ATTACHMENT 4 TO AEP:NRC:1067

PROPOSED REVISED TECHNICAL SPECIFICATION PAGES

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

SITE BOUNDARY
1.36 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased or otherwise controlled by the licensee.

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

ALLOWABLE POWER LEVEL (APL)
1.38 APL means "allowable power level" which is that power level, less than or equal to 100% RATED THERMAL POWER, at which the plant may be operated to ensure that power distribution limits are satisfied.

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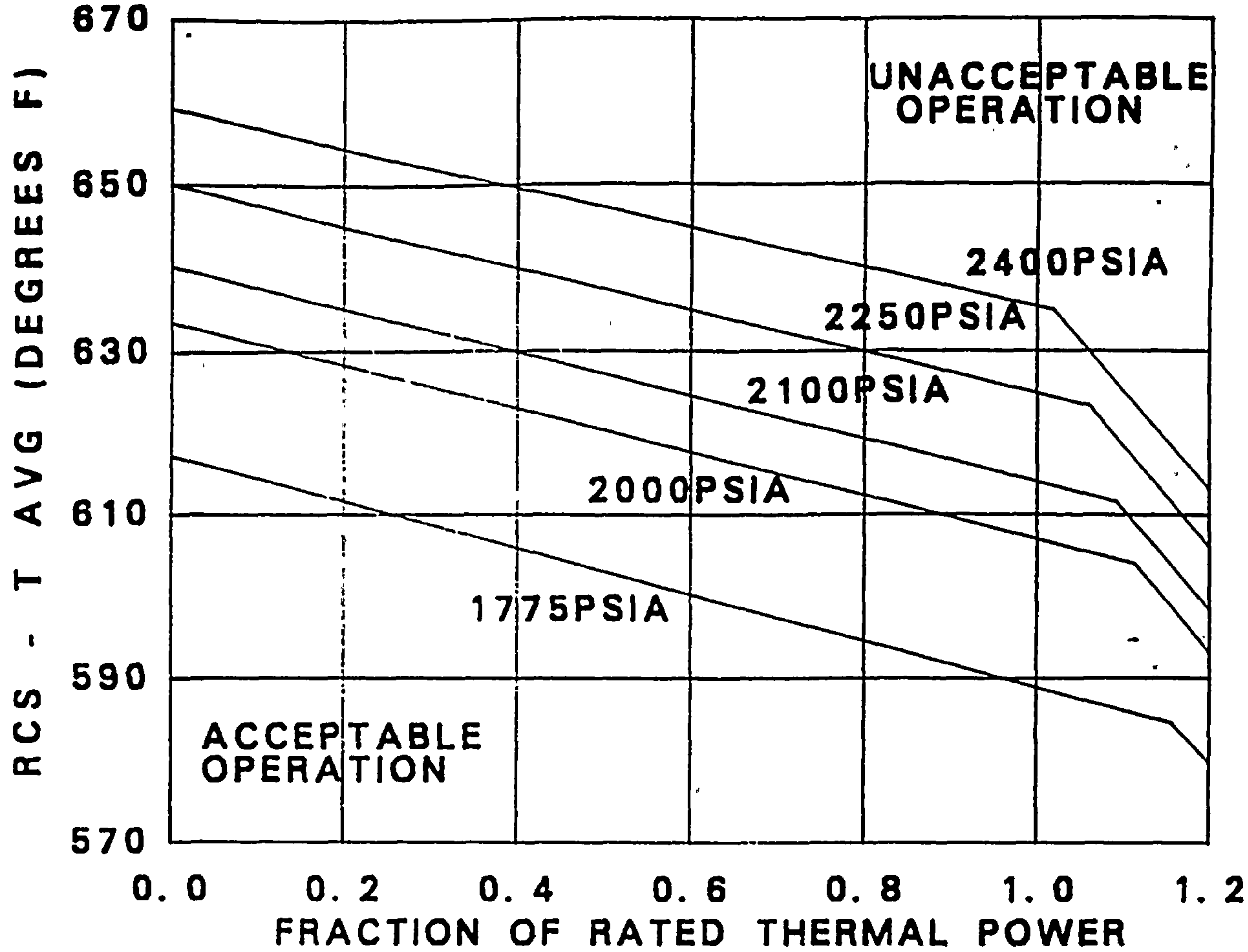

PRESSURE (PSIA)	BREAKPOINTS (FRACTION RATED THERMAL POWER, T AVG IN DEGREES F)
1775	(0. 0, 617. 1), (1. 16, 584. 5), (1. 20, 579. 7)
2000	(0. 0, 633. 5), (1. 11, 603. 9), (1. 20, 593. 1)
2100	(0. 0, 640. 3), (1. 09, 611. 5), (1. 20, 598. 3)
2250	(0. 0, 650. 0), (1. 06, 623. 2), (1. 20, 606. 0)
2400	(0. 0, 659. 0), (1. 02, 634. 8), (1. 20, 613. 0)

Figure 2. 1-1 Reactor Core Safety Limits
Four Loops In Operation

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Range, Neutron Flux	Low Setpoint - $\leq$ 25% of RATED THERMAL POWER High Setpoint $\leq$ 109% of RATED THERMAL POWER	Low Setpoint - $\leq$ 26% of RATED THERMAL POWER High Setpoint - $\leq$ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds
5. Intermediate Range, Neutron Flux	$\leq$ 25% of RATED THERMAL POWER	$\leq$ 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	$\leq 10^5$ counts per second	$\leq 1.3 \times 10^5$ counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 4
9. Pressurizer Pressure--Low	$\geq$ 1875 psig	$\geq$ 1865 psig
10. Pressurizer Pressure--High	$\leq$ 2385 psig	$\leq$ 2395 psig
11. Pressurizer Water Level--High	$\leq$ 92% of instrument span	$\leq$ 93% of instrument span
12. Loss of Flow	$\geq$ 90% of design flow per loop*	$\geq$ 89.1% of design flow per loop*

*Design flow is 91,600 gpm per loop.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

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

where: ΔT_o = Indicated Δ T at RATED THERMAL POWER

T = Average temperature, ^{o}F

T' = Indicated T_{avg} at RATED THERMAL POWER ($\leq 567.8^{o}F$)

P = Pressurizer pressure, psig

P' = Indicated RCS nominal operating pressure (2235 psig or 2085 psig)

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation

τ_1, τ_2 = Time constants utilized in the lead-lag controller for T_{avg} τ_1 = 22 secs. τ_2 = 4 secs.

S = Laplace transform operator

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATIONS (Continued)

Operation with 4 Loops

K_1 = 1.32

K_2 = 0.0230

K_3 = 0.00110

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) For $q_t - q_b$ between -37 percent and +2 percent, $f_1(\Delta I)=0$ (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) For each percent that the magnitude of $(q_t - q_b)$ exceeds -37 percent, the ΔT trip setpoint shall be automatically reduced by 0.33 percent of its value at RATED THERMAL POWER.

(iii) For each percent that the magnitude of $(q_t - q_b)$ exceeds +2 percent, the ΔT trip setpoint shall be automatically reduced by 2.17 percent of its value at RATED THERMAL POWER.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_o \left[K_4 - K_5 \left[\frac{\tau_3 S}{1+\tau_3 S}\right] T - K_6 (T-T") - f_2(\Delta I)\right]$

where:

ΔT_o = Indicated ΔT at RATED THERMAL POWER

T = Average temperature, ^oF

$T"$ = Indicated T_{avg} at RATED THERMAL POWER ($\leq 567.8^oF$)

K_4 = 1.083

K_5 = $0.0177/^oF$ for increasing average temperature and 0 for decreasing average temperature

K_6 = 0.0015 for $T > T"$; $K_6 = 0$ for $T \leq T"$

$\frac{\tau_3 S}{1+\tau_3 S}$ = The function generated by the rate lag controller for T_{avg} dynamic compensation

τ_3 = Time constant utilized in the rate lag controller for T_{avg} $\tau_3 = 10$ secs.

S = Laplace transform operator

$f_2(\Delta I)$ = 0

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 3.2 percent ΔT span.

Note 4: The channel's maximum trip point shall not exceed its computed trip point by more than 2.1 percent ΔT span.

HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

LIMITING CONDITION FOR OPERATION

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

$$F_Q(Z) \leq \frac{[2.15][K(Z)]}{P} \qquad P > 0.5$$

$$F_Q(Z) \leq [4.30]\ [K(Z)] \qquad P \leq 0.5$$

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

$F_Q(Z)$ is the measured hot channel factor including a 3% manufacturing tolerance uncertainty and a 5% measurement uncertainty.

K(Z) is the function obtained from Figure 3.2-3.

APPLICABILITY: MODE 1

ACTION:

With $F_Q(Z)$ exceeding its limit:

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

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

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

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

where: ΔT_o = Indicated ΔT at RATED THERMAL POWER

T = Average temperature, $^{\circ}F$

T' = Indicated T_{avg} at RATED THERMAL POWER ($\leq 567.8^{\circ}F$)

P = Pressurizer pressure, psig

P' = Indicated RCS nominal operating pressure (2235 psig or 2085 psig)

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation

τ_1, τ_2 = Time constants utilized in the lead-lag controller for T_{avg} τ_1 = 22 secs, τ_2 = 4 secs.

S = Laplace transform operator

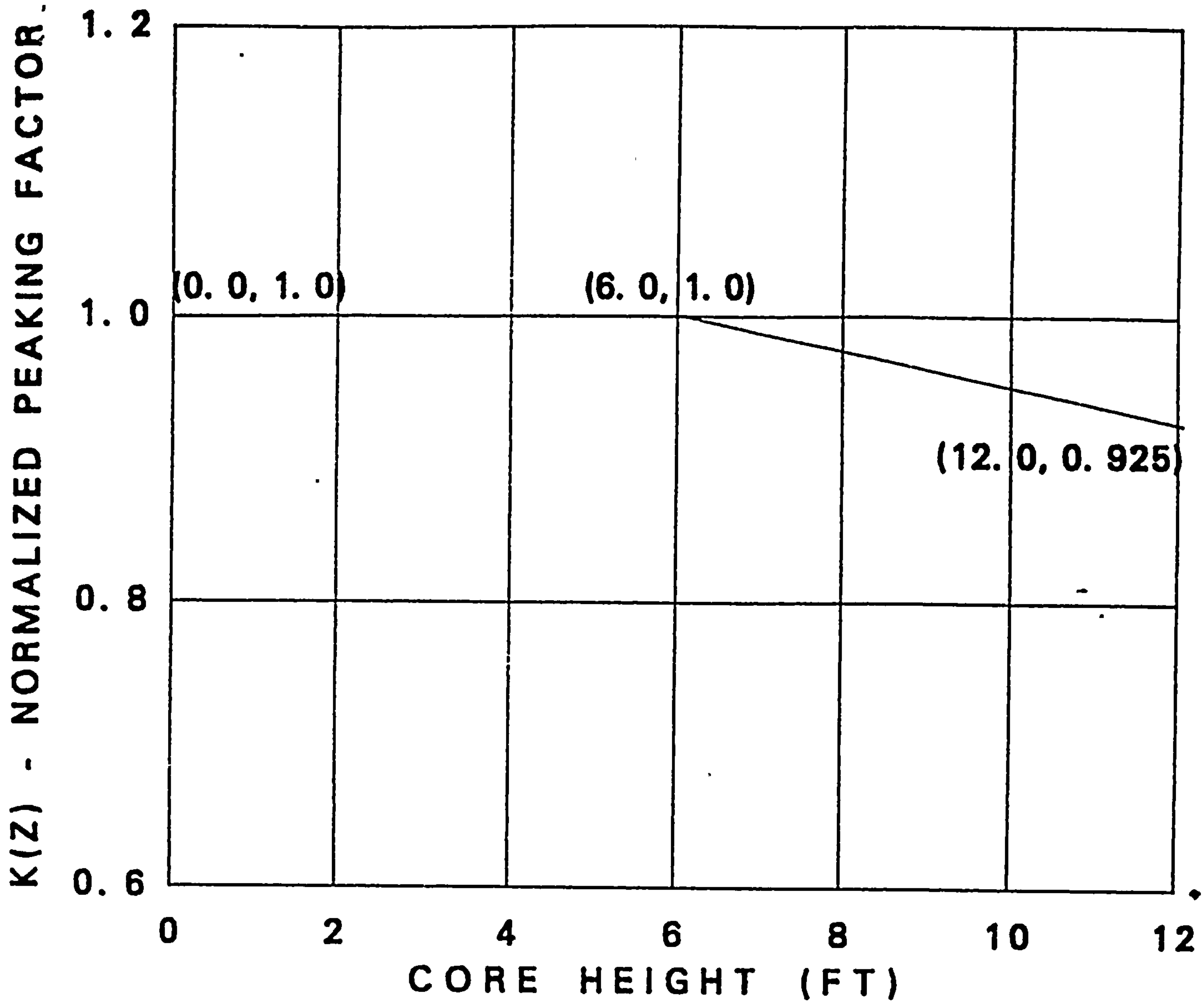

FIGURE 3. 2-3 K(Z) - Normalized F sub Q(z) as a function of Core Height

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

LIMITING CONDITION FOR OPERATION

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

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

Where P is the fraction of RATED THERMAL POWER

APPLICABILITY: MODE 1

ACTION:

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

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

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

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

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS 4 Loops in Operation at RATED THERMAL POWER
Reactor Coolant System Tavg	≤ 570.9°F*
Pressurizer Pressure	≥ 2050 psig**
Reactor Coolant System Total Flow Rate	≤ 366,400 gpm***

*Indicated average of at least three OPERABLE instrument loops.

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

***Indicated value.

ALLOWABLE POWER LEVEL - APL

LIMITING CONDITION FOR OPERATION

3.2.6 THERMAL POWER shall be less than or equal to ALLOWABLE POWER LEVEL (APL), given by the following relationship:

$$APL = \text{min over Z of } \frac{2.15\ K(Z)}{F_Q(Z) x V(Z) x F_p} \text{ x } 100\%, \text{ or } 100\%, \text{ whichever is less.}$$

- $F_Q(Z)$ is the measured hot channel factor including a 3% manufacturing tolerance uncertainty and a 5% measurement uncertainty.

- V(Z) is the function defined in the Peaking Factor Limit Report.

- F_p = 1.00 except when successive steady-state power distribution maps indicate an increase in max over Z of $\frac{F_Q(Z)}{K(Z)}$ with exposure. Then either of the penalties, F_p, shall be taken:

  F_p = 1.02, or

  F_p = 1.00 provided that Surveillance Requirement 4.2.6.2 is satisfied once per 7 Effective Full Power Days until two successive maps indicate that the max over Z of $\frac{F_Q(Z)}{K(Z)}$ is not increasing.

- The above limit is not applicable in the following core regions.

  1) Lower core region 0% to 10% inclusive.
  2) Upper core region 90% to 100% inclusive.

APPLICABILITY: MODE 1

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	NOT APPLICABLE
2. Power Range, Neutron Flux	
a. High Setpoint	$\leq$ 0.5 seconds*
b. Low Setpoint	$\leq$ 0.5 seconds*
3. Power Range, Neutron Flux, High Positive Rate	NOT APPLICABLE
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 0.5 seconds
5. Intermediate Range, Neutron Flux	NOT APPLICABLE
6. Source Range, Neutron Flux	NOT APPLICABLE
7. Overtemperature Delta T	$\leq$ 6.0 seconds*
8. Overpower Delta T	NOT APPLICABLE
9. Pressurizer Pressure--Low	$\leq$ 1.0 seconds
10. Pressurizer Pressure--High	$\leq$ 1.0 seconds
11. Pressurizer Water Level--High	$\leq$ 2.0 seconds

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION, TURBINE TRIP, FEEDWATER ISOLATION, AND MOTOR DRIVEN FEEDWATER PUMPS		
a. Manual Initiation	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High	$\leq$ 1.1 psig	$\leq$ 1.2 psig
d. Pressurizer Pressure--Low	$\geq$ 1815 psig	$\geq$ 1805 psig
e. Differential Pressure Between Steam Lines--High	$\leq$ 100 psi	$\leq$ 112 psi
f. Steam Flow in Two Steam Lines--High Coincident with T_{avg} --Low-Low or Steam Line Pressure--Low	$\leq 1.42 \times 10^6$ lbs/hr from 0% load to 20% load. Linear from 1.42×10^6 lbs/hr at 20% load to 3.88×10^6 lbs/hr at 100% load $T_{avg} \geq 541°F$ $\geq$ 500 psig steam line pressure	$\leq 1.56 \times 10^6$ lbs/hr from 0% load to 20% load. Linear from 1.56×10^6 lbs/hr at 20% load to 3.93×10^6 lbs/hr at 100% load. $T_{avg} \geq 539°F$ $\geq$ 480 psig steam line pressure

TABLE 3. . (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
2. Containment Radioactivity -- High Train A (VRS-1101, ERS-1301, ERS-1305)	See Table 3.3-6	Not Applicable
3. Containment Radioactivity -- High Train B (VRS-1201, ERS-1401, ERS-1405)	See Table 3.3-6	Not Applicable
4. STEAM LINE ISOLATION		
a. Manual	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure -- High-High	$\leq$ 2.9 psig	$\leq$ 3 psig
d. Steam Flow In Two Steam Lines -- High Coincident with T_{avg} Low-Low or Steam Line Pressure Low	1.42×10^6 lbs/hr from 0% load to 20% load. Linear from 1.42×10^6 lbs/hr at 20% load to 3.88×10^6 lbs/hr at 100% load. $T_{avg} \geq 541^{\circ}F$ $\geq$ 500 psig steam line pressure	1.56×10^6 lbs/hr from 0% load to 20% load. Linear from 1.56×10^6 lbs/hr at 20% load to 3.93×10^6 lbs/hr at 100% load. $T_{avg} \geq 539^{\circ}F$ $\geq$ 480 psig steam line pressure
5. TURBINE TRIP AND FEEDWATER ISOLATION		
a. Steam Generator Water Level -- High-High	$\leq$ 67% of narrow-range instrument span each steam generator	$\leq$ 68% of narrow-range instrument span each steam generator

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with a water volume less than or equal to 92% of span and at least 150 kW of pressurizer heaters.

APPLICABILITY: MODES 1,2, and 3.

ACTION:

With the pressurizer inoperable due to an inoperable emergency power supply to the pressurizer heaters either restore the inoperable emergency power supply within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours. With the pressurizer otherwise inoperable, be in at least HOT SHUTDOWN with the reactor trip breakers open within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.4.1 The pressurizer water volume shall be determined to be within its limits at least once per 12 hours.

4.4.4.2 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by transferring power from the normal to the emergency power supply and energizing the required capacity of heaters.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between 921 and 971 cubic feet,

c. A boron concentration of between 2400 ppm and 2600 ppm, and

d. A nitrogen cover-pressure of between 585 and 658 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within one hour or be in HOT SHUTDOWN within the next 8 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in HOT STANDBY within one hour and be in HOT SHUTDOWN within the next 8 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying the water level and nitrogen cover-pressure in the tanks, and

2. Verifying that each accumulator isolation valve is open.

* Pressurizer Pressure above 1000 psig.

d. At least once per 18 months by:*

   1. Verifying automatic isolation and interlock action of the RHR system from the Reactor Coolant System when the Reactor Coolant System pressure is above 600 psig.

   2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or abnormal corrosion.

e. At least once per 18 months, during shutdown, by:*

   1. Verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection test signal.

   2. Verifying that each of the following pumps start automatically upon receipt of a safety injection test signal:

      a) Centrifugal charging pump

      b) Safety injection pump

      c) Residual heat removal pump

f. By verifying that each of the following pumps develops the indicated discharge pressure on recirculation flow when tested pursuant to Specification 4.0.5 at least once per 31 days on a STAGGERED TEST BASIS.

1.	Centrifugal charging pump	$\geq$ 2405 psig
2.	Safety Injection pump	$\geq$ 1345 psig
3.	Residual heat removal pump	$\geq$ 165 psig

g. By verifying the correct position of each mechanical stop for the following Emergency Core Cooling System throttle valves:

   1. Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE.

*The provisions of Specification 4.0.6 are applicable.

2. At least once per 18 months.

	Boron Injection Throttle Valves		Safety Injection Throttle Valves
	Valve Number		Valve Number
1.	1-SI-141 L1	1.	1-SI-121 N
2.	1-SI-141 L2	2.	1-SI-121 S
3.	1-SI-141 L3		
4.	1-SI-141 L4		

h. By performing a flow balance test during shutdown following completion of modifications to the ECCS subsystem that alter the subsystem flow characteristics and verifying the following flow rates:

Boron Injection System Single Pump*	Safety Injection System Single Pump**
Loop 1 Boron Injection Flow 117.5 gpm	Loop 1 and 4 Cold Leg Flow $\geq$ 300 gpm
Loop 2 Boron Injection Flow 117.5 gpm	Loop 2 and 3 Cold Leg Flow $\geq$ 300 gpm
Loop 3 Boron Injection Flow 117.5 gpm	**Combined Loop 1,2,3 and 4 Cold Leg Flow (single pump) less than or equal to 640 gpm. Total SIS (single pump) flow, including miniflow, shall not exceed 700 gpm.
Loop 4 Boron Injection Flow 117.5 gpm	

*The flow rate in each Boron Injection (BI) line should be adjusted to provide 117.5 gpm (nominal) flow in each loop. Under these conditions there is zero miniflow and 80 gpm simulated RCP seal injection line flow. The actual flow in each BI line may deviate from the nominal so long as:

a) the difference between the highest and lowest flow is 10 gpm or less.

b) the total flow to the four branch lines does not exceed 470 gpm.

c) the minimum flow (total flow) through the three most conservative (lowest flow) branch lines must not be less than 345.8 gpm.

d) The charging pump discharge resistance ($2.31 x Pd/Qd^2$) must not be less than 4.73E-3 ft/gpm^2 and must not be greater than 5.33E-3 ft/gpm^2. (Pd is the pump discharge pressure at runout; Qd is the total pump flow rate.)

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two feedwater pumps, each capable of being powered from separate emergency busses, and

b. One feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two auxiliary feedwater pumps inoperable, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying that each motor driven pump develops an equivalent discharge pressure of $\geq$ 1375 psig at 60°F on recirculation flow.

2. Verifying that the steam turbine driven pump develops an equivalent discharge pressure of $\geq$ 1285 psig at 60°F and at a flow of $\geq$ 700 gpm when the secondary steam supply pressure is greater than 310 psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

[illegible]

2.1 SAFETY LIMITS

BASES

4 Loop Operation

Westinghouse Fuel
(15x15 OFA)

(WRB-1 Correlation)

	Typical Cell*	Thimble Cell**
Correlation Limit	1.17	1.17
Design Limit DNBR	1.33	1.32
Safety Analysis Limit DNBR	1.45	1.45

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

----

* represents typical fuel rod
** represents fuel rods near guide tube

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

$F^N_H = 1.49\ [1 + 0.3\ (1-P)]$

where P is the fraction of RATED THERMAL POWER

Note, do not include a 4% uncertainty value, since this measurement uncertainty has been included in the design DNBR limit values, which are listed in the bases for Section 2.1.1.

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

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

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

The Power Range Negative Rate Trip provides protection for control rod drop accidents. At high power, a rod drop accident could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate Trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate Trip for those control rod drop accidents for which the DNBR's will be greater than the applicable design limit DNBR value for each fuel type.

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor startup. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The source Range Channels will initiate a reactor trip at about 10^{-5} counts per second, unless manually blocked when P-6 becomes active. The Intermediate Range Channels will initiate a reactor trip at a current level proportional to approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

Overtemperature delta T

The Overtemperature delta T trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the High and Low Pressure reactor trips. This setpoint includes corrections for changes in density and heat capacity of water with temperature and dynamic compensation for piping delays from the core to the loop temperature detectors. The reference average temperature (T') and the reference operating pressure (P') are set equal to the full power indicated Tavg and the nominal RCS operating pressure, respectively, to ensure protection of the core limits and to preserve the actuation time of the Overtemperature delta T trip for the range of full power average temperatures assumed in the safety analyses. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table 2.2-1.

Overpower delta T

The Overpower delta T reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature delta T protection, and provides a backup to the High Neutron Flux trip. The setpoint includes corrections for changes in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The reference average temperature (T") is set equal to the full power indicated Tavg to ensure fuel integrity during overpower conditions for the range of full power average temperatures assumed in the safety analysis. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The High Pressure trip provides protection for a Loss of External Load event. The Low Pressure trip provides protection by tripping the reactor in the event of a loss of reactor coolant pressure.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief through the pressurizer safety valves. The pressurizer high water level trip precludes water relief for the Uncontrolled RCCA Withdrawal at Power event.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

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

The definitions of hot channel factors as used in these specifications are as follows:

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

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

3/4.2.1 AXIAL FLUX DEFFERENCE (AFD)

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

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 8 psig and 2) the containment peak pressure does not exceed the design pressure of 12 psig during LOCA conditions.

The maximum peak pressure resulting from a LOCA event is calculated to be 11.89 psig, which includes 0.3 psig for initial positive containment pressure.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that 1) the containment air mass is limited to an initial mass sufficiently low to prevent exceeding the design pressure during LOCA conditions and 2) the ambient air temperature does not exceed that temperature allowable for the continuous duty rating specified for equipment and instrumentation located within containment.

The containment pressure transient is sensitive to the initially contained air mass during a LOCA. The contained air mass increases with decreasing temperature. The lower temperature limit of 60°F will limit the peak pressure to 11.89 psig which is less than the containment design pressure of 12 psig. The upper temperature limit influences the peak accident temperature slightly during a LOCA; however, this limit is based primarily upon equipment protection and anticipated operating conditions. Both the upper and lower temperature limits are consistent with the parameters used in the accident analyses.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment steel vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that (1) the steel liner remains leak tight and (2) the concrete surrounding the steel liner remains capable of providing external missile protection for the steel liner and radiation shielding in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.