inch valves are capable of closing during a LOCA or steam line break accident. There-

3/4.7 PLANT SYSTEMS

BASES

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line Code safety valves ensures that the Secondary System pressure will be limited to within 110% (1413.5 psig) of its design pressure of 1285 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a Turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is 20.65×10^6 lbs/h which is 122% of the total secondary steam flow of 16.94×10^6 lbs/h at 100% RATED THERMAL POWER. A minimum of two OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

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

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

Where:

SP = Reduced Reactor Trip Setpoint in percent of RATED THERMAL POWER,

V = Maximum number of inoperable safety valves per steam line,

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

X = Total relieving capacity of all safety valves per steam line in lbs/hour, and

Y = Maximum relieving capacity of any one safety valve in lbs/hour

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the Auxiliary Feedwater System ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss-of-offsite power.

Each auxiliary feedwater pump is capable of delivering a total feedwater flow of 500 gpm at a pressure of 1363 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the Residual Heat Removal System may be placed into operation. The AFW pumps are tested using the test line back to the AFST and the AFW isolation valves closed to prevent injection of cold water into the steam generators. The STPEGS isolation valves are active valves required to open on an AFW actuation signal. Specification 4.7.1.2.1 requires these valves to be verified in the correct position.

3/4.7.1.3 AUXILIARY FEEDWATER STORAGE TANK (AFST)

The OPERABILITY of the auxiliary feedwater storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 4 hours with steam discharge to the atmosphere concurrent with total loss-of-offsite power, main feedwater line break and failure of the AFW flow control valve followed by a cooldown to 350°F at 25°F per hour. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 SPECIFIC ACTIVITY

The limitations on Secondary Coolant System specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of a steam line rupture. This dose also includes the effects of a coincident 1 gpm primary-to-secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blow down in the event of a steam line rupture. This restriction is required to: (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the Surveillance Requirements are consistent with the assumptions used in the safety analyses.

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The Exclusion Area shall be as shown in Figure 5.1-1.

LOW POPULATION ZONE

5.1.2 The Low Population Zone shall be as shown in Figure 5.1-2.

MAP DEFINING UNRESTRICTED AREAS AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

5.1.3 Information regarding radioactive gaseous and liquid effluents, which will allow identification of structures and release points as well as definition of UNRESTRICTED AREAS within the SITE BOUNDARY that are accessible to MEMBERS OF THE PUBLIC, shall be as shown in Figures 5.1-3 and 5.1-4.

The UNRESTRICTED AREA boundary may coincide with the Exclusion (fenced) Area boundary, as defined in 10 CFR 100.3(a), but the UNRESTRICTED AREA does not includes areas over water bodies. The concept of UNRESTRICTED AREAS, established at or beyond the SITE BOUNDARY, is utilized in the Limiting Conditions for Operation to keep levels of radioactive materials in liquid and gaseous effluents as low as is reasonable achievable, pursuant to 10 CFR 50.36a.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The reactor containment building is a steel-lined, reinforced concrete building of cylindrical shape, with a dome roof and having the following design features:

a. Nominal inside diameter = 150 feet.

b. Nominal inside height = 241.25 feet.

c. Minimum thickness of concrete walls = 4 feet.

d. Minimum thickness of concrete roof = 3 feet.

e. Minimum thickness of concrete floor mat = 18 feet.

f. Nominal thickness of steel liner = 3/8 inches.

g. Net free volume = 3.38×10^6 - 3.41×10^6 cubic feet.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 56.5 psig and a structural temperature of 286°F.

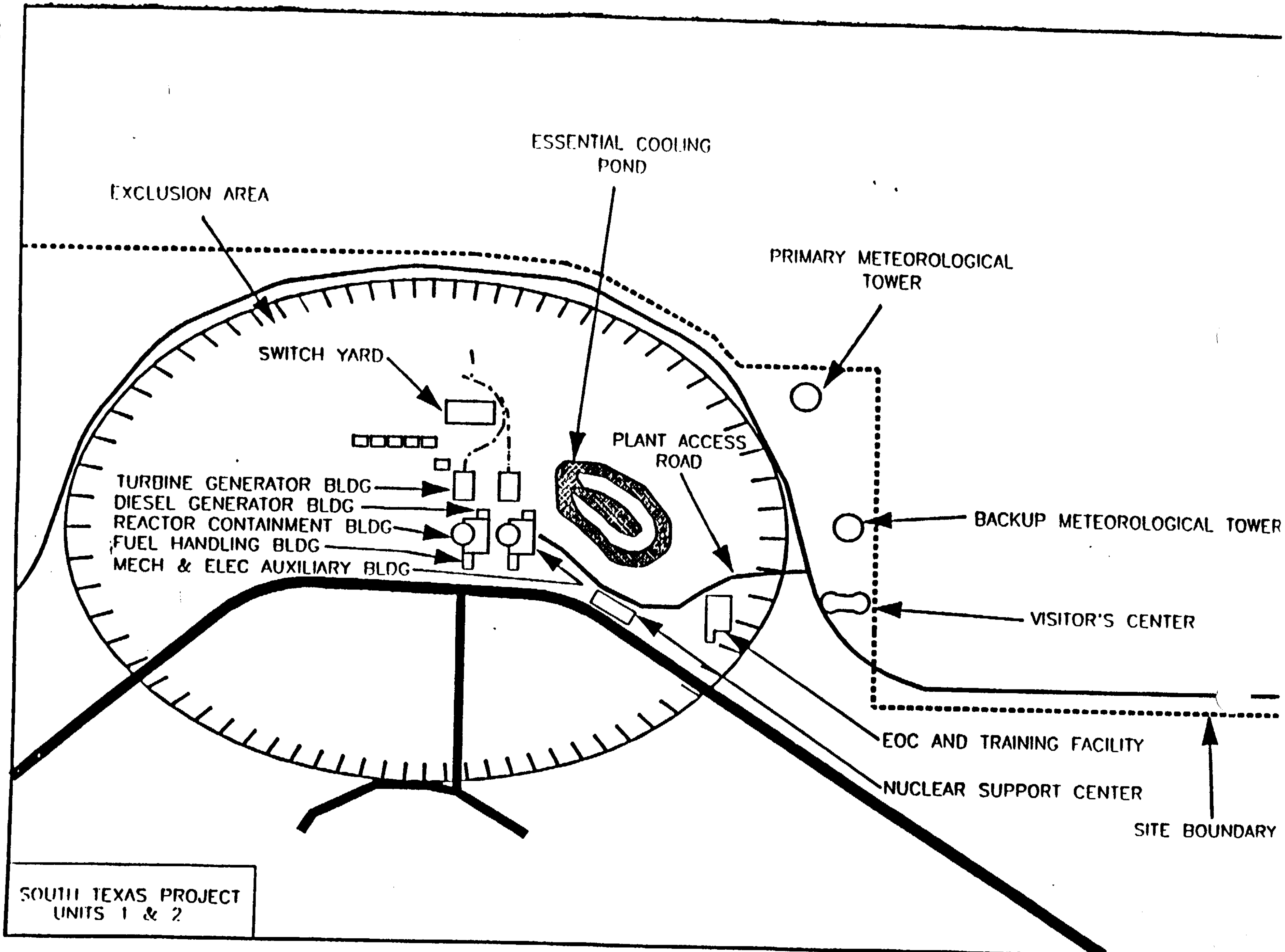
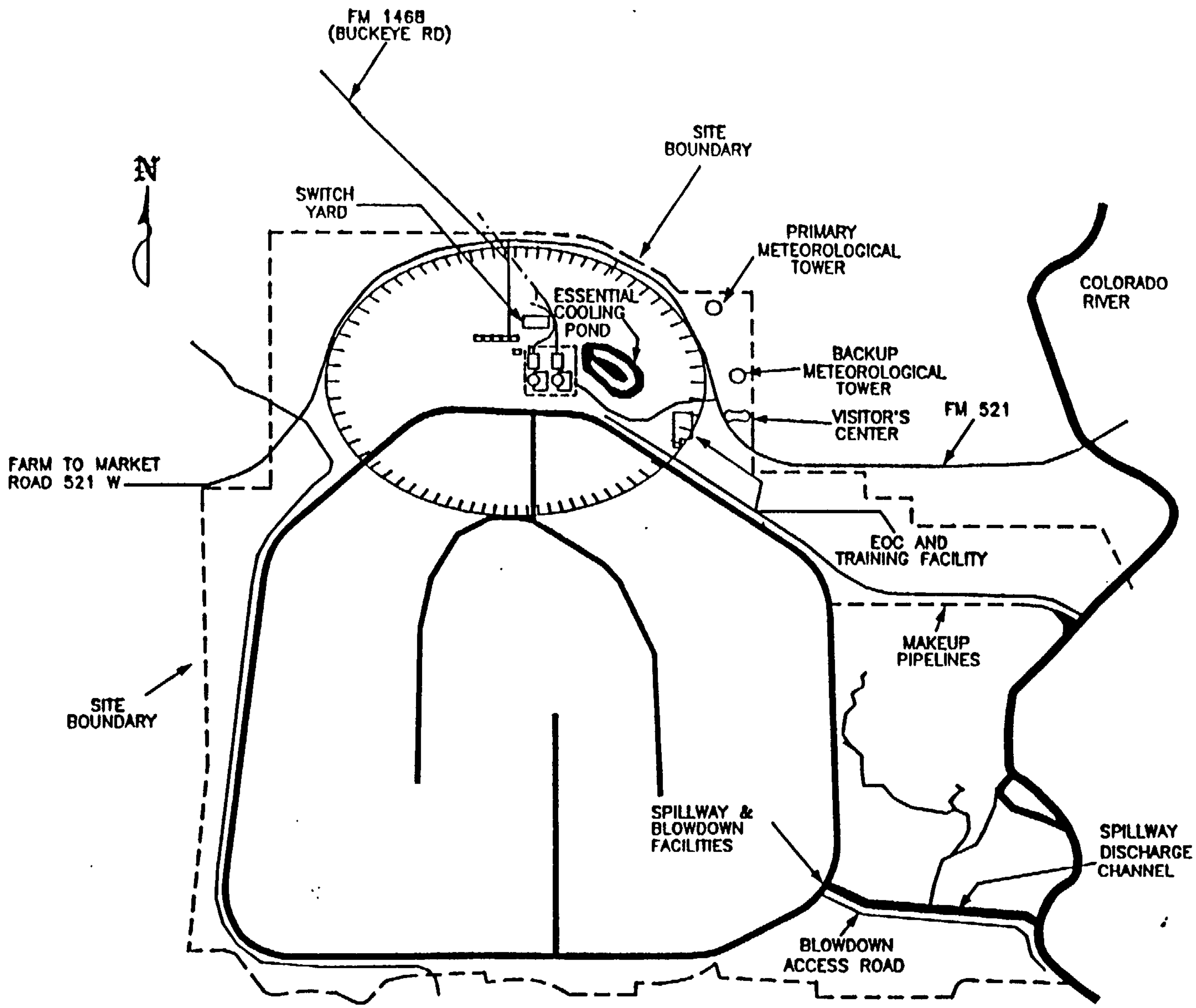


FIGURE 5.1-1
USION AREA



* The UNRESTRICTED AREA consists of the area beyond the SITE BOUNDARY.

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UNITS 1 & 2

FIGURE 5.1-4
UNRESTRICTED AREA* AND SITE BOUNDARY FOR RADIOACTIVE LIQUID EFFLUENTS

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 193 fuel assemblies with each fuel assembly containing 264 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 168 inches. The initial core loading shall have a maximum enrichment of 3.5 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 5.0 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 57 full-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 158.9 inches of absorber material. The absorber material within each assembly shall be silver-indium-cadmium or hafnium. Mixtures of hafnium and silver-indium-cadmium are not permitted within a bank. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 13,814 ± 100 cubic feet at a nominal T_{avg} of 561°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological towers shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

5.6.1 CRITICALITY

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water. This requirement shall be met by storing fuel in the spent fuel storage racks according to Specifications 5.6.1.3, 5.6.1.4, and 5.6.1.5. Additionally, credit may be taken for the presence of soluble boron in the spent fuel pool water, per Specification 3.9.13, to mitigate the misloading of one or more fuel assemblies, as described in Specification 5.6.1.6.

b. A nominal 10.95 inches center to center distance between fuel assemblies in Region 1 of the storage racks and a nominal 9.15 inches center to center distance between fuel assemblies in Region 2 of the storage racks.

c. Neutron absorber (Boraflex) installed between spent fuel assemblies in the storage racks in Region 1 and Region 2.

5.6.1.2 Prior to insertion into the spent fuel storage racks, each fuel assembly shall be categorized by reactivity, as discussed below, or be designated as a Category 1 fuel assembly. All fuel enrichment values are initial nominal uranium-235 enrichments. The reactivity categories are:

CATEGORY 1:
Fuel in Category 1 shall have an initial nominal enrichment of less than or equal to 5.0 w/o.

CATEGORY 2:
Fuel in Category 2 shall meet at least one of the following criteria:

1) a maximum initial nominal enrichment of 4.0 w/o; or,
2) a minimum burnup as shown on Figure 5.6-1; or,
3) a minimum number of Westinghouse Integral Fuel Burnable Absorber pins, as shown on Figure 5.6-2, or a K_{inf} of less than or equal to 1.445. The fuel assembly K_{inf} shall be based on a unit assembly configuration (infinite in the lateral and axial extent) in the reactor core geometry, assuming unborated water at 68°F.

The IFBA rod requirements shown in Figure 5.6-2 are based on a nominal IFBA linear B^{10} loading of 1.57 mg-B^{10}/inch. For higher IFBA linear B^{10} loadings, the required number of IFBA rods per assembly may be reduced by the ratio of the increased B^{10} loading to the nominal 1.57 mg-B^{10}/inch loading.

CATEGORY 3:
Fuel in Category 3 shall have the minimum assembly burnup shown on Figure 5.6-3.

CATEGORY 4:
Fuel in Category 4 shall have the minimum assembly burnup shown on Figure 5.6-4.

Data points for the curves presented in Figures 5.6-1 through 5.6-4 are presented in tables on the respective figures. Linear interpolation between table values may be used for intermediate points.

5.6.1.3 Region 1 racks may be used to store Category 1, 2, 3, and 4 fuel. Category 1 fuel shall be stored in a checkerboard pattern configuration with Category 3 or 4 fuel, alternating fuel assemblies as shown in Figure 5.6-5. Category 2, 3, and 4 fuel may be stored in a close packed arrangement.

Empty water cells may be substituted for fuel assemblies in all cases.

5.6.1.4 Region 2 racks may be used to store Category 1, 2, 3, and 4 fuel. Fuel in Categories 1, 2, and 3, shall be stored in a checkerboard pattern configuration alternating fuel assemblies with empty water cells in a 2 out of 4 pattern, as shown in Figure 5.6-6. Category 4 fuel may be stored either in a close packed arrangement or in the checkerboard pattern described above.

Empty water cells may be substituted for fuel assemblies in all cases.

5.6.1.5 Storage Configuration Interface Requirements. The transition schemes described below shall be used at the interface of two storage configuration areas in the spent fuel racks. Empty water cells may be substituted for fuel assemblies in all cases.

Internal Interfaces in Region 1 Racks

The interface between a closed packed fuel storage area in Region 1 and a checkerboarded storage area also in Region 1 shall be such that either:

1. Category 3 or 4 fuel assemblies in the checkerboard pattern are carried into the first row of the close packed storage area of fuel, as shown in Figure 5.6-5; or,

2. at least one row of empty water cells separate a close packed fuel storage area and a checkerboarded storage area.

Internal Interfaces in Region 2 Racks

The interface between a close packed fuel storage area in Region 2 and a checkerboarded storage area in Region 2 shall be such that either:

1. there is a one row carryover of alternating empty cells from the checkerboard area into the first row of the close packed area with the remaining cells of the row filled with Category 4 assemblies, as shown in Figure 5.6-6; or,

2. at least one empty row of cells separates the checkerboard pattern area and the close packed storage area.

Region 1 Close Packed Storage Area Adjacent to Region 2 Close Packed Area

There are no restrictions on the interface between Region 1 close packed storage areas and adjacent close packed storage areas in Region 2.

Region 1 Checkerboard Storage Area Adjacent to Region 2

The interface between a checkerboarded storage area in Region 1 and any Region 2 rack storage area shall be such that either:

1. the Region 1 checkerboard pattern is carried to the Region 1 boundary, but the last row at the Region 1 boundary leaves the Category 1 fuel assembly positions vacant; or,

2. at least one row of empty water cells in either Region 1 or Region 2 racks separate the Region 1 checkerboarded storage area and the Region 2 rack storage area.

Region 2 Checkboard Storage Area Adjacent to Region 1

The interface between a checkerboarded storage area in Region 2 and any Region 1 rack storage area shall be such that at least one row of empty water cells in either Region 1 or Region 2 racks separate the Region 2 checkerboarded storage area and the Region 1 rack storage area.

If checkerboarded storage areas in both Regions 1 and 2 are adjacent, at least one row of empty water cells in either Region 1 or Region 2 racks shall separate the checkerboarded storage areas in the respective racks.

5.6.1.6 The minimum boron concentration specified by Specification 3.9.13, "Spent Fuel Pool Minimum Boron Concentration" assures that the rack K_{eff} limit in Specification 5.6.1.1.a will not be violated under the following scenarios:

1. in Region 1, any misloading of Category 1, 2, 3, and 4 assemblies; or,

2. in Region 2, the misloading of one Category 1 assembly into the center of a fully loaded checkerboard area also containing Category 1 assemblies; or,

3. the misloading of a Category 1 assembly in a Region 1 rack adjacent to a Category 1 assembly in a Region 2 rack.

5.6.1.7 The new fuel storage racks are designed and shall be maintained with:

a. A K_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water and less than or equal to 0.98 when filled with aqueous foam moderation (low density water). This requirement shall be met by limiting the maximum fuel assembly nominal enrichments to 5.0 w/o or less.

b. A nominal 21 inches center to center distance between fuel assemblies.

5.6.1.8 The In-containment fuel storage racks are designed and shall be maintained with:

a. A K_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water. This requirement shall be met by satisfying at least one of the following criteria:

1) a maximum initial fuel assembly nominal enrichment to 4.5 w/o or less;

2) a minimum number of Westinghouse's Integral Fuel Burnable Absorbers, as a function of initial nominal assembly enrichment, as shown on Figure 5.6-7, or a K_{inf} of less than or equal to 1.484. The fuel assembly K_{inf} shall be based on a unit assembly configuration (infinite in the lateral and axial extent) in the reactor core geometry, assuming unborated water at 68°F.

The IFBA rod requirements shown in Figure 5.6-7 are based on a nominal IFBA linear B^{10} loading of 1.57 mg-B^{10}/inch. For higher IFBA linear B^{10} loadings, the required number of IFBA rods per assembly may be reduced by the ratio of the increased B^{10} loading to the nominal 1.57 mg-B^{10}/inch loading; or,

3) the fuel assembly is categorized as a Category 2, 3, or 4 assembly, per Specification 5.6.1.2.

b. A nominal 16 inches center to center distance between fuel assemblies.

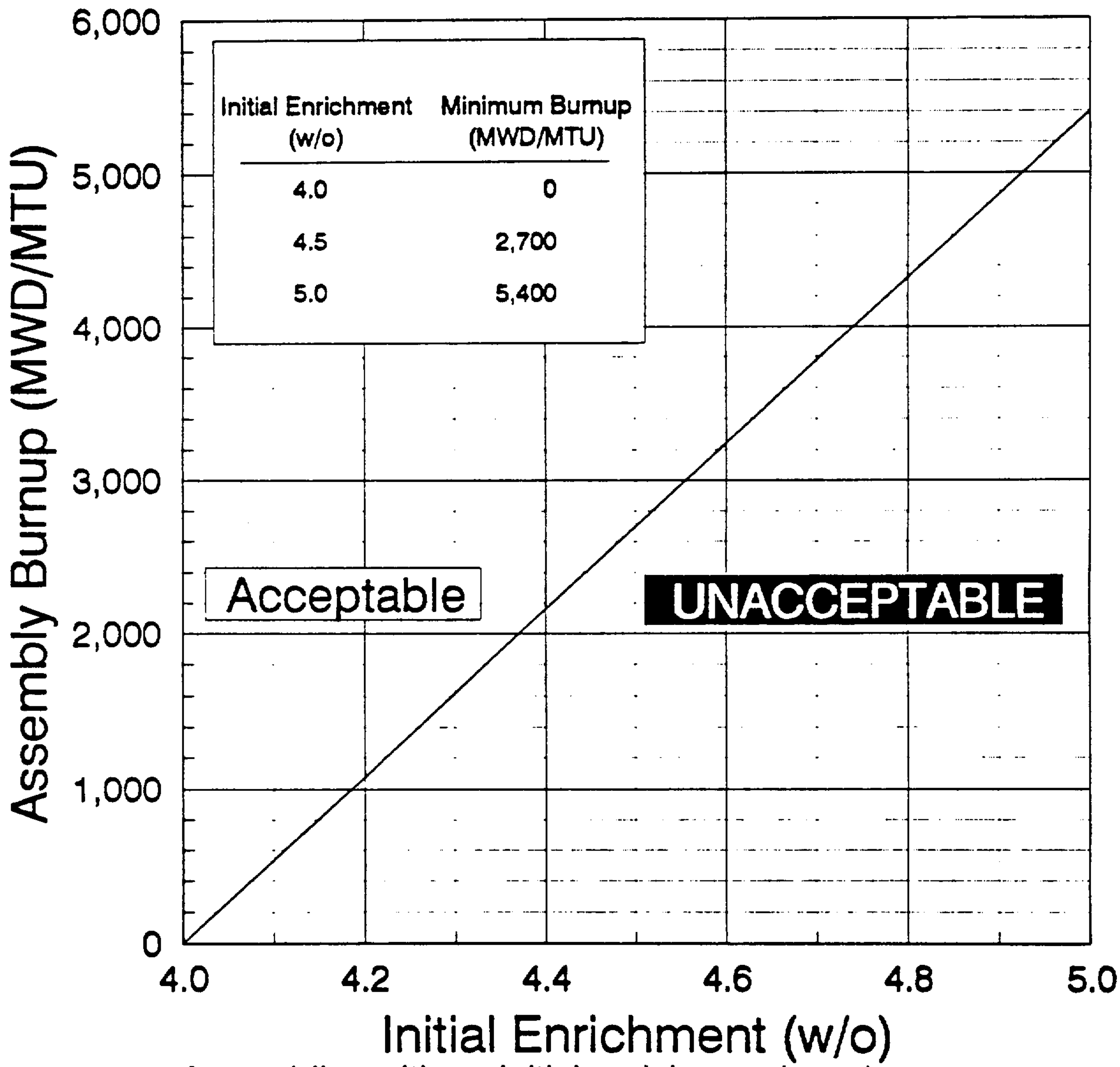
DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 62 feet-6 inches.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1969 fuel assemblies.

Minimum Burnup for Category 2 Fuel



Assemblies with an initial enrichment less than 4.0 w/o are ACCEPTABLE

Figure 5.6-1

Region 2 Close Packed and Checkerboard Fuel Storage

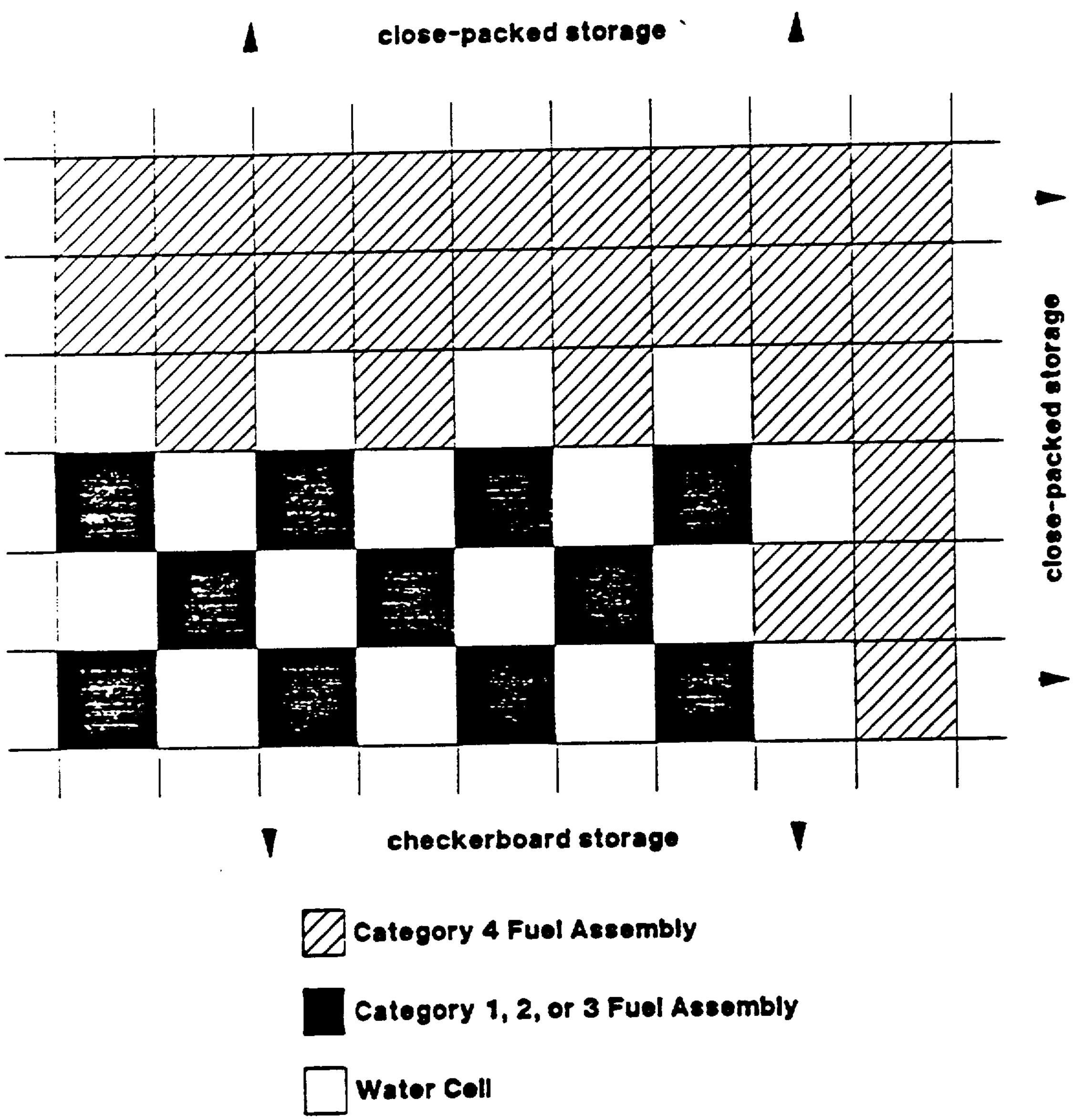
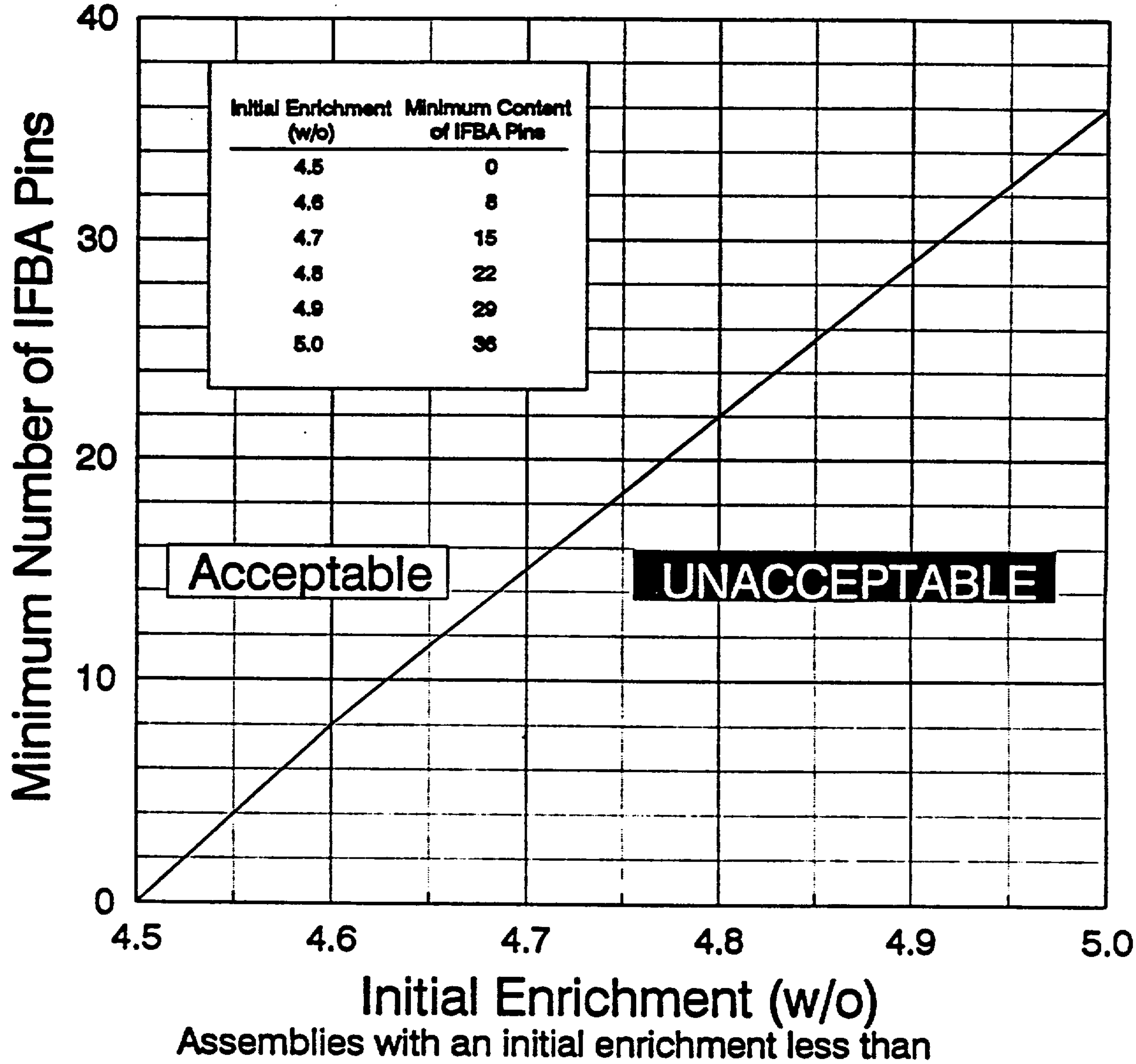


Figure 5.6-6

Minimum IFBA Content for In-Containment Rack Fuel Storage



Initial Enrichment (w/o)	Minimum Content of IFBA Pins
4.5	0
4.6	8
4.7	15
4.8	22
4.9	29
5.0	36

Assemblies with an initial enrichment less than 4.5 w/o are ACCEPTABLE

Figure 5.6-7

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components of the reactor coolant system are designed and shall be maintained within limits addressed in the Component Cyclic and Transient Limit Program as required by specification 6.8.3f.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 61 AND 50TO

FACILITY OPERATING LICENSE NOS. NPF-76 AND NPF-80

HOUSTON LIGHTING & POWER COMPANY

CITY PUBLIC SERVICE BOARD OF SAN ANTONIO

CENTRAL POWER AND LIGHT COMPANY

CITY OF AUSTIN, TEXAS

DOCKET NOS. 50-498 AND 50-499

SOUTH TEXAS PROJECT, UNITS 1 AND 2

1.0 INTRODUCTION

By letter dated May 27, 1993, as supplemented by letter dated April 18, 1994, Houston Lighting and Power Company (HL&P) proposed to amend the South Texas Project (STP) Units 1 and 2 Technical Specifications (TS) and Updated Final Safety Analysis Report (UFSAR) to upgrade the reload fuel assemblies to Westinghouse (W) VANTAGE 5 Hybrid (V5H) design. Currently, STP Units 1 and 2 utilize the Westinghouse 17x17XL standard (STD) fuel design for core reloads. This fuel is like W STD fuel except that it is longer (14 ft. long versus 12 ft. STD) to accommodate the longer STP core design. Fresh VANTAGE 5H fuel assemblies, manufactured to the STP 14 ft. length, will be used in each future reload until a full core loading of VANTAGE 5H fuel is achieved. The licensee also proposed to implement numerous safety analysis and operational margin improvements into the TS and UFSAR.

The first fuel loadings of VANTAGE 5H fuel are scheduled for South Texas Unit 1 Cycle 6 and South Texas Unit 2 Cycle 4. The safety analysis changes and associated setpoint changes will be implemented for both units during refueling outage 5 for Unit 1.

In addition to the proposed TS changes, HL&P submitted a safety evaluation report for the reload transition from the present XL STD fueled core to an all VANTAGE 5H fueled core. This report provided the results of the fuel, nuclear, thermal-hydraulic, and accident analyses which have been reviewed by the staff.

The April 18, 1994, letter revised the implementation date due to delays imposed by recently completed outages. The amendments will be fully implemented for both units upon completion of the fifth refueling outage for Unit 1.

2.0 EVALUATION

Background

The licensee has upgraded the fuel used in the South Texas Project units to the Westinghouse Vantage 5 Hybrid design in an effort to improve fuel economy and reduce the cobalt source term. In conjunction with the mechanical fuel upgrade, the licensee proposed the following changes which affect the safety analysis: an increased peaking factor allowance, a change to the RCS average temperature range, a revised thermal design procedure, a positive moderator coefficient, shutdown margin reduction, modified overtemperature and overpower delta T, 10 percent steam generator tube plugging, added tolerance for pressurizer and steam line safety relief valve drift, steamline break mass and energy release inside containment, increased fuel storage rack enrichment limit, and reduced auxiliary feedwater flow. By Amendment Nos. 51 (Unit 1); 40 (Unit 2) and Amendment Nos. 54 (Unit 1); 43 (Unit 2), the NRC had previously approved increases in the refueling water storage tank boron concentration, the accumulator boron concentration, and the boric acid storage tank volume to accommodate the change in fuel type.

The proposed amendment resolves the licensee's commitment to technical specification changes for Veritrak/Overtemperature delta T and the following Justifications for Continued Operation (JCO): JCO #920020 "Veritrak Transmitters," JCO #920698 "Containment System Response DBA", JCO #910393 "Pressurizer Safety Relief Valve Loop Seal Purge Time," and JCO #910049 "Steam Line Break Mass and Energy Release."

In addition to the changes to the technical specifications, the UFSAR and the Core Operating Limits Report (COLR) have been revised.

2.1 Fuel Mechanical Design

STP Units 1 and 2 are currently operating with Westinghouse 17x17 XL standard (STD) fuel. Beginning with Unit 1 Cycle 6 and Unit 2 Cycle 4, reload fuel will consist of the Westinghouse VANTAGE 5H fuel design eventually leading to an all VANTAGE 5H fueled core. The V5H fuel design is described in WCAP-10444-P-A, Addendum 2, which was approved for reference in a staff safety evaluation of November 1, 1988. Since then, V5H fuel has been approved for reload applications in numerous plants. The features of the VANTAGE 5H fuel design which differ from those of the current STP STD XL fuel design include the replacement of intermediate inconel structural grids with Zircaloy grids, and the use of integral fuel burnable absorbers (IFBA).

NRC Information Notice 93-82, "Recent Fuel and Core Performance Problems in Operating Reactors," pointed out that VANTAGE 5H fuel can be damaged by vibrational fretting wear caused by a flow condition adjacent to the core baffle. The fuel vendor, Westinghouse, proposed short-term and long-term corrective actions. The licensee informed the staff that the STP fuel design and core loading have adopted the Westinghouse recommendation of short-term corrective action to address the vibrational fretting wear problem. The staff considers that the licensee's corrective action is acceptable for STP.

The licensee analyzed stress, strain, rod internal pressure, fatigue, and rod bowing based on the approved methodologies for steady state and transient conditions. These analyses considered the longer fuel design of the STP core. The results showed that the VANTAGE 5H fuel performed satisfactorily. The staff considers these analyses adequate.

The licensee also analyzed the rod cluster control assemblies (RCCAs), control rod drive mechanisms (CRDMs), neutron source assemblies, burnable absorber assemblies, and thimble plug assemblies. The absorber materials used in the RCCAs are boron carbide pellets plus silver-indium-cadmium alloy. The burnable absorbers used are the Westinghouse designed wet annular burnable absorbers (WABAs). All the RCCAs and WABAs designs have been approved previously. Therefore, the staff concludes that the RCCAs, WABAs, and CRDMs are acceptable for STP.

Based on the approved mechanical methodologies, the staff concludes that the VANTAGE 5H fuel mechanical design for STP is acceptable.

2.2 Nuclear Design

The effects of the VANTAGE 5H fuel on the STP physics parameters as compared to the STD fuel are small and the STP spent fuel pool criticality analysis allows for the storage of VANTAGE 5H fuel assemblies. The nuclear design parameters characterizing the STP transition core have been computed by methods previously used and approved for Westinghouse reactors.

Included in the licensee's submittal is a proposal to increase the allowable fuel enrichment from 4.5 weight percent (w/o) uranium-235 to 5.0 w/o. Storage of spent fuel with the higher enrichment was discussed and approved in a staff safety evaluation of August 25, 1992 (Amendment Nos. 43 and 32). The current submittal provides additional discussion of new fuel racks and in-containment fuel storage racks. The acceptance criteria for criticality require the effective neutron multiplication factor, K_{eff}, in the fresh fuel storage rack to be less than or equal to 0.95 for fully flooded conditions or 0.98 under optimum moderation conditions, including uncertainties. For the in-containment fuel storage rack, K_{eff} must be maintained less than 0.95, including uncertainties, for all conditions.

The licensee's report shows that the acceptance criteria are met for STP fresh and in-containment fuel storage racks for the storage of all Westinghouse 17x17 fuel assemblies (including extra-length assemblies) with the following conditions and enrichment limits:

Fresh rack - Storage of fuel assemblies with nominal enrichments up to 5.0 w/o in any location. There are no requirements on position or IFBA for these assemblies.

In-Containment - Rack	Storage of fuel assemblies with nominal enrichments up to 4.5 w/o in any location. Fuel assemblies with enrichments above 4.5 w/o can also be stored, but each assembly must contain sufficient IFBA to satisfy the requirements shown in Figure 6 of Section 6 of the racks' criticality analysis included in the reference documents volume of the licensee's submittal.

The re-analysis of the reactivity effects of fuel storage in the fresh fuel and in-containment fuel storage racks was performed with the KENO Va Monte Carlo computer code with neutron cross sections generated by the AMPX code package from the 227 energy group ENDF/B-V library. Since the KENO Va code package does not have depletion capability, burnup analyses were performed with the two-dimensional transport theory code, PHOENIX. These codes are widely used for the analysis of fuel rack reactivity and burnup and have been benchmarked against results of numerous critical experiments. The staff concludes that the analysis methods used by the licensee are acceptable, and that the proposed storage rack provisions discussed above are acceptable.

Beginning with Cycle 6 of Unit 1, future cycles of operation for STP will use increased power peaking factors to increase nuclear design flexibility and allow loading patterns with reduced leakage which in turn will allow longer operating cycles without increasing vessel fluence. The maximum heat flux hot channel factor (F_Q) limit at rated thermal power (RTP) will increase from the current value of 2.50 for STD fuel to 2.7 for both fuels. The operational full power nuclear enthalpy rise hot channel factor ($F_{delta\ H}$) will increase from the current STD value of 1.46 to 1.49 for STD fuel and 1.55 for VANTAGE 5H fuel. The peaking factors assumed in the design and safety analyses are 2.7 F_Q, and 1.55 (STD) and 1.62 (V5H) $F_{delta\ H}$. The higher design values of $F_{delta\ H}$ account for analytical and surveillance uncertainties. The lower value for STD fuel is needed to comply with the local oxidation criterion in LOCA analyses. These increased limits on peaking factors continue to ensure that the design limits on peak local power density and minimum departure from nucleate boiling ratio (DNBR) are not exceeded during normal operation and anticipated operational occurrences (AOOs), and that the peak clad temperature will not exceed the emergency core cooling system (ECCS) acceptance criteria in the event of a LOCA, as discussed in Section 2.4.

The reduced thimble size of the VANTAGE 5H fuel design could affect the control rod scram time. The drop time is measured from the beginning of decay of stationary gripper coil voltage to dashpot entry. The effect of this increase on the STP safety analyses has been considered and it was determined that the 2.8 second rod drop time assumed in existing analyses remains bounding. The licensee will verify this conclusion in startup tests. The staff finds this acceptable.

Based on its review, the staff concludes that approved methods have been used and that the nuclear design parameters meet applicable criteria and are supported by design bases safety analyses discussed in Section 2.4 of this safety evaluation. Therefore, the proposed nuclear design and the analytical methods used are acceptable.

2.3 Thermal-Hydraulic Design

The thermal-hydraulic analysis, DNB performance, and hydraulic compatibility during the transition from a mixed VANTAGE 5H-STD fueled core to an all VANTAGE 5H core incorporate the WRB-1 and W-3 DNB correlations, the revised thermal design procedure (RTDP), and an improved THINC IV modeling. Each of these has been reviewed and approved by the NRC. For the WRB-1 DNB correlation, the NRC has approved a 95/95 DNBR limit of 1.17 for the 17x17 STD fuel assemblies. To account for uncertainties associated with rod bow, a flow anomaly associated with reactor coolant system (RCS) and nuclear instrumentation system parameters, safety analysis DNBR limits of 1.43 and 1.38 were used for typical cells and thimble cells, respectively. The W-3 DNB correlation, with a 95/95 DNBR limit of 1.30, and the standard thermal design procedure (STDP) thermal-hydraulic methods are still used when conditions are outside of the range of the WRB-1 DNB correlation and of the RTDP. These correlations are used for both STD and V5H fuel designs because of the thermal and hydraulic compatibility of these fuel types as demonstrated in WCAP-10444-P-A. Also, the W-3 correlation with a 95/95 DNBR limit of 1.45 is used for steam line break analyses in the pressure range of 500 to 1000 psia since this range is below the range of the primary DNB correlations.

The licensee has indicated that a rod bow penalty of less than 1.0 percent is applicable to the 17x17 XL STD and V5H fuel. There is not a DNBR penalty associated with mixed cores for these fuels. The licensee indicated that the DNBR penalty related to the flow anomaly is about 3.6 percent, and would be accommodated by the margin between the design limit for the STD and V5H fuels and the analysis limit. The licensee has indicated that the margin is also intended to accommodate DNBR penalties that may occur in the future, and to provide flexibility in the design and operation of the plant.

The staff concludes that the rod bow, flow anomaly, and transition core penalties are adequately covered by the margin maintained between the design and safety limit DNBR values. Maintenance of adequate DNBR margin to cover DNBR penalties is confirmed by the licensee on a cycle-specific basis during the reload safety evaluation process.

The licensee's submittal included WCAP-11273, Rev. 2, "Westinghouse Setpoint Methodology for Protection Systems South Texas Project Units 1 and 2." This STP-specific report describes the application of the Westinghouse setpoint methodology which has been used in applications to other operating plants, including the V. C. Summer plant, which was referenced in the report. In Section 2.5 of this safety evaluation, we conclude that this setpoint methodology is also applicable to the South Texas units.

The thermal-hydraulic evaluation of STP with VANTAGE 5H fuel as well as the evaluation of VANTAGE 5H demonstration assemblies in various operating reactors have shown that 17x17 XL STD and VANTAGE 5H fuel assemblies are hydraulically compatible, and that sufficient DNBR margin exists in the safety limit DNBR to cover any rod bow and transition core penalties.

Approved methodologies were used and all thermal-hydraulic design criteria were satisfied. Therefore, the staff finds the thermal-hydraulic design of the STP transition STD/V5H and final VANTAGE 5H cores acceptable.

2.4 Transient and Accident Analyses

The impact on the plant safety analyses of the transition from Westinghouse STD fuel to Westinghouse VANTAGE 5H fuel as well as other changes which represent a departure from those currently used for STP has been reviewed by the licensee to determine which events need to be re-analyzed. The review was based on event-specific sensitivities, and a decision was made for each transient with regard to the need for a formal analysis as opposed to simply evaluating the impact of the subject features and assumptions. Events were reanalyzed in accordance with methods described in the Westinghouse reload methodology report, WCAP-9272-P-A.

A nominal core thermal power of 3800 MWt was assumed. The safety evaluations also assumed 10 percent steam generator tube plugging, and were performed at a thermal design flow of 95,400 gpm per loop, which conservatively bounds the licensing minimum measured flow of 98,075 gpm per loop. No one steam generator was assumed to exceed 10 percent tube plugging. The analyses also account for added tolerance for pressurizer safety valve setpoint drift and loop seal purge time, and a reduced steam-driven and motor-driven auxiliary feedwater pump surveillance flow requirement of 500 gpm.

The RTDP methodology discussed above was used to define the initial conditions for those re-analyzed accidents which have DNB as a limiting criterion, and are initiated at or near full power conditions to demonstrate that the DNB design basis is met. The other reanalyzed accidents used the standard thermal design procedure (STDP) to obtain initial conditions by adding the maximum steady-state errors to nominal values. The NRC requires a review of the temperature, pressure, power, and flow uncertainties used in the safety evaluations when using the RTDP. For STP, the uncertainties have been calculated based on plant procedures for instrument calibration, heat balance calculations, and RCS flow measurement.

The staff has reviewed the accidents which were re-analyzed or re-evaluated. These re-analyses applied methods which have been previously found acceptable by the staff. The results, which include transition core effects, show changes in the consequences of transients and accidents previously analyzed. However, the results remain within the required acceptance criteria. Specifically, for non-LOCA events, during normal operation and anticipated operational occurrences, there is at least a 95 percent probability at a 95 percent confidence level (95/95 probability/confidence) that DNB will not occur on the limiting fuel rod. During these operational modes, there is also a 95/95 probability/confidence that the peak kw/ft fuel rods will not exceed the melting temperature of UO_2, taken as 4900°F (unirradiated) and 4800°F at end of life. For these events, peak RCS pressure does not exceed the acceptance criterion of 110 percent of the 2500 psia design pressure.

The submitted discussion proposes that the STP design basis limiting RCS peak pressure criterion for locked rotor events be changed from the current basis of 110 percent of design pressure (2750 psi) to faulted stress limits (about 2900 psi). The locked rotor event analyses submitted in support of proposed TS changes acceptably meet the current locked rotor RCS pressure criterion of 110 percent of design pressure, and take credit for delayed loss-of-offsite power. This indicates to the staff that there is no need for the proposed change in design basis pressure criterion. Furthermore, the staff considers acceptance criteria for accident analyses to be generic positions which the staff has historically supported, and are not within the scope of this review. The licensee's submittal does not provide justification for a plant specific exception to the staff generic position. Consequently, the staff does not accept this plant specific proposal and continues to evaluate STP locked rotor event analyses by its current RCS pressure criterion.

The maximum average fuel pellet enthalpy was less than 225 cal/gm (unirradiated) and 200 cal/gm (irradiated) for all control rod ejection events, thus meeting the NRC criterion of less than 280 cal/gm.

The radiological consequences of those accidents reanalyzed to reflect an increase in fuel burnup to 60,000 MWD/MTU associated with the use of VANTAGE 5H fuel, and the other changes being implemented in STP, such as the increased peaking factors and increased BOC MTC, are not significantly changed by the current reference analyses.

The large break LOCA analysis for STP 1 and 2, applicable to a full core of VANTAGE 5 fuel assemblies, was performed to develop specific peaking factor limits. The large break LOCA analyses assumed that the reactor was running at 3876 MWt (102 percent of rated power) with a total peaking factor (F_Q) of 2.7, a hot channel enthalpy rise factor ($F_{delta-H}$) of 1.62 (1.55 for once-burned STD), RCS flow of 95,400 gpm per loop, and hot leg temperature (T_{hot}) of 625.6 °F. The approved Westinghouse 1981 ECCS evaluation model with BASH was used and a spectrum of cold leg breaks was analyzed. The worst case peak clad temperature (PCT) was 2177 °F for a double-ended cold leg guillotine (DECLG) break with a discharge coefficient (C_d) of 0.6. The analysis assumed both maximum containment safeguards (lowest containment pressure) and maximum ECCS safeguards ("no failure" single failure). The maximum local Zirconium/water reaction of 15.39 percent was calculated for a different case, assuming DECLG break with a C_d of 0.8 and minimum low pressure safety injection (failure of one safety injection train). This was explained by differences in the limiting assumptions (single failure, etc.) between the cases, the high peaking factors, and the long STP core, resulting in differing calculations of burst node location versus the PCT node. The calculated maximum core-wide Zirconium/water reaction rate was less than 1 percent for all cases analyzed.

Therefore, the results demonstrated that the PCT acceptance criterion of 2200°F as well as the criteria related to clad oxidation and maximum hydrogen generation contained in 10 CFR 50.46 continue to be met. In addition, the core remains amenable to cooling during and after the LOCA, and will be maintained in a shutdown condition with borated water with no credit for control rod insertion.

The small break (SB) LOCA analyses were performed with the approved Westinghouse ECCS small break evaluation model using the NOTRUMP and LOCTA-IV codes. The analysis assumed a full core of VANTAGE 5H fuel to determine PCT for a spectrum of cold leg breaks. The small break LOCA analyses make the same assumptions regarding plant condition (RCS flow, reactor power, peaking factors, etc.) as listed above for large break LOCA analyses, with the failure of an emergency power train which results in loss of one complete train of ECCS components (including 2 auxiliary feedwater trains) identified as the most limiting single failure. The minimum delivered flow available to the RCS is based on this single failure. The results demonstrate that the remaining ECCS with 2 auxiliary feedwater trains provide sufficient core cooling to meet the acceptance criteria limits of 10 CFR 50.46 for the limiting SBLOCA, a 1.5-inch break. The calculated PCT for this case is 1816°F, the maximum calculated local Zirconium/water reaction is 5.96 percent (also for this case), and the calculated core-wide Zirconium/water reaction is less than 1 percent. These results also demonstrate that SB LOCA events meet the performance requirements of 10 CFR 50.46(b), and are not limiting.

Because the methodologies used to perform the transient and accident analyses supporting the proposed changes are applicable and approved methodologies, with clarifications as discussed above, the analyses are acceptable.

2.5 Setpoint Methodology

The staff reviewed the Westinghouse methodology for calculating instrument loop uncertainties and instrument trip setpoint as presented in Westinghouse documents, WCAP-11273 "Westinghouse Setpoint Methodology for Protection Systems," and WCAP-13411 "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology." On a continual basis, under contracts from the licensee, Westinghouse performs all periodic calculations and analyses for the revised thermal design procedures (RTDP) using this methodology for both units of STP. In addition, Westinghouse was contracted by the licensee to revise setpoint-calculations for all protection systems of both units of STP by incorporating this methodology.

For determining monitoring instrumentation errors, Westinghouse has taken an approach that the uncertainties can be described with random, normal, and two-sided probability distributions, and the sum of both sides is equal to the range for the parameter being monitored. The individual instrument error components for a channel-uncertainty are combined using the square root of the sum of the squares of those groups of components which are statistically independent. Those errors that are dependent are combined arithmetically into independent groups, which are then systematically combined. Channel uncertainties and uncertainties in trip settings, indications, and computer readouts were computed addressing the following attributes as applicable. These attributes included: process measurement accuracy (PMA), primary element accuracy (PEA), sensor calibration accuracy (SCA), sensor measurement and test equipment accuracy (SMTE), sensor pressure effects (SPE), sensor drift (SD), rack calibration accuracy (RCA), rack measurement and test equipment accuracy (RMTE), rack temperature effects (RTE), rack drift (RD), readout device accuracy (RDOUT), computer isolator drift (ID), analog to digital conversion accuracy (A/D), controller accuracy (CA), and environmental effects (EA). Also considered were: biases, allowances for process variable overshoot and/or undershoot, thermal inertia, deadbands, compensation for excessive thermal drift in Veritrack transmitters, and various coefficients and constants used for certain process conditions (the environment, material conditions and for the geometrical configurations used for instrument connections to the process.) Setpoint margin was calculated using Margin = TA - CSA equation; where TA is Total allowance (safety analysis limit - nominal trip setpoint), and CSA is channel statistical error allowance (Total of component uncertainties). Channel component uncertainties were grouped in groups such as: process allowances, sensor allowances, and rack allowances.

The relationship between the error components and the total error for a instrument channel was computed using following equation.

$$CSA = EA + \{(PMA)^2 + (PEA)^2 + (SCA+SMTE+SD)^2 + (STE)^2 + (SPE)^2 + (RCA+RMTE+RCSA+RD)^2 + (RTE)^2\}^{1/2} \pm \text{ Bias if any}$$

Process allowances are PMA and PEA are both considered independent. PMA includes the non-instrument related effects such as neutron flux, calorimetric power error assumptions, fluid density changes, and temperature stratification assumptions. PEA accounts for errors due to metering devices, such as elbows, venturi, and orifices.

Sensor allowances are SCA, SMTE, SD, STE, and SPE. SCA, SMTE and SD are considered interactive, and STE and SPE are considered independent. The procedures used for calibration and for determining instrument drift compare the instrument output to its known input. Thus, unless "AS LEFT/AS FOUND" data is recorded and tracked for some significant length of time for each component, it is impossible to determine differences between calibration errors, and the drift when the sensor is checked during calibration.

Rack allowances are RCA, RMTE, RCSA, RD and RTE. RCA, RMTE, RCSA and RD are considered interactive, and RTE is independent. Therefore, unless "AS LEFT/AS FOUND" data is recorded and tracked for a period of time for each component, it is impossible to determine differences between calibration errors, and the drift when the rack instrumentation is checked during calibration.

The Westinghouse methodology also considered the following:

a. Synergistic effects of aging, exposure of components to environment such as temperature, humidity, background radiation for all applicable uncertainty attributes.

b. Except for process measurement accuracy, rack drift, and sensor drift, all uncertainties assumed were extremes of the ranges. Therefore, the results were more conservative than using two sigma values. Rack drift and sensor drift were based on a survey of reported plant LERs.

c. During the life of the plant, insulation resistance (IR) of cable(s), terminations, and junctions will be degraded continuously due to synergistic effects of normal environment and/or due to an accident environment, and will introduce error in instruments. In the case that the value of such error was less than 0.1 percent of the span, Westinghouse considered it negligible, and it was omitted. Where its value exceeded 0.1 percent of span, it was considered as an environmental error. Westinghouse confirmed that for quantifying the error introduced due to IR changes, simulated aged cables and related terminations were used during testing. HL&P is aware of effects of IR degradation, and will keep track of value(s) of IR by performing periodic testing. In case the error due to IR degradation is found to exceed 0.1 percent, the affected calculation would be revised to account for the error.

d. Westinghouse calculations were based on a 1:1 ratio between sensor accuracy and accuracy of measurement and test equipment (M&TE), and on a 4:1 ratio between M&TE accuracy and accuracy of electronics (i.e., rack equipment). Westinghouse stated that these accuracy ratios were given to them by HL&P. The staff assumes that HL&P will keep track of the M&TE accuracies, and will maintain them throughout the life of the plant, or revise the setpoint calculations as necessary to address new accuracies of the M&TE.

As a result of the re-evaluation of uncertainties in instrument channels of safety systems, changes were proposed to: values of total allowance(s), Z(s), allowable value(s), sensor error(s), trip setpoint(s); terms used in the OTDT equation including time constants and constants K_1, K_2, and K_6; values of q_t and q_b (percent rated thermal power in the top and bottom halves of the core respectively); value of P_a used in containment leakage rate tests; primary containment average temperature for LCO; and the minimum flow value for surveillance tests of the auxiliary feedwater pump. The proposed changes also include a revision to TS Bases 2.2.1 for the minimum value of DNBR during steady state, and the value of shutdown margin in Bases 3/4.1.1, "Boration Control."

Based upon our review, the staff finds the Westinghouse Methodology used for determining the instrument channel uncertainties, trip setpoints and setpoint margins at STP acceptable.

2.6 Revised Maximum Containment Pressure and Temperature Response

The licensee has performed maximum containment pressure and temperature response reevaluation with the following changes to the safety analysis of that reviewed in the staff's safety evaluation that supported the issuance of the operating licenses (NUREG-0781):

a. Reduced containment free volume,
b. Reduced containment initial temperature,
c. Mass and energy release changes due to fuel upgrade, and effect of T_{hot} reduction.

The licensee indicated that as part of a probabilistic risk assessment model development effort, a review of pertinent calculations identified a mathematical error in the containment free volume calculation. Due to this error, the original calculation overestimated the containment free volume. The reduced containment free volume is 3.38×10^6 ft^3, including a -0.85 percent margin of error, a reduction of 5.1 percent. The original free volume was calculated as 3.56×10^6 ft^3.

The licensee has proposed to change the technical specification limit for the initial containment temperature from 120°F to 110°F.

The effect of the reduced containment free volume, reduced containment initial temperature, the mass and energy releases for the new fuel, and the T_{hot} reduction are evaluated below. The effects of these changes on the containment maximum temperature, containment maximum and minimum pressures, containment subcompartment analysis, containment safety-related equipment, containment leakage, and hydrogen generation were considered.

2.6.1 Containment Maximum Temperature

The licensee indicated that the containment maximum temperature occurs due to a design basis main steam line break (MSLB). The original MSLB mass and energy releases were calculated using the Westinghouse MARVEL code, and the containment temperature and pressure using the Bechtel COPATTA code. For the V5H fuel upgrade effort, the MSLB mass and energy release rates were re-calculated using the updated Westinghouse LOFTRAN code, and containment temperature and pressure were re-calculated using the Brookhaven CONTEMPT4 code. The LOFTRAN and CONTEMPT4 codes have been used at other plants for the above analyses, and the staff has found the use of these codes acceptable.

For the fuel upgrade effort, the MSLB analyses were reanalyzed using a reduced containment free volume of 3.38×10^6 ft^3, revised MSLB mass and energy releases, reduced containment initial temperature of 110°F, and updated passive heat sinks. The maximum containment temperature results from the design basis double-ended main steam line break coincident with one main steam isolation valve (MSIV) single active failure. The re-analysis of the MSLB increases the peak containment temperature from 323°F to 327°F.

The licensee stated that the containment structure is designed for 286°F. The revised design basis MSLB predicts that the peak temperature remains above 286°F for the first 110 seconds of the transient. During this brief period the heat transfer coefficient is not sufficiently high to result in heating the containment structures to the vapor temperature (327°F), and the structure design temperature of 286°F will remain bounding. The licensee also indicated that the containment safety-related equipment is qualified to operate in an accident environment with pressure and temperature equal to 57 psig and 340°F. Since the containment structure and safety-related equipment design temperatures remain bounding, the staff finds the proposed change in peak containment temperature acceptable.

2.6.2 Containment Maximum Pressure

The licensee indicated that the mass and energy release analyses for the V5H fuel upgrade were performed to conservatively maximize the mass and energy release available to the containment following a LOCA. The licensee has applied a multiplier of 1.0025 associated with the V5H fuel to the LOCA mass and energy release rates at time zero to the end of the blowdown portion of the transient. For the post-blowdown phase, this penalty is not required, since the currently listed releases remain bounding.

The net effect of the T_{hot} reduction is to increase the LOCA blowdown phase mass flowrate (during the first 25 seconds) by 2 percent, and decrease the energy releases by 0.6 percent. For the post-blowdown phase, the LOCA mass and energy releases remain unchanged. The licensee stated that based on the current Westinghouse models it is expected that these changes will have negligible effect on the long-term pressure transient results, and therefore, the long-term LOCA mass and energy releases due to T_{hot} will remain bounded by the existing design basis.

The maximum calculated peak containment pressure results from the current case of mass and energy releases of a double-ended pump suction guillotine (DEPSG) loss-of-coolant accident with maximum safety injection and minimum containment heat removal systems in operation. The re-evaluation increases peak containment pressure from 37.5 to 41.2 psig. Based on its review, the staff finds the proposed change acceptable since the peak containment pressure of 41.2 psig, calculated with approved methods, remains bounded by the containment design pressure of 56.5 psig.

2.6.3 Containment Minimum Pressure

The licensee indicated that the calculation for containment minimum pressure is not affected by either the containment free volume reduction, or the mass and energy release change due to the fuel upgrade. However, using an initial temperature of 110°F instead of 120°F, changes the minimum pressure from -3.5 psig to -2.9 psig. Therefore, the containment minimum design pressure of -3.5 psig (11.2 psia) is still applicable, and remains bounding. The staff finds the proposed change acceptable as the containment minimum design pressure of -3.5 psig remains bounding.

2.6.4 Containment Subcompartment Analysis

The licensee has indicated that it has analyzed the effects of reduced containment volume, reduced containment initial temperature, and the short term mass and energy releases due to fuel upgrade for the containment subcompartment analysis such as the pressurizer subcompartment, radioactive pipe chase subcompartment, regenerative heat exchanger subcompartment, RHR system valve room subcompartment, and steam generator loop compartments. The results of the analyses show that the original subcompartment design differential pressure remains bounding, and that the negligible change in peak differential pressure does not significantly affect the design margins or impact the structure design calculations. Based on the above results, the staff finds the proposed change acceptable, since it will not affect the subcompartment designs or the equipment located in them.

2.6.5 Containment Leakage

The licensee indicated that the Unit 1 containment was tested at 40.0 psig, and met the leakage criterion of the technical specifications. The Unit 2 containment was tested at 44.6 psig, and the leakage rate was also below the acceptance criterion of the technical specifications. The licensee is proposing to increase the peak containment pressure from 37.5 to 41.2 psig while maintaining the same leak rate in the technical specifications. Since Appendix J to 10 CFR Part 50 requires the licensee to perform leak testing at the peak accident pressure, 41.2 psig, and the technical specifications require the containment leakage limit to be satisfied, the staff finds the higher containment pressure with the present containment leakage limit to be acceptable.

2.6.6 Containment Hydrogen Generation

The licensee indicated that the revised analysis with reduction in containment free volume, reduced containment initial temperature, and changes due to fuel upgrade demonstrates that the requirements listed in Standard Review Plan Section 6.2.5 (including 10 CFR 50.44 and 10 CFR 50.46) continue to be met. The staff has reviewed this information and finds the proposed change for fuel upgrade acceptable.

2.6.7 Safety Injection/Containment Spray Operation

The licensee indicated that it has evaluated the effects of the reduced containment volume on safety injection and containment spray pump operation, and that the results indicate that the pumps are capable of providing required flow rates under increased containment pressure conditions. The staff has reviewed this information and finds the proposed change for fuel upgrade acceptable.

2.7 Technical Specification Changes

The specific changes proposed for the STP Technical Specifications are evaluated below.

(1) Figure 2.1-1, Reactor Core Safety Limits

The figure was revised to reflect the change of the limiting safety limits on the combination of the reactor thermal power, pressurizer pressure, and the highest operating loop coolant temperature. The changes reflect the DNB margin gained through use of the VANTAGE 5H IFM grid feature, the use of the improved THINC IV code, the WRB-1 DNB correlation, and RTDP. The limits are also reflected in the revised LOCA analyses for STP. Therefore, the limits given in TS Figure 2.1-1 are acceptable.

The Bases for TS 2.1.1 was also revised to describe the new DNB design basis methodology, and are acceptable.

(2) Table 2.2-1, Overtemperature Delta T (OTdT) and Overpower Delta T (OPdT) Trip Setpoints

The implementation of VANTAGE 5H fuel, the use of the RTDP, and the inclusion of parameters as determined by the Westinghouse setpoint methodology whose application is described in WCAP-11273, Rev. 2, cause the DNB core limits to change. These core limit changes result in OTdT and OPdT reactor trip setpoint changes. These setpoint changes are reflected in the STP safety analyses, which resulted in acceptable consequences, and are acceptable.

(3) Figure 3.1.2a, Moderator Temperature Coefficient

The licensee has proposed that the moderator temperature coefficient (MTC) limit specified in the STP TS 3.1.1.2/Figure 3.1.2a be revised for future core designs to permit a positive MTC. Several related safety analyses included in the submittal assumed a MTC of +5 pcm/F. The locked rotor analysis assumed a more limiting MTC of +5 pcm/F for powers up to 70 percent of rated thermal power and a linear ramp value from 70 percent power to 0 pcm/F at 100 percent power. The licensee proposes to incorporate the ramped MTC function assumed in the locked rotor analysis into the STP TS. The licensee is not proposing to implement this MTC revision at this time and the COLR MTC value remains 0 pcm/F. The staff finds the TS limit change acceptable because it is supported by related analyses.

(4) Figures 3.1-1 and 3.1-2, Required Shutdown Margin

The licensee has proposed to reduce the shutdown margin specified in TS 3.1.1.1/Figure 3.1-1 (covering operating Modes 1-4) and 3.1.1.2/Figure 3.1-2 (Mode 5) from 1.75 delta-K/K to 1.3 percent delta-K/K. These changes are supported by the results of affected analyses: main steam system depressurization; steamline break; feedline break; boron dilution events; and, post-LOCA shutdown. The results of these analyses indicate that design and

acceptance criteria will continue to be met assuming the reduced margin. The staff finds the proposed reduced shutdown margin is acceptable.
TS Bases 3/4.1.1 and 3/4.1.2 have been revised to reflect the reduced shutdown margin, and are acceptable.

(5) TS 3.2.5, DNB Parameters and associated Bases

The DNB-related parameters (RCS T_{avg}, pressurizer pressure, and RCS flow) and flow measurement uncertainty specified in TS 3.2.5 will be modified. The revised measured RCS average temperature range is 582.3 to 593.0 °F. Although the numerical value for maximum RCS T_{avg} (598°F) has not changed, its TS meaning has changed from an indicated value to an analytical value, the difference accounted for by measurement uncertainties. The revised pressurizer pressure is greater than 2189 psig. The revised RCS flow is greater than or equal to 392,300 gpm accounting for a flow measurement uncertainty of 2.8 percent with 10 percent steam generator tube plugging.

The DNB parameter changes reflect the use of the RTDP and implementation of the WCAP-11273, Rev. 2 setpoint methodology, and are supported by analyses in the submittal, as discussed above. The values used in the RTDP and transient and accident analyses conservatively bound these values. The staff finds the proposed changes, are acceptable.

The Bases associated with the above TS changes will be revised to reflect the proposed changes and are acceptable.

(6) Table 3.3-4, Engineered Safety Features Actuation System Instrumentation Trip Setpoints

Various trip setpoints and allowances have been changed based on the results of a revised Westinghouse reactor protection system setpoint study. The setpoint study implements the TS revisions to optimize trip setpoints, within the bounds of the safety analysis limits. The TS revisions to total allowance are direct results of uncertainty modifications. Changes to "Z" values are a direct result of PMA modifications. Allowable values and setpoints were changed to accommodate the modification in overall channel statistical allowance. The staff approved the methodology used in the setpoint study, as discussed in Section 2.5, and concluded that the revised values are acceptable.

(7) TS 3/4.6.1, Primary Containment

The containment maximum pressure specified in containment leak rate limits (TS 3.6.1.2.a), containment air lock leak rate limits (TS 3.6.1.2.b and TS 3.6.1.3.b), containment leak rate testing criteria (Surveillance Requirements 4.6.1.1.c and 4.6.1.2), and containment air lock leak rate testing criteria (Surveillance Requirement 4.6.1.3.b) is increased from 37.5 psig to 41.2 psig. The increase is based on a re-calculation of containment free volume, mass and energy release changes due to the fuel upgrade, containment initial temperature reduction, and T_{hot} reduction. The staff reviewed the licensee's analysis of the change in containment pressure as discussed in Section 2.6.2, and found it acceptable. Therefore, the related technical specification changes are acceptable.

(8) TS 3.6.1.5, Primary Containment Average Air Temperature

The maximum average containment air temperature specified in TS 3.6.1.5 is decreased from 120 degrees F to 110 degrees F. The limit was changed due to a reanalysis of main steam line break mass and energy releases for the V5H fuel. As discussed in Section 2.6, the licensee evaluated the effects of decreased initial temperature on containment maximum temperature and pressure, containment minimum pressure, hydrogen generation, and containment subcompartment analysis, and found that the results are bounded by the design. The staff found the licensee's evaluation acceptable, and this TS change is acceptable.

(9) TS 4.7.1.2.1, Auxiliary Feedwater System and associated Bases

The minimum flow of the motor-driven and steam-driven auxiliary feedwater pumps was reduced for surveillance requirements to 500 gpm. The new minimum flow value is reflected in the STP safety analyses, which resulted in acceptable consequences, and is, therefore, acceptable.

(10) TS 5.2.1.g, Containment Net Free Volume

The containment net free volume specified in TS 5.2.1.g is decreased from 3.56×10^6 ft^3 to 3.38×10^6 ft^3 due to a correction of the original calculation. Containment pressure and temperature response were re-analyzed using the corrected containment volume, and were found to remain bounded by the design. This change is acceptable.

(11) TS 5.3, Reactor Core

TS 5.3 was modified to reflect an increase in the maximum nominal enrichment for fuel assemblies from 4.5 weight percent (w/o) uranium-235 to 5.0 w/o. This change affects the criticality analyses for fuel storage racks, and increases the radiological source terms. As discussed in Section 2.2, the storage racks were re-analyzed, and the staff concluded that the acceptance criteria for criticality is met. The impact of the increased maximum enrichment on the radiological consequences of accidents is a slight increase in the doses reported in the UFSAR. However, the doses remain well within the acceptance limits. As discussed in Section 2.4, the radiological consequences of accidents were reanalyzed to consider the increased discharge burnup and were found not to be significantly changed. Therefore, the change to a maximum nominal enrichment of 5.0 w/o is acceptable.

(12) TS 5.6.1, Fuel Storage

TS 5.6.1.7, TS 5.6.1.8, and Figure 5.6-7 are added to describe the new fuel storage and in-containment storage rack requirements. The staff has concluded that the acceptance criteria are met, and storage provisions are acceptable for STP fresh and in-containment fuel storage racks for the storage of all Westinghouse 17x17 fuel assemblies as discussed in Section 2.2. The proposed change is acceptable.

3.0 CONCLUSIONS

The staff has reviewed the reports submitted to support the proposed STP TS changes for VANTAGE 5 fuel and concludes that appropriate material was submitted and that the fuel design, nuclear design, thermal-hydraulic design, instrument uncertainty and setpoint methodologies, containment building response, and transient and accident analyses are acceptable. The proposed TS changes suitably reflect the necessary modifications for operation of STP, and are adequately justified.

The staff will continue to evaluate RCS pressure response to a locked rotor event calculated for STP according to the criterion of 110 percent design pressure. Therefore, the licensee's proposal to amend the STP design basis by using faulted stress limits as a criterion for evaluation of locked rotor events is not accepted in this safety evaluation. As noted above, the analyses supporting the proposed TS changes - including locked rotor analyses - acceptably meet current criteria. Therefore, our present finding that the proposed RCS pressure criterion change for locked rotor events is not accepted in this safety evaluation does not alter our conclusions regarding the acceptability of the proposed TS changes.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Texas State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.33, an Environmental Assessment and finding of no significant impact was published in the Federal Register on May 25, 1994 (59 FR 27074). Accordingly, based upon the Environmental Assessment, the Commission has determined that issuance of these amendments will not have a significant effect on the quality of the human environment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and (3) issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

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