Attachment 2 to AEP:NRC:0916I

PROPOSED TECHNICAL SPECIFICATION CHANGES

INDEX

DEFINITIONS

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

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.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figure 2.1-1 for 4 loop operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

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.

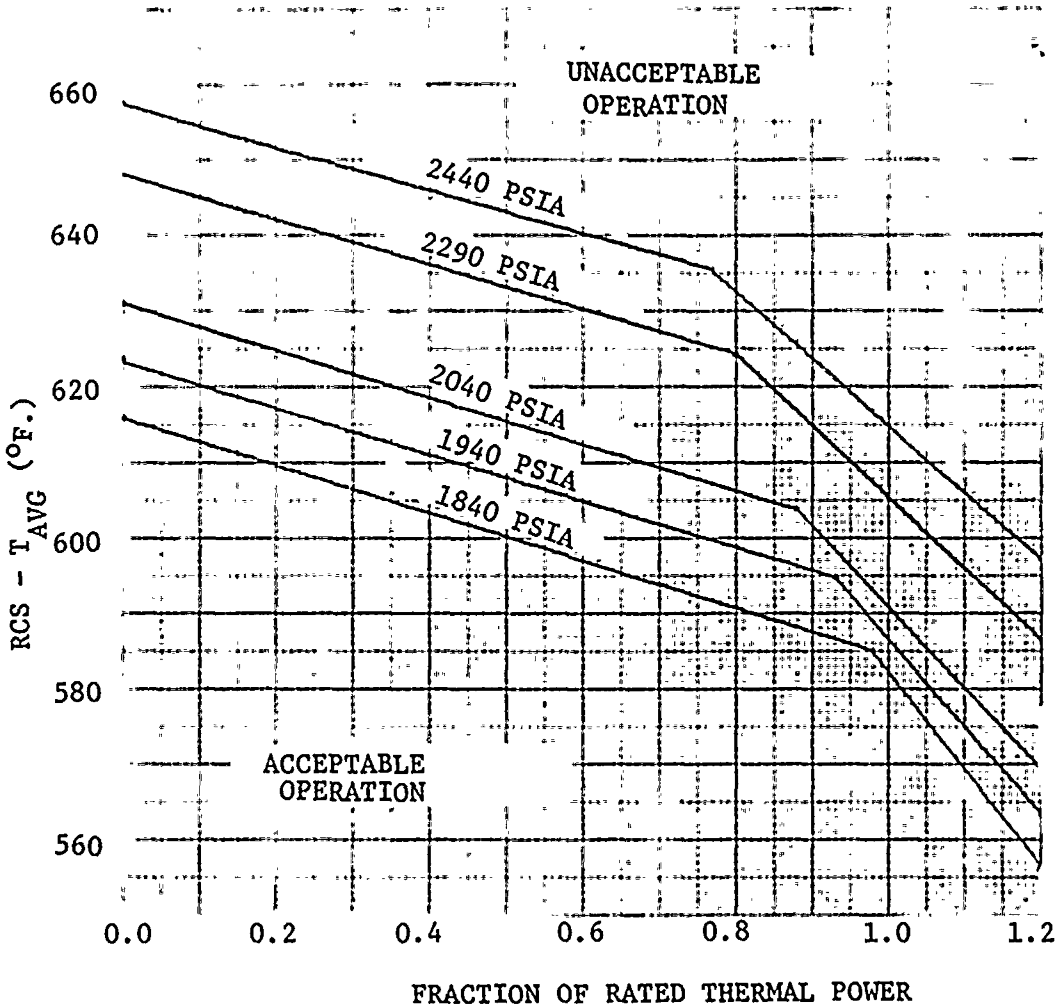
ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceede 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 5 minutes.



PRESSURE (PSIA)	BREAKPOINTS (FRACTION RATED THERMAL POWER, T_{AVG}, ^oF)
1840	(0.00,616.2),(0.98,585.1),(1.20,556.5)
1940	(0.00,623.8),(0.93,594.7),(1.20,563.5)
2040	(0.00,631.0),(0.88,603.8),(1.20,569.6)
2290	(0.00,647.9),(0.80,624.5),(1.20,586.5)
2440	(0.00,657.4),(0.77,635.6),(1.20,597.2)

FIGURE 2.1-1 Reactor Core Safety Limits -
Four Loops in Operation

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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,	$\leq$ 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,	$\leq$ 5% of RATED THERMAL POWER with 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	$\geq$ 1950 psig	$\geq$ 1940 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 91,240 gpm per loop.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
13. Steam Generator Water Level - Low-Low	$\geq$ 21% of narrow range instrument span - each steam generator	$\geq$ 19.2% of narrow range instrument span - each steam generator
14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	$\leq$ 1.47 x 10^6 lbs/hr of steam flow at RATED THERMAL POWER coincident with steam generator water level $\geq$ 25% of narrow range instrument span - each steam generator	$\leq$ 1.56 x 10^6 lbs/hr of flow at RATED THERMAL POWER coincident with generator water level $\geq$ 24% of narrow range instrument span - each steam generator
15. Undervoltage - Reactor Coolant Pumps	$\geq$ 2905 volts - each bus	$\geq$ 2870 volts - each bus
16. Underfrequency - Reactor Coolant Pumps	$\geq$ 57.5 Hz - each bus	$\geq$ 57.4 Hz - each bus
17. Turbine Trip		
A. Low Trip System Pressure	$\geq$ 58 psig	$\geq$ 57 psig
B. Turbine Stop Valve Closure	$\geq$ 1% open	$\geq$ 1% open
18. Safety Injection Input from ESF	Not Applicable	Not Applicable
19. Reactor Coolant Pump Breaker Position Trip	Not Applicable	Not Applicable

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 $\leq 574.1^{\circ}F$

P = Pressurizer Pressure, psig

P' = 2235 psig (indicated RCS nominal operating pressure)

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = 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} $\tau_1 = 33$ secs, $\tau_2 = 4$ secs.

S = Laplace transform operator

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

4 Loops in Operation

$K_1 = 1.2590$

$K_2 = 0.01374$

$K_3 = 0.000838$

and f_1 (ΔI) 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 -31 percent and +3 percent, f_1 (ΔI) = 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 -31 percent, the ΔT trip setpoint shall be automatically reduced by 2.9 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of ($q_t - q_b$) exceeds +3 percent, the ΔT trip setpoint shall be automatically reduced by 2.2 percent of its value at RATED THERMAL POWER.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_o \left[K_4 - K_5 \left(\frac{\tau_3 S}{1+\tau_3 S}\right) T - K_6 \left(T - T''\right) - f_2(\Delta I)\right]$

where:

ΔT_o	=	Indicated ΔT at rated power
T	=	Average temperature, ^oF
T''	=	Indicated T_{avg} at RATED THERMAL POWER $\leq$ 574.1 ^oF.
K_4	=	1.078
K_5	=	0.02/^oF for increasing average temperature and 0 for decreasing average temperature
K_6	=	0.00197 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} τ_3 = 10 secs.
S	=	Laplace transform operator
$f_2(\Delta I)$	=	$f_1(\Delta I)$ as defined in Note 1 above

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 3.3 percent ΔT span.

Note 4: The channel's maximum trip point shall not exceed its computed trip point by more than 2.6 percent ΔT span.

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - STANDBY, STARTUP, AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be $\geq$ 1.6% $\Delta k/k$.

APPLICABILITY: MODES 1, 2*, and 3.

ACTION:

With the SHUTDOWN MARGIN < 1.6% $\Delta k/k$, immediately initiate and continue boration at $\geq$ 10 gpm of 20,000 ppm boric acid solution or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be $\geq$ 1.6% $\Delta k/k$:

a. Within one 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 above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable control rod(s).

b. When in MODES 1 or $2^{\#}$, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6.

c. When in MODE $2^{\#\#}$, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6.

*See Special Test Exception 3.10.1

With $K_{eff} \geq 1.0$

With $K_{eff} < 1.0$

SURVEILLANCE REQUIREMENTS (Continued)

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the control banks at the maximum insertion limit of Specification 3.1.3.6.

e. When in MODE 3, at least once per 24 hours by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. Control rod position,
3. Reactor control system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within $\pm$ 1% $\Delta k/k$ at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

4.1.1.1.3 Prior to blocking ESF Functional Units in accordance with footnotes # and ## of Table 3.3-3, SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.6% $\Delta k/k$ by consideration of the factors of 4.1.1.1.1.e above. The Reactor Coolant System average temperature used in making this SHUTDOWN MARGIN determination shall be less than or equal to 350°F. This SHUTDOWN MARGIN shall be maintained at all times when the ESF functions are blocked in MODE 3.

SHUTDOWN MARGIN - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be :

a. In MODE 4:

1. $\geq$ 1.6% $\Delta k/k$ when operating with one or more Reactor Coolant Loops in accordance with Specification 3.4.1.3.

2. Greater than the value shown in Figure 3.1-3 when operating with no Reactor Coolant Loops but one or more Residual Heat Removal Loops in accordance with Specification 3.4.1.3.

b. In MODE 5:

1. $\geq$ 1.0% $\Delta k/k$ when operating with one or more Reactor Coolant Loops in accordance with Specification 3.4.1.3.

2. Greater than the value shown in Figure 3.1-3 when operating with no Reactor Coolant Loops but one or more Residual Heat Removal Loops in accordance with Specification 3.4.1.3

APPLICABILITY: MODES 4 and 5

ACTION:

With SHUTDOWN MARGIN less than the above limits, immediately initiate and continue boration at $\geq$ 10 gpm of 20,000 ppm boric acid solution or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIRMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than the above limits:

a. Within one hour after 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 increased by an amount at least equal to the withdrawn worth of the immovable or untrippable control rod(s).

SURVEILLANCE REQUIREMENTS (Continued)

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.

FIGURE 3.1-3 WILL BE PROVIDED

AT A LATER DATE

BORON DILUTION

LIMITING CONDITION FOR OPERATION

3.1.1.3 The flow rate of reactor coolant through the reactor coolant system shall be ≥ 3000 gpm whenever a reduction in Reactor Coolant System boron concentration is being made.*

APPLICABILITY: ALL MODES.

ACTION:

With the flow rate of reactor coolant through the reactor coolant system < 3000 gpm, immediately suspend all operations involving a reduction in boron concentration of the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The flow rate of reactor coolant through the reactor coolant system shall be determined to be ≥ 3000 gpm within one hour prior to the start of and at least once per hour during a reduction in the Reactor Coolant System boron concentration by either:

a. Verifying at least one reactor coolant pump is in operation, or

b. Verifying that at least one RHR pump is in operation and supplying ≥ 3000 gpm through the reactor coolant system.

*For purposes of this specification, addition of water from RWST does not constitute a dilution activity provided the boron concentration in the RWST is greater than or equal to the minimum required by specification 3.1.2.8.b.2 (MODES 1, 2, 3, and 4) or 3.1.2.7.b.2 (MODES 5 and 6).

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 One charging pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency bus.

APPLICABILITY: MODES 5 and 6.

ACTION:

a. With no charging pump OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.*

b. With more than one charging pump OPERABLE or with a safety injection pump(s) OPERABLE when the temperature of any RCS cold leg is less than or equal to 152°F, unless the reactor vessel head is removed, remove the additional charging pump(s) and the safety injection pump(s) motor circuit breakers from the electrical power circuit within one hour.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.1.2.3.1 The above required charging pump shall be demonstrated OPERABLE by verifying, that on recirculation flow, the pump develops a discharge pressure of ≥ 2405 psig when tested pursuant to Specification 4.0.5

4.1.2.3.2 All charging pumps and safety injection pumps, excluding the above required OPERABLE charging pump, shall be demonstrated inoperable by verifying that the motor circuit breakers have been removed from their electrical power supply circuits at least once per 12 hours, except when:

a. The reactor vessel head is removed, or

b. The temperature of all RCS cold legs is greater than 152°F.

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* For purposes of this specification, addition of water from the RWST does not constitute a positive reactivity addition provided the boron concentration in the the RWST is greater than the minimum required by Specification 3.1.2.7.b.2.

BORIC ACID TRANSFER PUMPS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 At least one boric acid transfer pump shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if only the flow path through the boric acid transfer pump of Specification 3.1.2.1a is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no boric acid transfer pump OPERABLE as required to complete the flow path of Specification 3.1.2.1a, suspend all operations involving CORE ALTERATIONS or positive reactivity changes* until at least one boric acid transfer pump is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.5 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*For purposes of this specification, addition of water from the RWST does not constitute a positive reactivity addition provided the boron concentration in the RWST is greater than the minimum required by Specification 3.1.2.7.b.2

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. A boric acid storage system and associated heat tracing with:

1. A minimum contained borated water volume 835 gallons,
2. Between 20,000 and 22,500 ppm of boron, and
3. A minimum solution temperature of 145°F.

b. The refueling water storage tank with:

1. A minimum contained borated water volume of ______ gallons,
2. A minimum boron concentration of 2000 ppm, and
3. A minimum solution temperature of 35°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one borated water source is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

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

a. At least once per 7 days by:

1. Verifying the boron concentration of the water,
2. Verifying the contained borated water volume, and
3. Verifying the boric acid storage tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water and the outside air temperature is < 35°F.

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) rod drop time from the fully withdrawn position shall be ≤ 2.2 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. $T_{avg} \geq 541^{\circ}F$, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

SURVEILLANCE REQUIREMENTS

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

a. For all rods following each removal of the reactor vessel head.

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. At least once per 18 months.

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1* and 2*#.

ACTION:

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

*See Special Test Exceptions 3.10.2 and 3.10.3
#With $K_{eff} \geq$ 1.0.

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ACTION: (Continued)

b. THERMAL POWER shall not be increased above 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER unless the indicated AFD is within the target band and ACTION 2.a) 1), above has been satisfied.

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD has not been outside of the target band for more than 1 hour penalty deviation cumulative during the previous 24 hours.

d. During power reductions using control rods, the reporting requirements of Specification 6.9.1.9 shall not apply provided the action items above are satisfied.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AXIAL FLUX DIFFERENCE shall be determined to be within its limits during POWER OPERATION above 15% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

1. At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

2. At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status.

b. Monitoring an logging the indicated AXIAL FLUX DIFFERENCE for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AXIAL FLUX DIFFERENCE Monitor Alarm is inoperable. The logged values of the indicated AXIAL FLUX DIFFERENCE shall be assumed to exist during the interval preceeding each logging.

SURVEILLANCE REQUIREMENTS (continued)

4.2.1.2 The indicated AFD shall be considered outside of its target band when at least 2 of 4 or 2 of 3 OPERABLE excore channels are indicating the AFD to be outside the target band. Penalty deviation outside of the target band shall be accumulated on a time basis of:

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

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

4.2.1.3 The target axial flux difference for the OPERABLE excore channels shall be determined in conjunction with the measurement of APL as defined in Specification 4.2.6.2. The provisions of Specification 4.0.4 are not applicable.

4.2.1.4 The axial flux difference target band about the target axial flux difference shall be determined in conjunction with the measurement of APL as defined in Specification 4.2.6.2. The allowable values of the target band are ±5% or ±3%. Redefinition of the target band from ±3% to ±5% between determinations of the target axial flux difference is allowed when appropriate redefinitions of APL are made. Redefinition of the target band from ±5% to ±3% is allowed only in conjunction with the determination of a new target axial flux difference. The provisions of Specification 4.0.4 are not applicable.

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:

Westinghouse Fuel	Exxon Nuclear Co. Fuel	
$F_Q(Z) \leq \frac{[1.97]}{P}$ [K(Z)]	$F_Q(Z) \leq \frac{[2.10]}{P}$ [K(Z)]	$P > 0.5$
$F_Q(Z) \leq$ [3.94] [K(Z)]	$F_Q(Z) \leq$ [4.20] [K(Z)]	$P \leq 0.5$

- $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$
- $F_Q(Z)$ is the measured $F_Q(Z)$, including a 3% manufacturing tolerance uncertainty and a 5% measurement uncertainty.
- K(Z) is the function obtained from Figure 3.2-2 for Westinghouse fuel and Figure 3.2-2(a) for Exxon Nuclear Company fuel.

APPLICABILITY: MODE 1

ACTION:

With $F_Q(Z)$ exceeding its limit:

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

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 $F_Q(Z)$ shall be determined to be within its limit above 5% of RATED THERMAL POWER according to the following schedule:

a. Whenever $F_Q(Z)$ is measured for reasons other than meeting the requirement of 4.2.6.2, or

b. At least once per 31 effective full power days, whichever occurs first.

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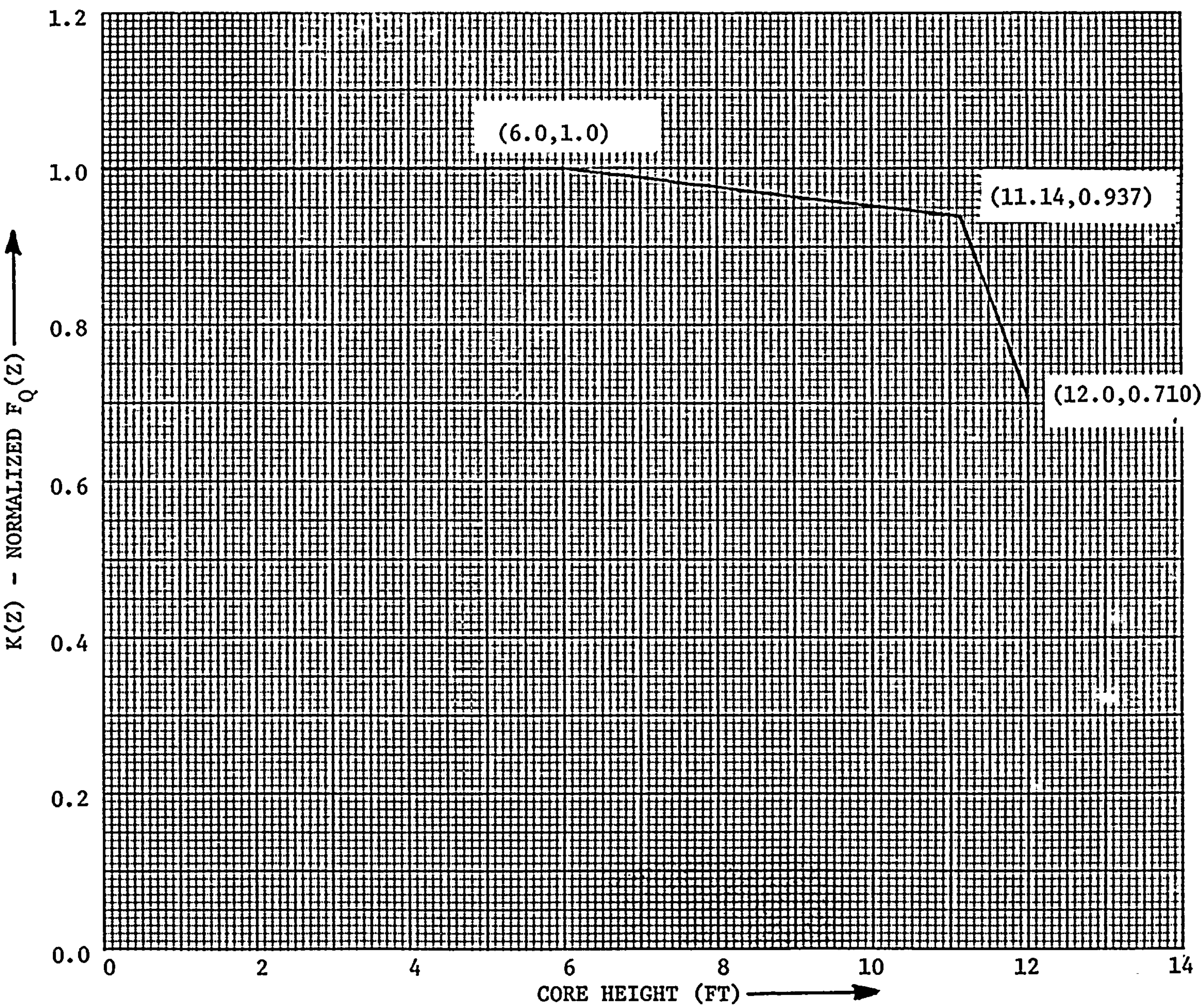

FIGURE 3.2-2(a) D. C. COOK UNIT 2, K(Z) - NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT FOR EXXON NUCLEAR CO. FUEL

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 relationships:

$F^N_{\Delta H} \leq 1.48\ [1 + 0.2\ (1-P)]$ (for Westinghouse fuel)

and $F^N_{\Delta H} \leq 1.49\ [1 + 0.2\ (1-P)]$ (for Exxon Nuclear Co. fuel)

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.

SURVEILLANCE REQUIREMENTS

4.2.3 $F^N_{\Delta H}$ shall be determined to be within its limit by using the movable incore detectors to obtain a power distribution map:

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

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

c. The provisions of Specification 4.0.4 are not applicable.

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POWER DISTRIBUTION LIMITS

DNB PARAMETERS - MODE 1

LIMITING CONDITION FOR OPERATION

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

a. Reactor Coolant System T_{avg}.

b. Pressurizer Pressure.

c. Reactor Coolant System Total Flow Rate.

APPLICABILITY: MODE 1.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

4.2.5.1.3 The RCS total flow rate shall be determined by a power balance around the steam generators at least once per 18 months.

4.2.5.1.4 The provisions of Specification 4.0.4 shall not apply to primary flow surveillances.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS 4 Loops in Operation
Reactor Coolant System T_{avg}**	$\leq$ 576.3°F. (indicated)
Pressurizer Pressure**	$\geq$ 2205 psig*
Reactor Coolant System Total Flow Rate	$\geq 138.6 \times 10^6$ lbs/hr

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

** Indicated average of OPERABLE instrument loops.

DNB PARAMETERS - MODES 2 and 3

LIMITING CONDITION FOR OPERATION

3.2.5.2 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-2:

a. Reactor Coolant System T_{avg}.

b. Pressurizer Pressure.

APPLICABILITY: MODES 2 and 3*

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or open the reactor trip system breakers within the next hour.

SURVEILLANCE REQUIREMENTS

4.2.5.2 Each of the parameters of Table 3.2-2 shall be verified to be within their limits at least once per 12 hours.

* With the reactor trip system breakers in the closed position and the control rod drive system capable of rod withdrawal.

TABLE 3.2-2

DNB PARAMETERS

PARAMETER	LIMIT
Reactor Coolant System T_{avg}	$\leq$ 549.2°F. (Reactor Subcritical)
Reactor Coolant System T_{avg}	$\leq$ 576.3°F. (Reactor Critical)
Pressurizer Pressure	$\geq$ 2176 psig

Reactor coolant loop operational requirements are contained in Specifications 3.4.1.1 and 3.4.1.2.c.

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 relationships:

$$APL = \min \text{ over } Z \text{ of } \frac{1.97\ K(Z)}{F_Q(Z) x V(Z) x F_p} \times 100\% \qquad \text{Westinghouse Fuel}$$

$$APL = \min \text{ over } Z \text{ of } \frac{2.10\ K(Z)}{F_Q(Z) x V(Z) x F_p} \times 100\% \qquad \text{Exxon Nuclear Co. Fuel}$$

- $F_Q(Z)$ is the measured $F_Q(Z)$, including a 3% manufacturing tolerance uncertainty and a 5% measurement uncertainty.
- V(Z) is the function defined in Figure 3.2-3 which corresponds to the target band.
- F_p = 1.00 except when successive steady-state power distribution maps indicate an increase in peak pin power, $F_{\Delta H}$, with exposure. Then either of the following 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 2 successive maps indicate that the peak pin $F_{\Delta H}$ 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

ACTION:

With THERMAL POWER exceeding APL:

a. Reduce THERMAL POWER to APL or less of RATED THERMAL POWER within 15 minutes. Then reduce the Power Range Neutron Flux-High Trip Setpoints by the same percentage which APL is below RATED THERMAL POWER within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced the same percentage which APL is below RATED THERMAL POWER.

b. THERMAL POWER may be increased to a new APL calculated at the reduced power by either redefining the target axial flux difference or by correcting the cause of the high $F_Q(Z)$ condition.

SURVEILLANCE REQUIREMENTS

4.2.6.1 The provisions of Specification 4.0.4 are not applicable.

4.2.6.2 APL shall be determined by measurement in conjunction with the target flux difference and target band determination* above 15% of RATED THERMAL POWER, according to the following schedule:

a. Upon achieving equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which APL was last determined**, or

b. At least once per 31 effective full power days, whichever occurs first.

*APL can be redefined by remeasuring the target axial flux difference in accordance with ACTION statement b of Specification 3.2.6.

**During power escalation at the beginning of each cycle, the design target may be used until a power level for extended operation has been achieved.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2 and *	12
2. Power Range, Neutron Flux	4	2	3	1, 2 and *	2#
3. Power Range, Neutron Flux, High Positve Rate	4	2	3	1, 2	2#
4. Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2	2#
5. Intermediate Range, Neutron Flux	2	1	2	1, 2 and *	3
6. Source Range, Neutron Flux					
A. Startup	2	1	2	2## and *	4
B. Shutdown	2	0	1	3, 4 and 5	5
7. Overtemperature ΔT Four Loop Operation	4	2	3	1, 2	6#
8. Overpower ΔT Four Loop Operation	4	2	3	1, 2	6#

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
16. Undervoltage-Reactor Coolant Pumps	4-1/bus	2	3	1	6#
17. Underfrequency-Reactor Coolant Pumps	4-1/bus	2	3	1	6#
18. Turbine Trip					
A. Low Fluid Oil Pressure	3	2	2	1	7#
B. Turbine Stop Valve Closure	4	4	3	1	6#
19. Safety Injection Input from ESF	2	1	2	1, 2	1
20. Reactor Coolant Pump Breaker Position Trip					
A. Above P-8	1/breaker	1	1/breaker	1	10
B. Above P-7	1/breaker	2	1/breaker per operating loop	1	11#
21. Reactor Trip Breakers	2	1	2	1, 2 and *	1
22. Automatic Trip Logic	2	1	2	1, 2 and *	1

TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the reactor trip system breakers in the closed position and the control rod drive system capable of rod withdrawal.

#The 'provisions of Specification 3.0.4 are not applicable.

##High voltage to detector may be de-energized above P-6.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minumum Channels OPERABLE requirement, be in HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.1.

ACTION 2 - With the number of OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour.

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.1.

c. Either, THERMAL POWER is restricted to $\leq$ 75% of RATED THERMAL POWER and the Power Range, Neutron Flux trip setpoint is reduced to $\leq$ 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours per Specification 4.2.4.c.

ACTION 3 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

TABLE 3.3-1 (Continued)

ACTION 10 - With one channel inoperable, restore the inoperable channel to OPERABLE status within 2 hours or reduce THERMAL POWER to below P-8 within the next 2 hours. Operation below P-8 may continue pursuant to ACTION 11.

ACTION 11 - With less than the Minimum Number of Channels OPERABLE, operation may continue provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 12 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours and/or open the reactor trip breakers.

REACTOR TRIP SYSTEM INTERLOCKS

DESIGNATION	CONDITION AND SETPOINT	FUNCTION
P-6	With 2 of 2 Intermediate Range Neutron Flux Channels $< 6 \times 10^{-11}$ amps.	P-6 prevents or defeats the manual block of source range reactor trip.

TABLE 3.3-1 (Continued)

DESIGNATION	CONDITION AND SETPOINT	FUNCTION
P-7	With 2 of 4 Power Range Neutron Flux Channels $\geq$ 11% of RATED THERMAL POWER or 1 of 2 Pressure Before the First Stage channels $\geq$ 51 psig.	P-7 prevents or defeats the automatic block of reactor trip on: Low flow in more than one primary coolant loop, reactor coolant pump under-voltage and under-frequency, turbine trip, pressurizer low pressure, and pressurizer high level.
P-8	With 2 of 4 Power Range Neutron Flux channels $\geq$ 31% of RATED THERMAL POWER.	P-8 prevents or defeats the automatic block of reactor trip on low coolant flow in a single loop.
P-10	With 3 of 4 Power Range Neutron Flux channels $\geq$ 9% of RATED THERMAL POWER.	P-10 prevents or defeats the manual block of: Power range low setpoint reactor trip, Intermediate range reactor trip, and intermediate range rod stops. Provides input to P-7.

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	$\leq$ 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 ΔT	$\leq$ 6.0 seconds*
8. Overpower ΔT	NOT APPLICABLE
9. Pressurizer Pressure--Low	$\leq$ 1.0 seconds
10. Pressurizer Pressure--High	$\leq$ 1.0 seconds
11. Pressurizer Water Level--High	NOT APPLICABLE

*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 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip##				
A. Shunt Trip Function	N.A.	N.A.	S/U(1)	N.A.
B. Undervoltage Trip Function	N.A.	N.A.	S/U(1)	N.A.
2. Power Range, Neutron Flux	S	D(2,8),M(3,8) and Q(6,8)	M and S/U(1)	1, 2 and *
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(6)	M	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R(6)	M	1, 2
5. Intermediate Range, Neutron Flux	S	R(6,8)	S/U(1)	1, 2 and *
6. Source Range, Neutron Flux	N.A.	R(6,8)	M(8) and S/U(1)	2(7), 3(7), 4 and 5
7. Overtemperature ΔT	S	R(9)	M	1, 2
8. Overpower ΔT	S	R(9)	M	1, 2
9. Pressurizer Pressure--Low	S	R	M	1, 2
10. Pressurizer Pressure--High	S	R	M	1, 2
11. Pressurizer Water Level--High	S	R	M	1, 2
12. Loss of Flow - Single Loop	S	R(8)	M	1

TABLE 4.3-1 (Continued)

NOTATION

* - With the reactor trip system breakers closed and the control rod drive system capable of rod withdrawal.

(1) - If not performed in previous 7 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER. Adjust channel if absolute difference > 2 percent.

(3) - Compare incore to excore axial offset above 15% of RATED THERMAL POWER. Recalibrate if absolute difference $\geq$ 3 percent.

(4) - Manual ESF functional input check every 18 months.

(5) - Each train tested every other month.

(6) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(7) - Below P-6 (BLOCK OF SOURCE RANGE REACTOR TRIP) setpoint.

(8) - The provisions of Specification 4.0.4 are not applicable.

(9) - The provisions of Specification 4.0.4 are not applicable for $f_1(\Delta I)$ and $f_2(\Delta I)$ penalties. (See also note 1 of Table 2.2-1)

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION, TURBINE TRIP, FEEDWATER ISOLATION AND MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS					
a. Manual Initiation	2	1	2	1, 2, 3, 4	18
b. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	13
c. Containment Pressure-High	3	2	2	1, 2, 3	14*
d. Pressurizer Pressure-Low	3	2	2	1, 2, 3#	14*
e. Differential Pressure Between Steam lines - High					
Four Loops Operating	3/steam line	2/steam line any steam line	2/steam line	1, 2, 3##	14*
Three Loops Operating	3/operating steam line	1####/steam line any operating steam line	2/operating steam line	3##	15

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
f. Steam Line Pressure-Low					
Four Loops Operating	1 pressure/ loop	2 pressures any loops	1 pressure any 3 loops	1, 2, 3##	14*
Three Loops Operating	1 pressure/ operating loop	1### pressure in any operating loop	1 pressure in any 2 operating loops	3##	15

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
2. CONTAINMENT SPRAY					
a. Manual	2	2	2	1, 2, 3, 4	18
b. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	13
c. Containment Pressure High-High	4	2	2	1, 2, 3	16

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. CONTAINMENT ISOLATION					
a. Phase "A" Isolation					
1) Manual	2	1	2	1, 2, 3, 4	18
2) From Safety Injection Automatic Actuation Logic	2	1	2	1, 2, 3, 4	13
b. Phase "B" Isolation					
1) Manual	2	2	2	1, 2, 3, 4	18
2) Automatic Actuation Logic	2	1	2	1, 2, 3, 4	13
3) Containment Pressure-High-High	4	2	3	1, 2, 3	16
c. Purge and Exhaust Isolation					
1) Manual	2	1	2	1, 2, 3, 4	17
2) Containment Radioactivity-High Train A	3	1	2	1, 2, 3, 4	17
3) Containment Radioactivity-High Train B	3	1	2	1, 2, 3, 4	17

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. STEAM LINE ISOLATION					
a. Manual	1/steam line	1/steam line	1/operating steam line	1, 2, 3	18
b. Automatic Actuation Logic	2	1	2	1, 2, 3	13
c. Containment Pressure--High-High	4	2	3	1, 2, 3	16
d. Steam Flow in Two Steam Lines--High					
Four Loops Operating	2/steam line	1/steam line any 2 steam lines	1/steam line	1, 2, 3##	14*
Three Loops Operating	2/operating steam line	1###/any operating steam line	1/operating steam line	3##	15
COINCIDENT WITH T_{avg}--Low-Low					
Four Loops Operating	1 T_{avg}/loop	2 T_{avg} any loops	1 T_{avg} any 3 loops	1, 2, 3##	14*

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
Three Loops Operating	1 T_{avg}/operating loop	1### T_{avg} in any operating loop	1 T_{avg} in any two operating loops	3##	15
e. Steam Line Pressure-Low					
Four Loops Operating	1 pressure/loop	2 pressures any loops	1 pressure any 3 loops	1, 2, 3##	14*
Three Loops Operating	1 pressure/operating loop	1### pressure in any operating loop	1 pressure in any 2 operating loops	3##	15
5. TURBINE TRIP & FEEDWATER ISOLATION					
a. Steam Generator Water Level--High-High	3/loop	2/loop in any operating loop	2/loop in each operating loop	1, 2, 3 and $	14*

TABLE 3.3-3 (Continued)

TABLE NOTATION

#Trip function may be bypassed in this MODE below P-11, provided that:

(1) All control and shutdown rods are verified inserted. Control and shutdown rods shall be maintained inserted at all times that ESF actuations are blocked in MODE 3.

(2) One cooldown steam dump valve is blocked. The blocked steam dump valve may be unblocked when the Reactor Coolant System average temperature is below 495°F or when SI functional units 1d, 1e, 1f, 4d and 4e are available. One cooldown steam dump valve shall be maintained blocked at all times that Reactor Coolant System average temperature is above 495°F and ESF actuations are blocked in MODE 3.

(3) The SHUTDOWN MARGIN has been determined in accordance with Specification 4.1.1.1.3.

##Trip function may be bypassed in this MODE below P-12, provided that:

(1) All control and shutdown rods are verified inserted. Control and shutdown rods shall be maintained inserted at all times that ESF actuations are blocked in MODE 3.

(2) One cooldown steam dump valve is blocked. The blocked steam dump valve may be unblocked when the Reactor Coolant System average temperature is below 495°F or when SI functional units 1d, 1e, 1f, 4d and 4e are available. One cooldown steam dump valve shall be maintained blocked at all times that Reactor Coolant System average temperature is above 495°F and ESF actuations are blocked in MODE 3.

(3) The SHUTDOWN MARGIN has been determined in accordance with Specification 4.1.1.1.3.

###The channel(s) associated with the protective functions derived from the out of service Reactor Coolant Loop shall be placed in the tripped mode.

####Trip all bistables which indicate low active loop steam pressure with respect to the idle loop steam pressure.

*The provisions of Specification 3.0.4 are not applicable.

$When feedwater is being supplied to the steam generators by a main feedpump.

TABLE 3.3-3 (Continued)

ACTION STATEMENTS

ACTION 13 - With the number of OPERABLE Channels one less than the Total Number of Channels, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.1.

ACTION 14 - With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed until performance of the next required CHANNEL FUNCTIONAL TEST provided the inoperþble channel is placed in the tripped condition within 1 hour.

ACTION 15 - With a channel associated with an operating loop inoperable, restore the inoperable channel to OPERABLE status within 2 hours or be in HOT SHUTDOWN within the following 12 hours; however, one channel associated with an operating loop may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.1.

ACTION 16 - With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is met; one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.1.

TABLE 3.3-3 (Continued)

ACTION 17 - With less than the Minimum Channels OPERABLE, operation may continue provided the containment purge and exhaust valves are maintained closed.

ACTION 18 - With the number of OPERABLE Channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 19 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour.

b. The Minimum Channels OPERABLE requirements is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

ENGINEERED SAFETY FEATURES INTERLOCKS

DESIGNATION	CONDITION AND SETPOINT	FUNCTION
P-11	With 2 of 3 Pressurizer Pressure channels $\geq$ 2010 psig.	P-11 prevents or defeats manual block of safety injection actuation on low pressurizer pressure.
P-12	With 2 of 4 T_{avg} channels $\leq$ Setpoint. Setpoint $\geq$ 541^{o}F.	P-12 allows manual block of safety injection actuation on low steam line pressure. Causes steam line isolation on high steam flow. Affects steam dump blocks. With 3 of 4 T_{avg} channels above the reset point, prevents or defeats the manual block of safety injection actuation on low steam line pressure.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
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	$\leq$ 3.0 psig
d. Steam Flow in Two Steam Lines--High Coincident with T_{avg} --Low-Low	$\leq$ A function defined as follows: A Δp corresponding to 1.47×10^6 lbs/hr steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 110% of full steam flow at full load. $T_{avg} \geq 541^{\circ}F$	$\leq$ A function defined as follows: A Δp corresponding to 1.62×10^6 lbs/hr steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 111.5% of full steam flow at full load. $T_{avg} \geq 539^{\circ}F$
e. Steam Line Pressure--Low	$\geq$ 600 psig steam line pressure	$\geq$ 585 psig steam line pressure
5. TURBINE TRIP AND FEED WATER 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

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
6. MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS		
a. Steam Generator Water Level -- Low-Low	≥ 21% of narrow range instrument span each steam generator	≥ 19.2% of narrow range instrument span each steam generator
b. 4 kv Bus Loss of Voltage	3196 volts with a 2 second delay	3196, +18, -36 volts with a 2 ± 0.2 second delay
c. Safety Injection	Not Applicable	Not Applicable
d. Loss of Main Feedwater Pumps	Not Applicable	Not Applicable
7. TURBINE DRIVEN AUXILIARY FEEDWATER PUMPS		
a. Steam Generator Water Level--Low-Low	≥ 21% of narrow range instrument span each steam generator	≥ 19.2% of narrow range instrument span each steam generator
b. Reactor Coolant Pump Bus Undervoltage	≥ 2750 Volts--each bus	≥ 2725 Volts--each bus
8. LOSS OF POWER		
a. 4 kv Bus Loss of Voltage	3196 volts with a 2 second delay	3196, +18, -36 volts with a 2 ± 0.2 second delay
b. 4 kv Bus Degraded Voltage	3596 volts with a 2.0 minute time delay	3596, +36, -18 volts with a 2.0 minute ± 6 second time delay

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
4. STEAM LINE ISOLATION				
a. Manual	N.A.	N.A.	M(1)	1, 2, 3
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3
c. Containment Pressure--High-High	S	R	M(3)	1, 2, 3
d. Steam Flow in Two Steam Lines--High Coincident with T_{avg} --Low-Low	S	R	M	1, 2, 3
e. Steam Line Pressure--Low	S	R	M	1, 2, 3
5. TURBINE TRIP AND FEEDWATER ISOLATION				
a. Steam Generator Water Level--High-High	S	R	M	1, 2, 3 and $
6. MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS				
a. Steam Generator Water Level -- Low-Low	S	R	M	1, 2, 3
b. 4 kv Bus Loss of Voltage	S	R	M	1, 2, 3
c. Safety Injection	N.A.	N.A.	M(2)	1, 2, 3
d. Loss of Main Feedwater Pumps	N.A.	N.A.	R	1, 2

TABLE 4.3-2 (Continued)

TABLE NOTATION

$ When feedwater is being supplied to the steam generators by a main feedpump.

(1) Manual actuation switches shall be tested at least once per 18 months during shutdown. All other circuitry associated with manual safeguards actuation shall receive a CHANNEL FUNCTIONAL TEST at least once per 31 days.

(2) Each train or logic channel shall be tested at least every other 31 days.

(3) The CHANNEL FUNCTIONAL TEST shall include exercising the transmitter by applying either a vacuum or pressure to the appropriate side of the transmitter.

MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The movable incore detection system shall be OPERABLE with:

a. At least 75% of the detector thimbles,

b. A minimum of 2 detector thimbles per core quadrant, and

c. Sufficient movable detectors, drive, and readout equipment to map these thimbles.

APPLICABILITY: When the movable incore detection system is used for:

a. Recalibration of the excore neutron flux detection system,

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$ and $F_Q(Z)$.

ACTION:

With the movable incore detection system inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The movable incore detection system shall be demonstrated OPERABLE by normalizing each detector output when required for:

a. Recalibration of the excore neutron flux detection system, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$ and $F_Q(Z)$.

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HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 a. The reactor coolant loops listed below shall be OPERABLE and in operation as required by items b and c:

1. Reactor Coolant Loop 1 and its associated steam generator and reactor coolant pump,

2. Reactor Coolant Loop 2 and its associated steam generator and reactor coolant pump,

3. Reactor Coolant Loop 3 and its associated steam generator and reactor coolant pump,

4. Reactor Coolant Loop 4 and its associated steam generator and reactor coolant pump.

b. At least two of the above coolant loops shall be OPERABLE and at least one loop in operation if the reactor trip breakers are in the open position, or the control rod drive system is not capable of rod withdrawal.*

c. At least three of the above coolant loops shall be OPERABLE and in operation when the reactor trip system breakers are in the closed position and the control rod drive system is capable of rod withdrawal.

APPLICABILITY: MODE 3

ACTION:

a. With less than the above required reactor coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With no reactor coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System** and immediately initiate corrective action to return the required coolant loop to operation.

---

* All reactor coolant pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration**, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

** For purposes of this specification, addition of water from the RWST does not constitute a dilution activity provided the boron concentration in the RWST is greater than or equal to the minimum required by specification 3.1.2.8.b.2.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 At least one cooling loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 a. At least two of the reactor coolant loops listed below shall be OPERABLE:

1. Reactor Coolant Loop 1 and its associated steam generator and reactor coolant pump,*

2. Reactor Coolant Loop 2 and its associated steam generator and reactor coolant pump,*

3. Reactor Coolant Loop 3 and its associated steam generator and reactor coolant pump,*

4. Reactor Coolant Loop 4 and its associated steam generator and reactor coolant pump,*

5. Residual Heat Removal - East, **

6. Residual Heat Removal - West **

b. At least one of the above coolant loops shall be in operation.***

APPLICABILITY: MODES 4 and 5

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; be in COLD SHUTDOWN within 20 hours.

b. With no coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System**** and immediately initiate corrective action to return the required coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required residual heat removal loop(s) shall be determined OPERABLE per Specification 4.0.5.

4.4.1.3.2 The required reactor coolant pump(s), if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.3 The required steam generator(s) shall be determined OPERABLE by verifying secondary side level to be greater than or equal to 25% of wide range instrument span at least once per 12 hours.

4.4.1.3.4 At least one coolant loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

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* A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to 152°F unless 1) the pressurizer water volume is less than 62.00% of span or 2) the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures. Operability of a reactor coolant loop(s) does not require an OPERABLE auxiliary feedwater system.

** The normal or emergency power source may be inoperable in MODE 5.

*** All reactor coolant pumps and residual heat removal pumps may be de-energized for up to 1 hour provided 1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration****, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

**** For purposes of this specification, addition of water from the RWST does not constitute a dilution activity provided the boron concentration in the RWST is greater than or equal to the minimum required by specification 3.1.2.8.b.2 (MODE 4) or 3.1.2.7.b.2 (MODE 5).

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SAFETY VALVES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2485 PSIG ± 1%.*

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer code safety valve OPERABLE:

a. Immediately suspend all operations involving positive reactivity changes and place an OPERABLE RHR loop into operation in the shutdown cooling mode.

b. Immediately render all Safety Injection pumps and all but one charging pump inoperable by removing the applicable motor circuit breakers from the electrical power circuit within one hour.

SURVEILLANCE REQUIREMENTS

4.4.2 No additional Surveillance Requirements other than those required by Specification 4.0.5.

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

RELIEF VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.11 Three power operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. PORVs inoperable:*

1. With one PORV inoperable,

within 1 hour either restore the PORV to OPERABLE status or close the associated block valve and remove power from the block valve; otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

2. With two PORVs inoperable,

within 1 hour either restore at least one PORV to OPERABLE status or close the associated block valves and remove power from the block valves; restore at least one PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3. With three PORVs inoperable,

within 1 hour either restore at least one of the PORVs to OPERABLE status or close their associated block valves and remove power from the block valves and be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

b. Block valves inoperable:*

1. With one block valve inoperable,

within 1 hour either (1) restore the block valve to OPERABLE status, or (2) close the block valve and remove power from the block valve, or (3) close the associated PORV and remove power from the associated solenoid valve; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

---

* PORVs isolated to limit RCS leakage through their seats and the block valves shut to isolate this leakage are not considered inoperable.

LIMITING CONDITION FOR OPERATION (Continued)

2. With two or more block valves inoperable,

   within 1 hour either (1) restore a total of at least two block valves to OPERABLE status, or (2) close the block valves and remove power from the block valves, or (3) close the associated PORVs and remove power from their associated solenoid valves; and apply the portions of ACTION a.2 or a.3 above for inoperable PORVs, relating to OPERATIONAL MODE, as appropriate.

c. With PORVs and block valves not in the same line inoperable,*

   within 1 hour either (1) restore the valves to OPERABLE status or (2) close and de-energize the other valve in each line. Apply the portions of ACTION a.2 or a.3 above, relating to OPERATIONAL MODE, as appropriate for two or three lines unavailable.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.11.1 Each of the three PORVs shall be demonstrated OPERABLE:

a. At least once per 31 days by performance of a CHANNEL FUNCTIONAL TEST, excluding valve operation, and

b. At least once per 18 months by performance of a CHANNEL CALIBRATION.

4.4.11.2 Each of the three block valves shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel. The block valve(s) do not have to be tested when ACTION 3.4.11.a or 3.4.11.c is applied.

4.4.11.3 The emergency power supply for the PORVs and block valves shall be demonstrated OPERABLE at least once per 18 months by operating the valves through a complete cycle of full travel while the emergency buses are energized by the onsite diesel generators and onsite plant batteries. This testing can be performed in conjunction with the requirements of Specifications 4.8.1.1.2.c and 4.8.2.3.2.d.

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* PORVs isolated to limit RCS leakage through their seats and the block valves shut to isolate this leakage are not considered inoperable.

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line code safety valves associated with each steam generator shall be OPERABLE with lift settings as specified in Table 3.7-4.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With 4 reactor coolant loops and associated steam generators in operation and with one or more main steam line code safety valves inoperable, operation in MODES 1, 2 and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With 3 reactor coolant loops and associated steam generators in operation and with one or more main steam line code safety valves associated with an operating loop inoperable, operation in MODE 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the reactor trip breakers are opened; otherwise, be in COLD SHUTDOWN within the next 30 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

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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 $\geq$ 1240 psig at 60°F on recirculation flow.

2. Verifying that the steam turbine driven pump develops an equivalent discharge pressure of $\geq$ 1180 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.

STEAM GENERATOR STOP VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each steam generator stop valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

MODE 1 - With one steam generator stop valve inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise, reduce power to less than or equal to 5 percent of RATED THERMAL POWER within the next 2 hours.

MODES 2 and 3 - With one steam generator stop valve inoperable, subsequent operation in MODES 2 or 3 may proceed provided:

a. The stop valve is maintained closed.

b. The provisions of Specification 3.0.4 are not applicable.

Otherwise, be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5.1 Each steam generator stop valve shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to Specification 4.0.5.

4.7.1.5.2 The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

4.7.1.5.3 The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 when performing PHYSICS TESTS at the beginning of a cycle provided the steam generator stop valves are maintained closed.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one residual heat removal loop shall be in operation.

APPLICABILITY: MODE 6.

ACTION:

a. With less than one residual heat removal loop in operation, except as provided in b. below, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System.* Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

b. The residual heat removal loop may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8.1 A residual heat removal loop shall be determined to be in operation and circulating reactor coolant at a flow rate of ≥ 3000 gpm at least once per 24 hours.

* For purposes of this specification, addition of water from the RWST does not constitute a dilution activity provided the boron concentration in the RWST is greater than or equal to the minimum required by specification 3.1.2.7.b.2.

GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

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

a. The THERMAL POWER is maintained ≤ 85% of RATED THERMAL POWER, and

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

APPLICABILITY: MODE 1

ACTION:

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

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

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

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

4.10.2.2 The Surveillance Requirements of Specifications 4.2.2.2 and 4.2.3 shall be performed at the following frequencies during PHYSICS TESTS:

a. Specification 4.2.2.2 - At least once per 12 hours.*

b. Specification 4.2.3 - At least once per 12 hours.

* The provisions of Specification 4.0.4 are not applicable.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB. This relation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the XNB correlation in this application). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average tempeature below which the calculated DNBR is no less than the correlation DNBR limit value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid. Uncertainties in primary system pressure, core temperature, core thermal power, primary coolant flow rate, and fuel fabrication tolerances have been included in the analyses from which Figure 2.1-1 is derived.

The Power Range Negative Rate trip provides protection to ensure that the calculated DNBR is maintained above the design DNBR value for multiple control rod drop accidents. The analysis of a single control rod drop accident indicates a return to full power may be initiated by the automatic control system in response to a continued full power turbine load demand or by the negative moderator temperature feedback. A single control rod drop analysis with automatic rod control has not been performed for Cycle 6. The plant will be operated under the "Interim Criteria for Single Dropped Rod."

Intermediate and Source Range, Neutron 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 ΔT

The Overtemperature ΔT 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. This reactor trip limit is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are more severe 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.

BASES

Overpower ΔT

The Overpower ΔT reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature ΔT 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. 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. If axial peaks are more severe 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.

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. 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.

Loss of Flow

The Loss of Flow trips provide core protection to prevent DNB in the event of a loss of one or more reactor coolant pumps.

Above 11 percent of RATED THERMAL POWER, an automatic reactor trip will occur if the flow in any two loops drop below 89% of design flow. Above the P-8 setpoint, ≤ 31% of RATED THERMAL POWER, automatic reactor trip will occur if the flow in any single loop drops below 89% of design flow. Design flow is defined as the minimum indicated loop flow supported by the safety analysis.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity. The specified setpoint provides allowance that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays of the auxialiary feedwater system.

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

The Steam/Feedwater Flow Mismatch in coincidence with a Steam Generator Low Water Level trip is not used in the transient and accident analyses but is included in Table 2.2-1 to ensure the functional capability of the specified trip settings and thereby enhance the overall reliability of the Reactor Protection System. This trip is redundant to the Steam Generator Water Level Low-Low trip. The Steam/Feedwater Flow Mismatch portion of this trip is activated when the steam flow exceeds the feedwater flow by greater than or equal to 1.47×10^6 lbs/hour. The Steam Generator Low Water level portion of the trip is activated when the water level drops below 25 percent, as indicated by the narrow range instrument. These trip values include sufficient allowance in excess of normal operating values to preclude spurious trips but will initiate a reactor trip before the steam generators are dry. Therefore, the required capacity and starting time requirements of the auxiliary feedwater pumps are reduced and the resulting thermal transient on the Reactor Coolant System and steam generators is minimized.

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump bus trips provide reactor core protection against DNB as a result of loss of voltage or underfrequency to more than one reactor coolant pump. The specified set points assure a reactor trip signal is generated before the low flow trip set point is reached. Time delays are incorporated in the underfrequency and undervoltage trips to prevent spurious reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed 0.9 seconds. For underfrequency, the delay is set so that the time required for a signal to reach the reactor trip breakers after the underfrequency trip set point is reached shall not exceed 0.3 seconds.

Turbine Trip

A Turbine Trip causes a direct reactor trip when operating above P-7. Each of the turbine trips provide turbine protection and resuce the severity of the ensuing transient. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the specified trip settings is required to enhance the overall reliability of the Reactor Protection System.

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

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

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition for increased load events 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 1.6% $\Delta k/k$ is initially required to control the reactivity transient and automatic ESF is assumed to be available.

Technical Specification requirements call for verification that the SHUTDOWN MARGIN is greater than or equal to that which would be required for the MODE 3 low temperature value, 350°F, prior to blocking safety injection on either the P-11 or P-12 permissive interlocks. This assures in the event of an inadvertent opening of two cooldown steam dump valves that adequate shutdown reactivity is available to allow the operator to identify and terminate the event.

With T_{avg} < 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% $\Delta k/k$ SHUTDOWN MARGIN provides adequate protection for this event.

In shutdown MODES 4 and 5 when heat removal is provided by the residual heat removal system, active reactor coolant system volume may be reduced. Increased SHUTDOWN MARGIN requirements when operating under these conditions is provided for high reactor coolant system boron concentrations to ensure sufficient time for operator response in the event of a boron dilution transient.

The SHUTDOWN MARGIN requirements are based upon the limiting conditions described above and are consistent with FSAR safety analysis assumptions.

3/4.1.1.3 BORON DILUTION

A minimum flow rate of at least 3000 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 3000 GPM will circulate an equivalent Reactor Coolant System volume of 12,612 cubic feet in approximately 30 minutes. The reactivity change rate associated with boron reductions will therefore be within the capability for operator recognition and control.

BASES

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on MTC are provided to ensure that the value of this coefficient remains within the limiting conditions assumed for this parameter 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.

It is confirmed by cycle specific neutronic analyses that the value of the MTC at EOC, HZP (All rods in) is greater than the value at EOC, HFP (All rods out), thus assuring that the surveillance at the latter condition is adequate to maintain MTC within safety analysis assumptions.

The surveillance requirements for measurement of the MTC at the beginning and near the end of each 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.

3/4.1.1.5 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 541°F. This limitation is required to ensure 1) the moderator temperature coefficient is within its analyzed temperature range, 2) the protective instrumentation is within its normal operating range, 3) the pressurizer is capable of being in a OPERABLE status with a steam bubble, and 4) the reactor pressure vessel is above its minimum RT_{NDT} temperature. Administrative procedures will be established to ensure the P-12 blocked functions are unblocked before taking the reactor critical.

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, 5) associated heat tracing systems, and 6) an emergency power supply from OPERABLE diesel generators.

BASES

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that nimor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps and safety injection pumps, except the required OPERABLE charging pump, to be inoperable below 152°F, unless the reactor vessel head is removed, provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

THIS PARAGRAPH TO BE SUBMITTED AT A LATER DATE

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

THIS PARAGRAPH TO BE SUBMITTED AT A LATER DATE

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

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The OPERABILITY of boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

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 calculated DNBR in the core at or above design 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 certain hot channel and peaking factors as used in these specifications are as follows:

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

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

The limits on $F_Q(Z)$ and $F^N_{\Delta H}$ for Westinghouse supplied fuel at a core average power of 3411 MWt are 1.97 and 1.48, respectively, which assure consistency with the allowable heat generation rates developed for a core average thermal power of 3391 MWt. The limits on $F_Q(Z)$ and $F^N_{\Delta H}$ for ENC supplied fuel have been established for a core thermal power of 3411 MWt. The limit on $F_Q(Z)$ is 2.10. The limit on $F^N_{\Delta H}$ is 1.49. The analyses supporting the Exxon Nuclear Company limits are valid for an average steam generator tube plugging of up to 10% and a maximum plugging of one or more steam generators of up to 15%. In establishing the limits, a plant system description with improved accuracy was employed during the reflood portion of the LOCA Transient. With respect to the Westinghouse supplied fuel, the minimum projected excess margin to ECCS limits will more than offset the impact of increased steam generator tube plugging.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(Z)$ upper bound envelope is not exceeded during either normal operation or in the event of xenon redistribution following power changes. The $F_Q(Z)$ upper bound envelope is 1.97 times the average fuel rod heat flux for Westinghouse supplied fuel and 2.10 times the average fuel rod heat flux for Exxon Nuclear Company supplied fuel.

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

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor and nuclear enthalpy rise hot channel factor ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2.1, 4.2.2.2, 4.2.3, 4.2.6.1 and 4.2.6.2. This periodic surveillance is sufficient to ensure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps from the group demand position.

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

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

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided conditions a. through d. above are maintained. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. The form of this relaxation for DNBR limits is discussed in Section 2.1.1 of the basis.

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

A measurement error allowance of 4% on $F^N_{\Delta H}$ has been included in the Technical Specification limit so that measured values may be compared directly to the $F^N_{\Delta H}$ limit.

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.

The two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the power by 3 percent from RATED THERMAL POWER for each percent of tilt in excess of 1.0.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters in MODE 1 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 assumptions of the safety analysis and have been analytically demonstrated adequate to maintain design DNBR throughout each analyzed transient. The indicated values of T_{avg}, pressurizer pressure, and flow include allowances for instrument errors.

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. The 12-hour surveillance of the RCS flow measurement is adequate to detect flow degradation. The CHANNEL CALIBRATION performed after refueling ensures the accuracy of the shiftly flow measurement. The total flow is measured after each refueling based on a secondary side calorimetric and measurements of primary loop temperatures.

The limits on pressurizer pressure and T_{avg} in MODES 2 and 3 provide protection against DNB resulting from an uncontrolled rod withdrawal from a subcritical condition. The indicated values of T_{avg} and pressurizer pressure include allowances for instrument errors.

3/4.2.6 ALLOWABLE POWER LEVEL - APL

The power distribution control procedure, PDC-II, manages core power distributions such that Technical Specification limits on $F_Q(Z)$ are not violated during normal operation and limits on MDNBR are not violated during steady-state, load-follow, and anticipated transients. The V(Z) factor given in the Technical Specifications provides the means for predicting the maximum $F_Q(Z)$ distribution anticipated during operation under the PDC-II procedure taking into account the incore measured equilibrium power distribution. A comparison of the maximum $F_Q(Z)$ with the Technical Specification limit determines the power level (APL) below which the Technical Specification limit can be protected by PDC-II. This comparison is done by calculating APL, as defined in specification 3.2.6.

BASES

3/4.3.1 and 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION

The OPERABILITY of the protective and ESF instrumentation systems and interlocks ensure that 1) the associated ESF action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof exceeds its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available for protective and ESF purposes from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundance and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

Protection has been provided for main feedwater system malfunctions in MODES 3 and 4. This protection is required when main feedpumps are aligned to feed steam generators in MODES 3 and 4. The availability of feedwater isolation on high-high steam generator level terminates the addition of cold water to the steam generators in any main feedwater system malfunction. The total volume that can be added to the steam generators by the main feedwater system in MODES 3 and 4 is limited by this safeguards actuation and the fact that feedwater isolation on low T_{avg} setpoint coincident with reactor trip can only be cleared above the low-low steam generator level trip setpoint.

The restrictions associated with bypassing ESF trip functions below either P-11 or P-12 provide protection against an increase in steam flow transient and are consistent with assumptions made in the safety analysis.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the accident analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

BASES

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

3/4.3.3.2 MOVABLE INCORE DETECTORS

The OPERABILITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core. The OPERABILITY of this system is demonstrated by irradiating each detector used and normalizing its respective output.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

3/4.3.3.6 POST-ACCIDENT INSTRUMENTATION

The OPERABILITY of the post-accident instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident.

BASES

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with all reactor coolant loops in operation, and maintain calculated DNBR above the design DNBR value during Condition I and II events. A loss of flow in two loops will cause a reactor trip if operating above P-7 (11 percent of RATED THERMAL POWER) while a loss of flow in one loop will cause a reactor trip if operating above P-8 (31 percent of RATED THERMAL POWER).

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE. Three loops are required to be OPERABLE and to operate if the control rods are capable of withdrawal and the reactor trip breakers are closed. The requirement assures adequate DNBR margin in the event of an uncontrolled rod withdrawal in this mode.

In MODES 4 and 5, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE; this specification requires two RHR loops to be OPERABLE.

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

The restrictions on starting a Reactor Coolant Pump with one or more RCS cold legs less than or equal to 152°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into, or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

BASES

3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE in MODES 4 and 5, an operating RHR loop, connected to the RCS, provides overpressure relief capability. Additionally, if no safety valves are OPERABLE, then all Safety Injection pumps and all but one charging pump will be rendered inoperable to preclude overpressurization due to an inadvertent increase in the RCS inventory.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip set point is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operated relief valves or steam dump valves.

3/4.4.11 RELIEF VALVES

The power operated relief valves (PORVs) operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should the relief valve become inoperable. The electrical power for both the relief valves and the block valves is supplied from an emergency power source to ensure the ability to seal this possible RCS leakage path.

3/4.4.12 REACTOR COOLANT VENT SYSTEM

The Reactor Coolant Vent System is provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. It has been designed to vent a volume of Hydrogen approximately equal to one-half of the Reactor Coolant System volume in one hour at system design pressure and temperature.

The Reactor Coolant Vent System is comprised of the Reactor Vessel head vent system and the pressurizer steam space vent system. Each of these subsystems consists of a single line containing a common manual isolation valve inside containment, splitting into two parallel flow paths. Each flow path provides the design basis venting capacity and contains two 1E DC powered solenoid isolation valves, which will fail closed. This valve configuration/redundancy serves to minimize the probability of inadvertent or irreversible actuation while ensuring that single failure of a remotely-operated vent valve, power supply, or control system does not prevent isolation of the vent path. The pressurizer steam space vent is independent of the PORVs and safety valves and is specifically designed to exhaust gases from the pressurizer in a very high radiation environment. In addition, the OPERABILITY of one Reactor Vessel head vent path and one Pressurizer steam space vent path will ensure that the capability exists to perform this venting function.

The function, capabilities, and testing requirements of the Reactor Coolant Vent System are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirement," November 1980.

The minimum required systems to meet the Specification and not enter into an action statement are one vent path from the Reactor Vessel head and one vent path from the Pressurizer steam space.

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% of its design pressure of 1085 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 17,153,800 lbs/hr which is 117 percent of the total secondary steam flow of 14,674,000 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 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 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:

For 4 loop operation

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

Where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER

V = maximum number of inoperable safety valves per steam line

X = total relieving capacity of all safety valves per steam line in lbs./hours = 4,288,450

Y = maximum relieving capacity of any one safety valve in lbs./hour = 857,690

109 = Power Range Neutron Flux-High Trip Setpoint for 4 loop operation

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 off-site power.

Each electric driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 450 gpm at a pressure of 1065 psig to the entrance of the steam generators. The steam driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 900 gpm at a pressure of 1065 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 acceptance discharge pressures for the auxiliary feedwater pumps are based on a fluid temperature of 60°F. Water density corrections are permitted to allow comparison of test results which vary depending on ambient conditions.

In addition to its safety design function, the AFW system is used to maintain steam generator level during startup (including low power operation). During this time, the system design allows for automatic initiation of the auxiliary feedwater pumps and their related automatic valves in the flow path.

Attachment 4 to AEP:NRC:0916I

REQUEST FOR WITHHOLDING PROPRIETARY INFORMATION

ASSOCIATED WITH RdF RTD SAFETY EVALUATION

IN ATTACHMENT 3